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Maine Yankee

321 OLD FERRY RD. • WISCASSET, ME 04578-4922

February 27, 2009
MN-09-005 RA-09-006

UNITED STATES NUCLEAR REGULATORY COMMISSION
Attention: Document Control Desk
Washington, DC 20555

References:

1. License No. DPR-36 (Docket No. 50-309, 72-030, 71-0465)
2. MYAPC Letter MN-00-004 dated January 13, 2000, Maine Yankee License Termination Plan
3. MYAPC Letter MN-01-023 dated June 1, 2001, Revision 1, Maine Yankee's License Termination Plan
4. MYAPC Letter MN-01-032 dated August 13, 2001, Revision 2, Maine Yankee's License Termination Plan
5. MYAPC Letter MN-02-048 dated October 15, 2002, Revision 3, Maine Yankee's License Termination Plan
6. NRC Letter to MYAPC dated February 28, 2003, Issuance of Amendment No. 168 to Facility Operating License No. DPR-36 -- Maine Yankee Atomic Power Station (TAC No. M8000).
7. MYAPC Letter MN-05-010 dated February 28, 2005, Revision 4, Maine Yankee's License Termination Plan

Subject: **Revision 5, Maine Yankee's License Termination Plan**

In accordance with 10 CFR 50.71(e), Maine Yankee (MY) hereby submits an update to the License Termination Plan (LTP). The updated LTP (Revision 5) continues to demonstrate that the remainder of decommissioning activities: (1) will be performed in accordance with Title 10 Code of Federal Regulations, (2) will not be inimical to the common defense and security or to the health and safety of the public, and (3) will not have a significant effect on the quality of the environment.

This 10 CFR 50.71(e) update[1] contains changes necessary to reflect current information[2] and analyses, approved license amendments, and changes made in accordance with 10 CFR 50.59. The LTP revision history and the update approach to Revision 5 are discussed further below. Attachment 1 provides a summary listing of the key LTP Revision 5 changes. Attachment 2 provides a List of Effective Sections and Attachments (Only those Sections and Attachments requiring update are reflected as "Revision 5". The Sections and Attachments requiring no update remain labeled by their respective revision number. Attachment 3 provides the complete, updated LTP via ~~CD-ROM.~~ No CD attached. Processed Paper copy only JG

1 Also in accordance with LTP Section 1.1 and 1.4.

2 Note that Section 3 of the LTP has not been updated to reflect the current status of dismantlement activities. Dismantlement status was provided via the Final Status Survey (FSS) Reports submitted to the NRC.

Maine Yankee LTP Revision History

Reference 2 transmitted the initial Maine Yankee LTP for NRC review and approval. Reference 3 transmitted LTP Revision 1 and reflected changes in the approach for decommissioning and revised criteria for completion of decommissioning activities. LTP Revision 1 also incorporated MY responses to comments and questions from the State of Maine, Friends of the Coast, and the NRC. Reference 4 transmitted LTP Revision 2 which incorporated additional changes resulting from on-going stakeholder interface, as well as internal MY LTP review and refinement. Reference 5 transmitted LTP Revision 3 which incorporated additional changes addressing NRC Requests for Additional Information (RAI), and other decommissioning updates. Reference 6 provided the NRC SER on LTP Revision 3, and incorporated the NRC approved LTP Revision 3 and associated LTP addenda correspondence[3] into the MY license. Reference 7 transmitted Revision 4 which incorporated changes related to an NRC approved license amendment, changes evaluated and implemented under 10 CFR 50.59 and the LTP addenda correspondence referred to above.

LTP Revision 5 Update Approach

The LTP is part of the FSAR per 10 CFR 50.82 and therefore the 10 CFR 50.71(e) update rule applies. However, the LTP has certain attributes that make it somewhat different from the FSAR and therefore the update rule is applied in a slightly different manner. First, the LTP is inherently a plan and as such is subject to change. The LTP approval process established the special criteria under which the LTP is changed, identifying which changes require prior NRC notification and/or approval. The second element of the decommissioning licensing process that differs from the FSAR is that the fulfillment of the plan is described in additional submittals to the NRC staff in the form of FSS reports (see footnote 2). These reports describe the final implementation of demolition activities and the approach and results of the final status survey, survey unit by survey unit. As such, updates to the plan, following 10 CFR 50.71(e), include those changes to the plan that: (1) have been submitted to the NRC for review and approval or (2) evaluated and implemented under 10 CFR 50.59 and the special criteria related to changes in the LTP.

Finally, it should be understood that Revision 5 does not intend to represent a complete revision of the LTP such that it would reflect current status of plan implementation. Such a revision is considered to beyond the intent of the rule. As such, the following is intended to describe the change rationale and content, on a section by section basis (Attachment 1 provides a detailed

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3 Three addenda letters were submitted to the NRC: (1) MN-02-058, LTP Revision 3 Addenda dated November 21, 2002 - Clarifications and Minor Corrections to Maine Yankee License Termination Plan Revision 3; (2) MN-02-061, dated November 26, 2002, Maine Yankee License Termination Plan, Rev. 3 Addenda and Additional Information Related to the Eberline Model E600 Instrument; (3) MN-02-063, dated December 12, 2002, Update on Forebay Dike Coring Results and Associated Changes to LTP Attachment 2H (LTP Revision 3 Addenda).

listing of changes; changes in Revision 5 are identified by right margin revision bars):

Chapter 1 - Changes were primarily to reflect LTP correspondence and revision history.
Chapter 2 and 3 - No changes
Chapter 4 – Attachment 4C "Remediation Survey – Gamma Scan" and associated references were eliminated under 10 CFR 50.59.
Chapters 5 through 9 – No changes

If you have any questions, please contact us.

Sincerely,

James Connell
Vice President

Attachments
1. Listing of Key Changes - LTP Revision 5
2. List of Effective Sections and Attachments – LTP Revision 5
3. License Termination Plan, Revision 5 (CD-ROM)

cc: Mr. D. R. Lewis, Esq., Shaw Pittman
Mr. J. Hyland, State of Maine
Mr. P. J. Dostie, State of Maine, Division of Health Engineering
Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. J. Goshen, NRC NMSS Project Manager, Decommissioning
Mr. M. Roberts, NRC Region I
Mr. R. Shadis, Friends of the Coast

STATE OF MAINE

Then personally appeared before me, James Connell, who being duly sworn did state that he is the Vice President of Maine Yankee Atomic Power Company, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Maine Yankee Atomic Power Company, and that the statements therein are true to the best of his knowledge and belief.

Notary Public

JOHN P. RZASA, NOTARY PUBLIC
STATE OF MAINE
MY COMMISSION EXPIRES 6/19/2011

Listing of Key Changes - LTP Revision 5

LTP Section	LTP Section Title	Source of LTP Change
1.2 and 1.7	"Operating and Decommissioning History" and "References"	Added MY correspondence submitting the activated concrete proposed changes and NRC Letters issuing Amendment Nos. 170 and 172 to the Maine Yankee Operating License.
1.6 & Preface	Maine Yankee LTP Information Contact	Made contact information generic.
Section 4 Table of Contents	Attachment 4C Remediation Surveys – Gamma Scans - Eliminated	50.59 Screen dated July 19, 2005 – Eliminated since this clarification was not necessary for NRC review of FSS Report No. 1 and 2.
4.2.1	Reference to Attachment 4C – deleted	50.59 Screen dated July 19, 2005 – Eliminated since this clarification was not necessary for NRC review of FSS Report No. 1 and 2.
Attachment 4C	"Remediation Gamma Scans - Eliminated	50.59 Screen dated July 19, 2005 – Eliminated since this clarification was not necessary for NRC review of FSS Report No. 1 and 2.

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Attachment 3

MAINE YANKEE LICENSE TERMINATION PLAN
REVISION 5
(Enclosed CD-ROM)

MAINE YANKEE

PREFACE

LICENSE TERMINATION PLAN REQUIREMENTS - A NON-TECHNICAL SUMMARY

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P.0 LICENSE TERMINATION PLAN REQUIREMENTS - A NON-TECHNICAL SUMMARY

P.1 Introduction

Maine Yankee received feedback from a number of different stakeholders concerning plans for license termination and releasing the site for other uses. These stakeholders include the Maine Department of Environmental Protection, Department of Human Services Division of Health Engineering, the State Nuclear Safety Advisor, the Governor's Technical Advisory Panel, the Maine Yankee Community Advisory Panel, the Environmental Protection Agency, town of Wiscasset officials, Friends of the Coast, and various private individuals.

The feedback has generally indicated a desire for Maine Yankee to go beyond the NRC regulatory requirements (including ALARA) in reducing residual radiation exposure on-site. To that end, in the Preface to the Original License Termination Plan submitted on January 13, 2000, Maine Yankee proposed a site release standard of not more than 10 mrem/year for all pathways, including not more than 4 mrem/year from groundwater sources of drinking water. On April 26, 2000, the Governor of the State of Maine signed into law LD 2688-SP1084, "An Act to Establish Clean-up Standards for Decommissioning Nuclear Facilities." This legislation amended the Maine State definition of Low Level Radioactive Waste to exclude radioactive material remaining at the site of a decommissioned nuclear power plant if the enhanced state standards described in the new law are met. Prior to the passage of this legislation, on April 14, 2000, Maine Yankee had signed an agreement with several Maine groups to support this legislation and to fulfill our mutual intent to reduce the radiological burden at the Maine Yankee site. These groups included "Safe Power for Maine," "Citizens Against Nuclear Trash," "Friends of the Coast - Opposing Nuclear Pollution," and the Town of Wiscasset. The state law and the agreement identified above go beyond the NRC regulatory requirements in reducing residual radioactive contamination remaining on-site at license termination.

This section of the LTP has two purposes:

- To discuss key elements of the LTP while lending perspective with respect to public health and safety and
- To review the steps beyond the NRC regulatory requirements Maine Yankee will implement in being responsive to stakeholder feedback and state legislation. These steps are factored into Sections 1 through 9 of the License Termination Plan.

This Preface of the LTP is not intended for the NRC. Rather it is directed to the wide audience of readers who have a stake or an interest in the ultimate re-use of the Maine Yankee site.

P.2 Background and Definitions

NRC regulations require that decommissioning nuclear facilities clean up residual radioactivity (i.e., plant derived radioactive contamination above natural background radiation) so that the average member of the critical group would receive no more than a 25 millirem (mrem) dose over a year's period of time. This is total dose from any exposure pathway (e.g., drinking water, food, etc.). The enhanced state clean-up standard requires that this residual radioactivity result in not more than 10 mrem/year for all pathways, including not more than 4 mrem/year from groundwater sources of drinking water.

Dose is a measure of exposure to radioactivity. Naturally occurring radioactivity in Maine - i.e., from rock and minerals such as granite or from cosmic radiation - amounts to about 200-300 mrem/year. People are routinely exposed to many other sources of radioactivity.

In addition to the NRC's site release limit, the NRC also requires that the residual radioactivity be ALARA - "As Low As Reasonably Achievable." Using NRC guidance, "reasonably achievable" is determined by the amount of dose reduction achieved compared to the cost of additional dose reduction.

The Environmental Protection Agency (EPA) has also issued site release guidance for facilities other than commercial nuclear power plants. Their criterion is risk-based rather than dose-based. Without accounting for radioactive decay, the EPA calculates a "surrogate" limit of 15 mrem/year which, when decay is accounted for, results in guidance in excess of 30 mrem/year. The EPA also fosters an additional criterion of 4 mrem/year due to groundwater ingestion. The EPA does not have an ALARA standard.

Dose in this range (4 -25 mrem per year above background) is such a low value that it cannot be directly measured, particularly considering that total radiation exposure is the sum of many different exposure pathways such as eating, drinking and direct exposure.

In order to demonstrate compliance with a dose limit, one must convert it to a surrogate value that can be directly measured. This value is called Derived Concentration Guideline Level or "DCGL." The DCGL is a limit for residual radioactive contamination levels in soil, buildings, etc. that, when put into a computer model to account for all exposure pathways, will result in doses not more than a pre-defined limit (e.g., not more than 10 mrem/year for all pathways, including not more than 4 mrem/year from groundwater sources of drinking water).

In order to identify the exposure pathways, one must answer the question, "Who receives the dose ?" The answer is found in regulatory guidance which requires the dose calculations to model the average member of the critical population group. In other words, a hypothetical, conservative scenario is created which includes theoretical individuals who could receive more radiation dose than could be expected for a member of the public.

As a result, Maine Yankee has chosen the so-called resident farmer scenario. In this case, a farmer is resident on the site, obtains drinking and irrigation water from the most contaminated portion of the site, and eats the crops and animals grown from the well water. As discussed in more detail in the LTP, this is an extremely conservative scenario because high quality community water service is readily available, and a resident farmer is unlikely to inhabit the property given the potential for certain commercial uses.

All of the exposure pathways applicable to the resident farmer scenario are considered in the computer model. This model results in the calculation of the DCGL. The DCGL is a value that can be directly measured. For instance, the residual contamination on a building foundation wall may be 15,000 dpm/100 cm^2. The term "dpm" stands for "disintegrations per minute" and is the number of radioactive atoms that decay in a minute. The "100 centimeter squared" provides an area over which the measurement is made. If the DCGL for this building example is below 18,000 dpm/100 cm^2 (e.g., in the Containment Building) then, under "MARSSIM" (see below), we can be assured that radiation exposure due to this portion of the building, combined with the remainder of the site, will be not more than 10 mrem/year for all pathways, including not more than 4 mrem/year for groundwater sources of drinking water.

Once the dose limit is converted to a readily measurable value, the question is "How should one measure it in a widely recognized manner with high confidence that the site meets the limit for release?" To solve this problem, various government agencies including the NRC, the EPA, the Department of Energy and the Department of Defense spent a number of years pooling their resources to come up with a solution. They developed a method that would ensure, on a rigorous statistical basis with a high degree of confidence, that any site areas to be released meet the site release criteria. These methods were published in December, 1997 under the title "Multi-Agency Radiation Survey and Site Investigation Manual," or MARSSIM for short. NRC and EPA recommend, and Maine Yankee has committed to, the use of MARSSIM.

P.3 Relationship Between the LTP and Site Cleanup Levels and Doses

The LTP's primary purpose is to demonstrate compliance with the NRC's annual dose limit of 25 mrem plus ALARA and the enhanced state clean-up levels of not more than 10 mrem/year for all pathways, including not more than 4 mrem/year from groundwater sources of drinking water. Due to conservatisms employed in demonstrating compliance, the ultimate site cleanup level will be lower.

Under NRC guidance and MARSSIM, Maine Yankee assumes that the site and buildings are contaminated. For dose calculation purposes, we further assume the contamination is everywhere at the DCGL limit. (Remember that the DCGL is that measured value that ensures that dose to the average member of the critical group is not more than 10 mrem/year for all pathways including not more than 4 mrem/year from groundwater sources of drinking water.)

In general, remediating higher contamination levels combined with pre-existing low contamination levels will result in actual contamination levels being a medium to small fraction of DCGLs. Recognizing that contamination levels lower than DCGLs translate directly into lower doses, we can also say that dose to the resident farmer will most likely be a medium to small fraction of 10 mrem/year for all pathways, including not more than 4 mrem/year from groundwater sources of drinking water.

In this sense, the LTP and the associated DCGLs are founded on very conservative assumptions only useful for proving prior to decommissioning that Maine Yankee's approach will meet regulatory requirements.

P.4 Additional Information

As discussed at the outset, this Section of the LTP is provided for stakeholders other than the NRC. It is intended to provide a point of reference and perspective on license termination issues associated with public health and safety. Additional information is available through several means: raising questions during meetings of Maine Yankee's Community Advisory Panel; reviewing Maine Yankee's web site at www.maineyankee.com; written correspondence via mail or e-mail; or simply contacting us at:

ISFSI Manager
Maine Yankee Atomic Power Company
321 Old Ferry Road
Wiscasset, Maine 04578
(207) 882-1300

Please feel free to use whatever communications means is available, and we'll do our best to answer your question.

License Termination Plan

Revision 5 February 27, 2009

Submitted by:
Maine Yankee Atomic Power Company

MAINE YANKEE

LTP SECTION 1

GENERAL INFORMATION

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1.0 GENERAL INFORMATION

1.1 Introduction

This License Termination Plan (LTP) has been prepared by the Maine Yankee Atomic Power Company (MYAPC) nuclear power plant located at 321 Old Ferry Road, Wiscasset Maine, 04578. For the location of the plant with respect to population centers see Figure 1-1. The site boundary is defined in MYAPC Defueled Safety Analysis Report (DSAR) Figure 2.1-1. In accordance with requirements of 10 CFR 50.82(a)(9), the LTP has been prepared and submitted as a supplement to the DSAR and is intended to support an application for amendment of license number DPR-36; Docket Number 50-309. An application for amendment of the license has been provided to facilitate authorization/approval of the LTP as required by 10 CFR 50.82(a)(9).

The license condition includes a LTP change process similar to that required for the DSAR. The LTP will be updated in accordance with 10 CFR 50.71(e).

1.2 Operating and Decommissioning History

The plant is owned by a consortium of 10 New England electric utilities representing consumers in Maine, New Hampshire, Vermont, Massachusetts, Connecticut and Rhode Island. It began commercial operation in December 1972 under Atomic Energy Commission Docket No. 50-309, License No. OL-FP DPR-36, and last operated in December 1996 (Certification of cessation of operation under 10 CFR50.82(a)(1) submitted August 7, 1997). Over its lifetime, the plant operated for a total of approximately 16 effective full power years based on its rated thermal power. The Maine Yankee board of directors voted to permanently cease further operation and decommission the plant in August 1997. On August 27, 1997, Maine Yankee submitted the Post Shutdown Decommissioning Activities Report (PSDAR). On November 6, 1997, a public meeting was held in Wiscasset to hear public comments on the PSDAR. On November 3, 1998, Maine Yankee submitted the Site-Specific Decommissioning Cost Estimate along with a PSDAR Update.

On October 20, 1997, Maine Yankee submitted a request to revise the Technical Specifications to reflect the permanently defueled status of the plant. On March 30, 1998, the Nuclear Regulatory Commission (NRC) issued Amendment #161 approving those revised Technical Specifications. This amendment revised the Maine Yankee Technical Specifications to reflect the permanently defueled condition of the plant, and regulatory requirements and operating restrictions to ensure the safe storage of spent fuel.

The Final Safety Analysis Report (FSAR) was revised to reflect the permanently defueled plant condition and was re-titled "Defueled Safety Analysis Report" (DSAR).

The DSAR was submitted to the NRC on February 6, 1998 and has since been revised in accordance with 10 CFR 50.71(e). Additional licensing basis documents were also revised and submitted to reflect the plant's defueled condition (Defueled Security Plan, Fire Protection Plan, QA Plan, Training Plan and Emergency Plan).

On January 13, 2000, Maine Yankee submitted the original version of the LTP in accordance with 10 CFR 50.82(a)(9). This submittal was preceded by meetings with the NRC and other federal, state and local stakeholders. Draft copies of the Maine Yankee LTP had been circulated and docketed to enhance dialogue and encourage feedback. On March 16, 2000, the NRC completed its acceptance review of the LTP and determined that the LTP provides sufficient information for the staff to proceed with its detailed technical review. Accordingly, a public meeting was held at the Wiscasset High School on May 15, 2000 to solicit public comments. On May 17, 2000, the NRC published notice of the license amendment application proposing to authorize the LTP in the Federal Register (65FR31357-31358).

In an effort meet stakeholder expectations that site cleanup be conducted to the highest reasonable standards and beyond current federal regulatory requirements if feasible, Maine Yankee made a commitment in the original LTP preface to achieve a clean up of the site to a dose of less than 10 mrem for all pathways and less than 4 mrem to groundwater pathway. Nevertheless, on April 26, 2000, the Governor of the State of Maine signed into law LD 2688-SP1084 "An Act to Establish Clean-up Standards for Decommissioning Nuclear Facilities." This legislation amended the Maine State definition of Low Level Radioactive Waste to exclude, from that definition, radioactive material remaining at the site of a decommissioned nuclear power plant if the enhanced state standards described in the new law are met. These enhanced state standards include dose-based residual radioactivity limits of 10 mrem/year (mrem/yr) or less for all pathways and 4 mrem/year or less for groundwater drinking sources and other limits for construction demolition debris. Prior to the passage of this legislation, on April 14, 2000, Maine Yankee had signed an agreement with several Maine groups to support this legislation and to fulfill our mutual intent to reduce the radiological burden at the Maine Yankee site. These groups included "Safe Power for Maine," "Citizens Against Nuclear Trash," "Friends of the Coast - Opposing Nuclear Pollution" ("Friends of the Coast"), and the Town of Wiscasset. The implementation of the state law and the agreement identified above are both described in detail in Section 6 of this LTP.

In a letter dated May 9, 2000, the NRC requested that Maine Yankee describe what action it would take in response to the new state legislation. In a letter dated June 8, 2000, Maine Yankee generally explained the expected impact of the newly enacted legislation and indicated that Maine Yankee was continuing a dialogue with state agencies and other stakeholders concerning the end state of the site, verification of cleanup to state standards and other issues.

On June 15, 2000, the Friends of the Coast submitted a petition to intervene and a request for a hearing. On June 16, 2000, the State of Maine submitted a petition to intervene and a request for a hearing or, alternatively, to participate as an interested state. Accordingly, on July 7, 2000, an Atomic Safety and Licensing Board (ASLB) was established. During a telephone conference on July 20, 2000 with the participants in the LTP license amendment proceeding, Maine Yankee stated that it intended to submit a revised LTP addressing a number of new matters and suggested that the proceeding be held in abeyance until the revised LTP is filed. The other participants generally agreed with this suggestion. Accordingly, on July 20, 2000, the ASLB issued an order for, among other things, Maine Yankee to file a revised LTP by October 31, 2000 or on November 1, 2000 submit a status report.

During the summer and fall of 2000, Maine Yankee received over 400 comments on the LTP from a range of stakeholders. Many of these comments led to changes which have been included in Revision 1 to the LTP[1]. In addition, Maine Yankee initiated and participated in two facilitated stakeholder meetings on decommissioning topics including the disposition of above grade concrete. As a result of these meetings, Maine Yankee agreed to remove and dispose of offsite the concrete debris which results from the demolition of buildings above three feet below grade. The effects of this agreement have led to additional changes to dose models, final status survey methodology, ALARA evaluations, and dismantlement activities which have been included in this revised LTP.

On October 31, 2000, Maine Yankee submitted to the NRC a status report including Maine Yankee's current best estimated schedule for submitting the revised LTP and progress in settling outstanding matters with stakeholders. Efforts associated with incorporating the above agreements and stakeholder comments resulted in the call for additional data collection and analysis. Based on these efforts and the desire to continue a responsive dialogue with stakeholders, Maine Yankee estimated that the revised LTP would be submitted to the NRC by April 15, 2001. On January 29, 2001 and April 3, 2001, Maine Yankee submitted status reports updating the Board on Maine Yankee's interactions with stakeholders. In the latter report, Maine Yankee extended the revised LTP submittal schedule to June 1, 2001. Accordingly, on June 1, 2001, Maine Yankee submitted LTP Revision 1.

On October 13, 2000 and again on February 5, 2001, the NRC issued requests for additional information (RAI). On August 8, 2001 (following the issuance of Revision 1 of the LTP on June 1, 2001), Maine Yankee submitted responses to the NRC RAIs of

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1 "Revised LTP" or "original LTP" will be used in the text where needed for clarity; however, in general, "LTP" is intended to mean the revised LTP in all references in this document subsequent to this point.

October 13, 2000 and February 5, 2001. Many of the RAI issues were incorporated, as appropriate, into Revision 1 of the LTP.

On June 8, 2001, Maine Yankee submitted a joint request to the ASLB for a ten-week period for LTP settlement discussions. On July 12, 2001, Maine Yankee provided responses to the State of Maine and Friends of the Coast comments and questions on the LTP. On August 13, 2001, Maine Yankee submitted LTP Revision 2 incorporating many of the remaining NRC, State of Maine and Friends of the Coast issues, as appropriate. On August 31, 2001, the State of Maine, Friends of the Coast, and Maine Yankee reached a Settlement Agreement (SA) related to the ASLB issues. The SA eliminated the need for an ASLB hearing and established a framework for the Parties to resolve the remaining issues. On October, 2, 2001, the ASLB issued an order approving the Settlement Agreement and terminating the proceeding.

One item of the SA was the establishment of a Technical Issue Resolution Panel (TIRP). The TIRP consisted of two members each from the State of Maine and Maine Yankee. The TIRP met several times between September 26 2001 and December 13, 2001. On December 13, 2001 the Team reached consensus on the five issues on it's agenda, and issued a Participant Settlement Agreement. The results of the TIRP consensus have been incorporated in Revision 3 of the LTP.

On December 18, 2001 and January 17, 2002, the NRC issued a further round of RAIs on LTP Revision 2. On March 13, 2002, Maine Yankee responded to the RAIs. As appropriate, the resolution of the RAIs were incorporated in Revision 3 of the LTP. LTP Revision 3 was submitted on October 15, 2002.

Three addenda letters were submitted to the NRC: (1) MN-02-058, LTP Revision 3 Addenda dated November 21, 2002 - Clarifications and Minor Corrections to Maine Yankee License Termination Plan Revision 3; (2) MN-02-061, dated November 26, 2002, Maine Yankee License Termination Plan, Rev. 3 Addenda and Additional Information Related to the Eberline Model E600 Instrument; (3) MN-02-063, dated December 12, 2002, Update on Forebay Dike Coring Results and Associated Changes to LTP Attachment 2H (LTP Revision 3 Addenda).

On February 28, 2003, the NRC issued Amendment 168 to the MY Facility Operating License; approving and incorporating LTP Revision 3, and associated addendum, into the MY License. Maine Yankee provided comments on the Amendment 168 Safety Evaluation in letter MN-03-023, dated May 6, 2003.

On September 11, 2003, Maine Yankee submitted letter MN-03-049 to the NRC proposing a change to the activated concrete DCGL using more realistic activated dose modeling. On February, 18, 2004, NRC issued Amendment No. 170 approving this change.

On March 15, 2004, Maine Yankee submitted letter MN-04-020 requesting an amendment to the facility operating license pursuant to 10 CFR 50.90 and in accordance with the NRC Approved License Termination Plan (LTP) for Maine Yankee, to indicate NRC's approval of the release of the Non-ISFSI site land from the jurisdiction of the license. From March 2004 to July 2005, Maine Yankee submitted supporting final status survey reports, supplements to the amendment and responses to NRC requests for additional information. On September 30, 2005, NRC issued Amendment No. 172 consisting of the unrestricted release of the remaining land under License No. DPR-36 with the exception of the land where the Independent Spent Fuel Storage Installation (ISFSI) is located and a 3.17 acre parcel of land adjacent to the ISFSI.

1.3 Plant Description

The plant is a three-loop pressurized water reactor with a power rating of 2,700 Megawatts thermal. It has a Nuclear Steam Supply System supplied by Asea Brown Boveri/Combustion Engineering. The secondary plant consists of three Asea Brown Boveri turbines, one high pressure and two low pressure, coupled with a 950 MVA Westinghouse electric generator and associated auxiliary systems. The site also includes ancillary facilities used to support normal plant operations. These facilities consist of warehouses, administrative office buildings, security structures, an environmental sampling complex, a substation and a fire protection system.

The plant is located on an 820-acre site in Lincoln County, Wiscasset, Maine as indicated in Figure 1-1. The site boundary is indicated in DSAR Figure 2.1-1. This location is approximately 0.43 miles from the nearest residence and is within 5 miles of the nearest population center, Town of Wiscasset, as shown in Figure 1-1.

1.4 LTP Submittal Change and Early Release of Land

1.4.1 LTP Submittal and Changes

Maine Yankee is submitting this LTP as a supplement to the Defueled Safety Analysis Report. Upon NRC approval, Maine Yankee's license will authorize and require Maine Yankee to implement and maintain in effect all provisions of the approved LTP. This license termination plan describes an acceptable approach for demonstrating compliance with the radiological criteria for unrestricted use, as defined by 10 CFR 20.1402, by meeting a site release criteria of 10 millirem TEDE per year over background (all pathways) and 4 millirem (as distinguishable from background) TEDE per year for groundwater sources of drinking water using appropriate dose modeling methods, pathways and parameters and acceptable final radiation survey methods. The LTP describes dose modeling methods, pathways and parameters which produce derived concentration guideline levels (DCGL's) for a given dose based release criteria. The LTP also describes the final radiation survey methods to demonstrate compliance with the DCGL's. The dose based release criteria used in the LTP is the site release criteria, namely 10 millirem TEDE per year over background (all pathways) and 4 millirem (as distinguished from background) TEDE per year for groundwater sources of drinking water in accordance with state law.[1] While it is understood that NRC may not agree with or adopt this criteria, it is expected that NRC will be confirming that compliance with NRC regulations is being demonstrated by meeting this site release criteria. Maine Yankee will certify in its application for license termination that it has met this site release criteria (10/4) and will at that time request NRC to confirm this certification.

Changes requiring NRC approval will be submitted via application for a license amendment in accordance with 10 CFR 50.90.

Pursuant to license condition 2.B (10) of Maine Yankee's Facility Operating License No. DPR-36, the licensee may make changes to the LTP without prior approval provided the proposed changes do not:

a. Require Commission approval pursuant to 10 CFR 50.59;

b. Violate the requirements of 10 CFR 50.82(a)(6);

c. Reduce the coverage requirements for scan measurements;

d. Increase the radioactivity level, relative to the applicable derived concentration guideline level, at which an investigation occurs; or

e. Increase the probability of making a Type I decision error.

Maine Yankee will submit an updated License Termination Plan in accordance with 10 CFR 50.71(e).

Items a and b of the above LTP change criteria regarding 10 CFR 50.59 and 50.82(a)(6) are established in current regulation. Item c regarding the coverage requirements for scan measurements, is established in LTP Section 5.4.1, Table 5-

[1] LD 2688-SP 1084, "An Act to Establish Clean-up Standards for Decommissioning Nuclear Facilities," enacted on April 26, 2000.

3. Item d regarding investigation levels, sets a limit on the action thresholds that would trigger an investigation. These thresholds are specified in LTP Section 5.6, Table 5-7. Item e limits the probability of releasing a survey unit, which contains residual radioactivity above the release criterion. This probability value is discussed in LTP Section 5.4.2 and 5.8.1.

As appropriate, Maine Yankee will evaluate changes to the LTP using the Data Quality Objective (DQO) process outlined in NUREG-1575, "Multi Agency Radiological Survey and Site Investigation Manual" and/or the considerations described in section 3.2. Changes to the LTP not requiring NRC approval will be submitted as an updated supplement to the DSAR in accordance with 10 CFR 50.71e.

In addition to the above license condition LTP change criteria, Maine Yankee will notify the State of Maine promptly prior to making a change to the LTP that would result in an increase, of any amount, in a Derived Concentration Guideline Level (DCGL) and will request NRC approval if a change to the LTP would result in an increase in a DCGL, as specified in Table 6-11, by more than a factor of two. Note that any DCGL increase is only allowable provided the resulting "Total Annual Dose" remains less than or equal to10 mrem/y and the "Drinking Water" (dose) remains less than or equal to 4 mrem/y (as presented Table 6-11). In other words, the individual contaminated material DCGLs listed in Table 6-11 must always collectively result in a total annual dose of 10 mrem/y or less and a drinking water dose of 4 mrem/y or less. As discussed above, Maine Yankee will certify in its application for license termination that it has met this site release criteria (pursuant to license condition 2.B (10) of Maine Yankee's Facility Operating License).

In the event that Maine Yankee elects to reduce a survey unit's classification as listed in Section 5, i.e., from Class 1 to Class 2 or 3, or from Class 2 to 3, prior notification will be provided to the NRC. Criteria for reclassification is discussed in Section 5.6.4. Maine Yankee will provide the NRC as much early notice of this decision as practical but not less than two weeks. (See Reference 1.7.16.)

1.4.2 Phased Release and License Termination

Maine Yankee will make changes to the site boundary footprints to allow unrestricted release and license termination of parcels of property. The following process will be used for making these changes:

a. Following the completion of LTP activities in a given area, Maine Yankee will provide to the NRC a license amendment request covering the area which it seeks to release from the Part 50 license. This report will contain the information which the NRC needs to make a determination similar to 10 CFR 50.82(a)(11) and will include:

1. A description of the boundaries associated with the area to be released.

2. A statement that the remaining dismantlement activities for the affected area described in the license termination plan have been performed.

3. Final Status Survey (FSS) results for the area. FSS is not required for non-impacted areas.

4. An evaluation of the potential for possible re-contamination of the area and a description of the specific controls established to prevent re-contamination.

5. An evaluation of the impact on the exclusion area for the site lands remaining within the domain of the Part 50 license.

6. An evaluation of the potential combined dose effects on the critical group at license termination as a result of partial releases of land

7. An evaluation of the impact on the following license programs for the site lands remaining within the domain of the Part 50 license: Offsite Dose Calculation Manual (ODCM), Emergency Plan, Security Plan, Fire Protection Plan, QA Plan, Training Plan, DSAR, and Post Shutdown Decommissioning Activities Report (PSDAR).

8. A no significant hazards determination evaluation.

This process has been informed by NRC Regulatory Issue Summary 2000-19 "Partial Release of Reactor Site for Unrestricted Use Before NRC Approval of the License Termination Plan."

Upon satisfactory NRC review, the NRC will provide a license amendment to Maine Yankee that the NRC has made the required 10 CFR 50.82(a)(11) and 50.91 determinations regarding the area to be released from the Part 50 license and that the area is henceforth released from the Part 50 license. This license amendment will carry the same authority as that associated with terminating a license under 10 CFR 50.82(a)(11).

b. Once an area is so released, it is understood that the NRC will not require additional surveys or decontamination of these areas by Maine Yankee in response to future NRC criteria or standards, new information or third party survey results, unless, similar to 10 CFR 20.1401(c), the NRC determines that the criteria of 10 CFR Part 20, Subpart E were not met and residual activity remaining at the site could result in significant threat to public health and safety. With regard to each release, Maine Yankee will work with the NRC and the State of Maine in facilitating confirmatory surveys.

c. Maine Yankee anticipates a three-phased release of land from the operating license:

1. Approximately 641 acres of land associated with the Eaton Farms and the land north of Ferry Road. A portion of this land will be transferred for the purpose of an environmental center in accordance with the FERC rate case settlement.

   Reference: Maine Yankee to USNRC letters dated August 16, 2001 (MN–01-034) Early Release of Backlands (Combined), Proposed Change No. 211, Supplement No. 1, and November 19, 2001 (MN-01-044) same subject, Proposed Change No. 211, Supplement No. 2.

   Approval: The NRC provided approval of the subject request for release of site lands by issuance of the license amendment granted by the NRC letter to Maine Yankee, dated July 30, 2002, Issuance of Amendment No. 167.

2. The remainder of the site not associated with the ISFSI

   Reference: Maine Yankee to USNRC letter dated March 15, 2004 (MN-04-020) "Release of Non-ISFSI Site Land" as supplemented by letters dated September 2, 2004 and May 16, 2005.

Approval: NRC letter dated September 30, 2005, Issuance of Amendment No. 172.

3. The portion of the site associated with the ISFSI

1.5 Plan Description

1.5.1 General Information

This section summarizes each of the seven (7) LTP sections required by 10 CFR 50.82(a)(9)(ii).

1.5.2 Site Characterization

Section 2 summarizes the radiological surveys that have been conducted to characterize the nature and extent of contamination at Maine Yankee.

A site radiological characterization was performed to support decommissioning planning during November 1997 through March 1998. This resulted in GTS Duratek's "Characterization Survey Report for the Maine Yankee Atomic Power Plant." Following the initial characterization effort, additional data was required and collected (referred to as "continuing characterization), as discussed in Section 2.1. The additional ("continuing") characterization will continue to be performed as required during the term of the decommissioning project. The site characterization results have been and will be used to identify areas of the site that are likely to require remediation, to plan remediation strategies, and to support final status survey and dose assessment activities.

1.5.3 Identification of Remaining Site Dismantlement Activities

Section 3 presents the sequence of dismantlement and decontamination (D&D) activities for the remaining systems, structures, and components at Maine Yankee.

The overall project schedule identifies the remaining site dismantlement activities. These activities include: (1) the removal of structures to increase the free area needed for large vehicles and equipment; (2) commodity removal; (3) decontamination and remediation; (4) movement of spent fuel to dry storage; and (5) demolition of structures to three feet below grade. The extent to which these activities are expected to be conducted under 10 CFR 50.59 is described. The final state of the site, including any underground remnants, is also described.

The strategies for disposal of waste generated during decommissioning are discussed including the disposition of the materials from above grade structures which will be demolished. These strategies include the removal of radioactive material from the site in order to meet the radiological release criteria of 10 CFR 20.1402 and the state clean-up standards. These state clean-up standards specify, among other things, that any construction demolition debris (CDD), including concrete, disposed of at the site meets the limits specified in Table 1 in the 1974 United States Atomic Energy Commission (AEC) Regulatory Guide 1.86. However, Maine Yankee does not expect to dispose of CDD on site.

This section also includes: estimates of the quantity of radioactive material to be released; control mechanisms; and radioactive waste characterization.

A detailed description of the coordination of activities, requirements, permits and licenses covered by other regulatory agencies is included. These activities, requirements, permits and licenses include Comprehensive Environmental Response, Compensation and Liabilities Act (CERCLA), Resource Conservation and Recovery Act (RCRA), Site Location of Development Permitting, Natural Resources Protection Act (NRPA), Solid Waste Storage and Disposal Permits, Hazardous Waste Storage and Disposal Permits, National Pollution Discharge Elimination System (NPDES) Permits, Waste Discharge Licensing, Tank Closure Certification, Stormwater Management, Erosion and Sedimentation Control, Asbestos and PCB characterization and remediation, Noise Regulations, Air Emissions License, etc. These efforts involve coordination between Maine Yankee and other stakeholders including: the Maine Department of Environmental Protection, the Maine Department of Human Services including the State Nuclear Inspectors, the Governor's Nuclear Safety Advisor, the Governor's Technical Advisory Panel, the Advisory Committee on Radiation and Nuclear Waste, etc. In addition to describing the coordination of the efforts described above, this section of the LTP also describes the various agreements between Maine Yankee and the State of Maine and other parties.

For the purpose of this LTP, it is assumed that the installation and operation of an Independent Spent Fuel Storage Installation will be conducted, separate from the LTP, under a general license which has already been issued in accordance with 10 CFR 72.210. However, the decommissioning of the ISFSI is described in this section. If Maine Yankee submits an application for a 10 CFR Part 72 specific license, this LTP will be revised to eliminate from its scope the decommissioning of the ISFSI.

1.5.4 Remediation Plans

The methods used to reduce the levels of radioactivity to meet the radiological release criteria of 10 CFR 20.1402 (Radiological Criteria for Unrestricted Use) and the enhanced state cleanup standards are described in Section 4. The calculations used to verify that the residual activity levels have been reduced to levels that are as low as reasonably achievable (ALARA) are presented. These calculations, and the applied methodology generally conform to the guidance provided in Draft Regulatory Guide DG-4006 or as superceded by NUREG-1727, "NMSS Decommissioning Standard Review Plan (SRP) [Demonstrating Compliance with the Radiological Criteria for License Termination]."

1.5.5 Final Status Survey (FSS)

Section 5 of this LTP describes the methods that will be used by Maine Yankee to demonstrate that residual contamination levels at the plant site have been reduced to levels below the site release criteria. The derived concentration guideline (DCGL) is calculated in Section 6 of this LTP and represents the residual contamination levels that will result in a Total Effective Dose Equivalent (TEDE) to the average member of the critical population group that is less than 25 mrem per year in accordance with the radiological release criteria of 10 CFR 20.1402 and less than the enhanced state clean-up standards of 10 mrem per year from all pathways and 4 mrem per year from groundwater sources of drinking water. The methods for conducting the final status survey generally follow the guidance in Draft Regulatory Guide 4006 or as superceded by the Standard Review Plan (SRP). NUREG-1575 (Multi-Agency Radiation Survey and Site Investigation Manual [MARSSIM]) is also used to the extent it is referenced in DG-4006 as appropriate. Additional sections of NUREG-1575 are followed as required for specific applications. The FSS plan describes methodology for the division of the site into survey units, the classification of survey areas, and the requirement that all survey units meet the DCGL with a 95% confidence level. Survey areas have been classified. These survey areas will be divided into survey units as work progresses. Management controls over all aspects of the project are discussed in detail, including quality assurance, data processing, and final status survey reports.

1.5.6 Compliance With the Specified Radiological Criteria for License Termination

Section 6 of the LTP describes the methods used for conducting a dose assessment to develop the DCGLs for demonstrating compliance with the unrestricted use criteria in Subpart E of 10 CFR 20 and the enhanced state clean-up standards established by State of Maine Public Law - LD 2688-SP 1084.

10 CFR 20.1402, "Radiological Criteria for Unrestricted Use," allows termination/amendment of a license and release of a site for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of a critical group that does not exceed 25 mrem per year and the residual radioactivity has been reduced to levels that are ALARA. The enhanced state cleanup standards require that the residual radioactivity distinguishable from background radiation will result in a total effective dose equivalent to an average member of a critical group not more than 10 mrem/year for all pathways and 4 mrem/year for groundwater sources of drinking water. In addition, the enhanced state cleanup standards require that any construction demolition debris, including concrete, disposed of at the site meet the limits of Table 1 in the 1974 AEC Regulatory Guide 1.86.

1.5.7 Update of the Site-Specific Decommissioning Costs

Section 7 provides an updated estimate of remaining decommissioning costs and a comparison of these estimated costs with the present funds set aside for decommissioning. A site-specific decommissioning cost analysis was prepared by TLG Services in October of 1997. Subsequent to that, a revision to the decommissioning cost estimate was presented in the MYAPC Site Specific Decommissioning Cost Estimate, dated November 3, 1998. As decommissioning activities are initiated and completed, the actual costs are compared against the estimates previously submitted.

1.5.8 Supplement to the Environmental Report

Section 8 satisfies the requirements stated in:

a. 10 CFR 50.82(a)(9)(ii)(G)
A supplement to the Environmental Report pursuant to 51.53 describing any new information or significant environmental change associated with the licensee's proposed termination activities.

b. 10 CFR 51.53(d)
Post operating license stage. Each applicant for a license amendment authorizing decommissioning activities for a production or utilization facility either for unrestricted use or based on continuing use restrictions applicable to the site; and each

> applicant for a license amendment approving a license termination plan or decommissioning plan under §§50.82 of this chapter either for unrestricted use or based on continuing use restrictions applicable to the site; and each applicant for a license or license amendment to store spent fuel at a nuclear power reactor after expiration of the operating license for the nuclear power reactor shall submit with its application the number of copies, as specified in §§51.55, of a separate document, entitled "Supplement to Applicant's Environmental Report -- Post Operating License Stage," which will update "Applicant's Environmental Report -- Operating License Stage," as appropriate, to reflect any new information or significant environmental change associated with the applicant's proposed decommissioning activities or with the applicant's proposed activities with respect to the planned storage of spent fuel. Unless otherwise required by the Commission, in accordance with the generic determination in §§51.23(a) and the provisions in §§51.23(b), the applicant shall only address the environmental impact of spent fuel storage for the term of the license applied for. The "Supplement to Applicant's Environmental Report -- Post Operating License Stage" may incorporate by reference any information contained in "Applicants Environmental Report -- Construction Permit Stage.

The purpose of Section 8 of the LTP is to upgrade the Maine Yankee Environmental Report with any new information or significant environmental change associated with Maine Yankee's proposed decommissioning/license termination activities. This section of the LTP constitutes a supplement to Maine Yankee's Environmental Report pursuant to 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G). In October, 1970, Maine Yankee submitted to the US Atomic Energy Commission (AEC: NRC's predecessor) its Environmental Report, which was further appended in February 1971 with supplementary information. On April 19, 1972, Maine Yankee submitted to the AEC a "Supplement to Environmental Report." It is this latest supplement which is being updated by this LTP section pursuant to the above regulations. On July 1972 the AEC issued the Final Environmental Statement related to the operation of Maine Yankee Atomic Power Station.

Any identified new information or significant environmental change associated with Maine Yankee's proposed decommissioning/license termination activities has been evaluated to determine whether it is bounded by the site-specific decommissioning activities described in Maine Yankee's PSDAR or AEC's Final Environmental Statement. Pursuant to 10 CFR 51.53, this supplement identifies any changes in Maine Yankee's decommissioning activities as previously identified in revision of its submittal, and provides the reasons for concluding that the impacts associated with those changes remain bounded by the Final Generic Environmental Impact Report Statement (FGEIS), NUREG-0586.

1.5.9 Special Agreement With Friends of the Coast - Opposing Nuclear Pollution

a. As a result of its review of the draft revised LTP, Friends of the Coast raised questions regarding the characterization of radioactivity deposition in off-site marine sediment. The plant derived activity is the result of licensed plant effluent releases offsite into the intertidal zone surrounding Bailey Point. A separate agreement was reached between Maine Yankee and Friends of the Coast to conduct a special marine sediment study in the intertidal zone areas with the overall purpose of enhancing public confidence in the decommissioning process. The key elements of this agreement, "Maine Yankee Decommissioning Supplementary Radiological Characterization and Analysis," dated May 31, 2001, are described in this section. The full text of the agreement is included as Attachment 1-A to this section.

b. It is recognized that the intertidal zone, beyond the site boundary (per the Maine Yankee DSAR Section 2.1 and DSAR Figure 2.1-1), is an area subject to the periodic discharge of low levels of radioactive effluents, released under the plant's operating license per the regulations governing off-site releases, monitoring, dose assessment, sampling, and reporting [i.e., 10 CFR Part 20, Subpart D, Part 50 Appendix I, and 10 CFR 50.36a(2)]. These discharges have been made and evaluated in accordance with the Offsite Dose Calculation Manual and the Radiological Effluent Monitoring Program which are the principal site administrative programs that implement the above requirements. Because this intertidal zone area is beyond the site boundary, addressed by regulations associated with the Part 50 plant license, and involve dose commitment to the public already assessed by these programs and regulations, the area is not included within the scope of the LTP.

c. Regardless of regulatory considerations, Maine Yankee recognizes the community interest in future potential public uses of this area. Although all measurements to date have identified intertidal zone levels of radioactivity well below that allowed to be left on-site, Maine Yankee acknowledges a public benefit in enhanced confidence that can be achieved by additional radiological characterization of the intertidal zone near the end of decommissioning.

d. Per the subject agreement, Maine Yankee will work with Friends of the Coast to contract a radiological survey to characterize the intertidal zone (which is defined in the agreement). This survey is distinct from and in addition to that formerly agreed upon in the partial settlement of the FERC rate case settlement which also provides for a survey of off-site marine sediment (Reference 1.7.12). The intertidal zone characterization will include the "non-affected" Eaton Farm location as well as Bailey Point (to an agreed point, south of Ferry Road).

e. The methods and protocols used in the survey are discussed in the agreement. Dose pathways associated with the intertidal zone, considering current and future uses, will be identified and agreed upon between Maine Yankee and Friends of the Coast. The characterization results and dose assessment will be reported in a form to allow comparison to appropriate on-site DCGLs established in the LTP and to the resident farmer dose. Based on prior sampling in these areas, Maine Yankee anticipates that the future surveys will report intertidal zone activities and dose levels that are well below federal and state limits for site decommissioning.

f. Maine Yankee and Friends of the Coast will define the survey scope by the end of 2001 and implement the survey following final liquid discharge from spend fuel pool operations (currently planned for late in the first quarter of 2003).

g. Results of the characterization will be reported to Maine Yankee and Friends of the Coast. The written report will be publicly available, and Friends of the Coast will receive sufficient copies to disseminate to interested parties and members of the public who request copies.

1.6 Maine Yankee LTP Information Contact

For information or comments regarding the Maine Yankee License Termination Plan, please contact the following party:

ISFSI Manager
Maine Yankee Atomic Power Company
321 Old Ferry Road
Wiscasset, Maine 04578
(207) 882-1300

1.7 References

1.7.1 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities".

1.7.2 NUREG-1496, Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities.

1.7.3 Maine Yankee Environmental Report, dated October 1970

1.7.4 "Final Environmental Statement Related To Operation of MY Atomic Power Station," dated July 1972.

1.7.5 Supplement One to the Maine Yankee Environmental Report, dated April 19, 1972.

1.7.6 NRC Regulatory Issue Summary 2000-19, "Partial Release of Site for Unrestricted Use Before NRC Approval of the License Termination Plan"

1.7.7 GTS Duratek, "Characterization Survey Report for the Maine Yankee Atomic Power Plant," Volumes 1-9, 1998 (ICS).

1.7.8 NUREG-1727 "NMSS Decommissioning Standard Review Plan," September 15, 2000.

1.7.9 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM), Revision 1 (June 2001)

1.7.10 AEC Regulatory Guide 1.86

1.7.11 MYAPC Site Specific Decommissioning Cost Estimate, November 3, 1998

1.7.12 June 1, 1999 Federal Energy Regulatory Commission approval of rate case settlement agreement, Docket Nos. ER98-570-000, EL98-13-000, and EL98-14-00

1.7.13 Post Shutdown Decommissioning Activities Report, Maine Yankee letter to the NRC, MN-97-99, dated August 27, 1997.

1.7.14 MYAPC Defueled Safety Analysis Report (DSAR)

1.7.15 State of Maine Public Law LD 2688-SP1084 "An Act to Establish Clean-up Standards for Decommissioning Nuclear Facilities", April 26, 2000

1.7.16 NRC letter to Maine Yankee, dated August 23, 2002, "Maine Yankee Atomic Power Station re: License Termination Plan Issue" (dealing with survey unit reclassification and the PAB test pit issue).

1.7.17 NRC letter to Maine Yankee, dated July 30, 2002, Issuance of Amendment No. 167, license amendment approving partial release of site lands.

1.7.18 Maine Yankee letter to the NRC, MN-00-004 dated January 13, 2000, Maine Yankee License Termination Plan

1.7.19 Maine Yankee letter to the NRC, MN-01-023 dated June 1, 2001, Revision 1, Maine Yankee's License Termination Plan

1.7.20 Maine Yankee letter to the NRC, MN-01-032 dated August 13, 2001, Revision 2, Maine Yankee's License Termination Plan

1.7.21 NRC Letter to Maine Yankee, dated December 18, 2001, Request for Additional Information (RAI) for Maine Yankee Atomic Power Station License Termination Plan (TAC No. MA8000).

1.7.22 NRC Letter to Maine Yankee, dated January 17, 2002, Request for Additional Information (RAI) for Maine Yankee Atomic Power Station License Termination Plan (TAC No. MA8000).

1.7.23 Maine Yankee letter to the NRC, MN-02-011 dated March 13, 2002, Response to NRC Request(s) for Additional Information for Maine Yankee Atomic Power Station License Termination Plan

1.7.24 MYAPC Letter MN-02-048 dated October 15, 2021, Revision 3, Maine Yankee's License Termination Plan

1.7.25 NRC Letter to MYAPC dated February 28, 2003, Issuance of Amendment No. 168 to Facility Operating License No. DPR-36 -- Maine Yankee Atomic Power Station (TAC No. M8000).

1.7.26 MYAPC Letter MN-0-023 dated May 6, 2003, Maine Yankee's Comments on NRC's Safety Evaluation on the Maine Yankee License Termination Plan

1.7.27 Maine Yankee Letter to the NRC, MN-02-058, LTP Revision 3 Addenda dated November 21, 2002 - Clarifications and Minor Corrections to Maine Yankee License Termination Plan Revision 3

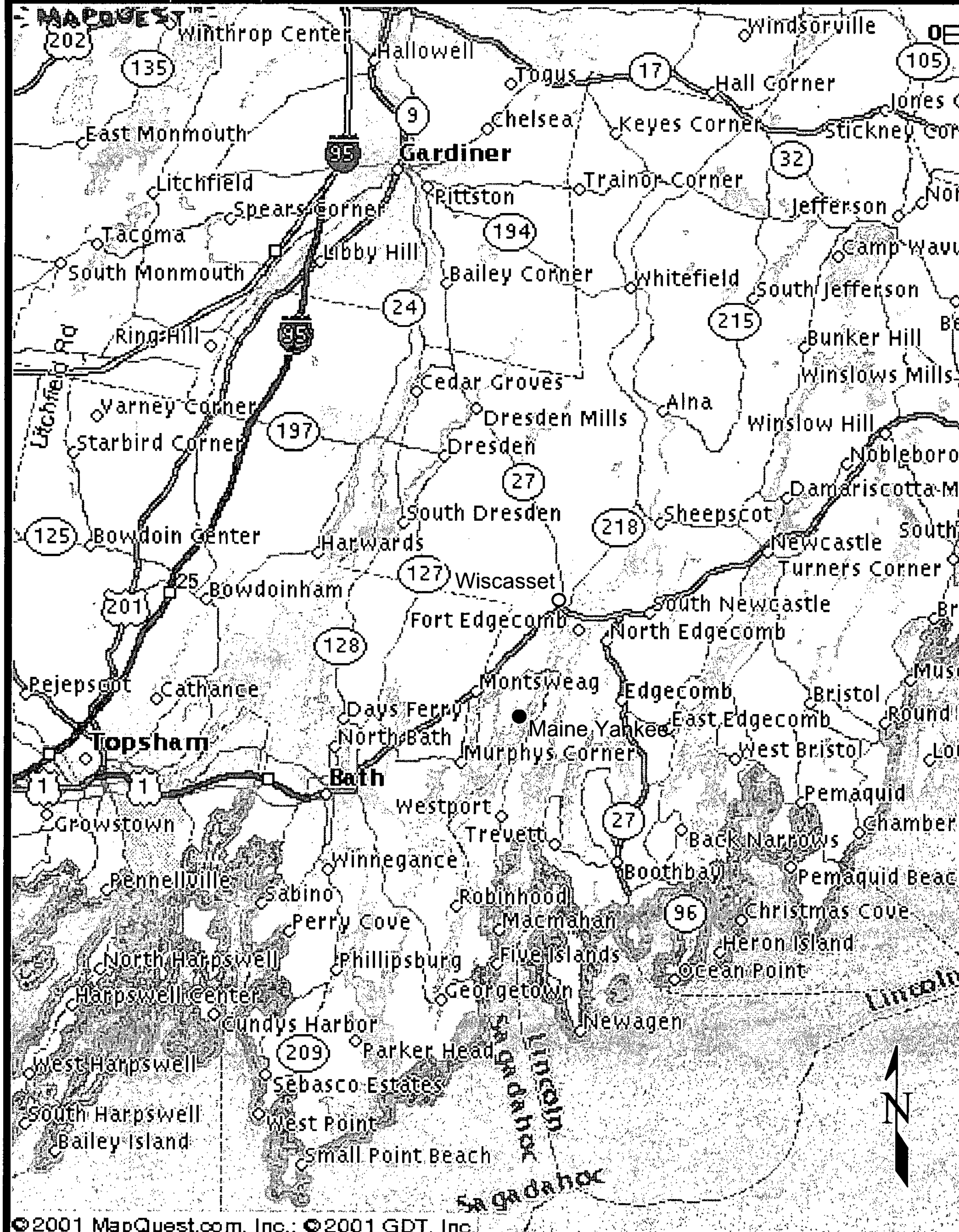
1.7.28 MN-02-061, dated November 26, 2002, Maine Yankee License Termination Plan, Rev. 3 Addenda and Additional Information Related to the Eberline Model E600 Instrument

1.7.29 MN-02-063, dated December 12, 2002, Update on Forebay Dike Coring Results and Associated Changes to LTP Attachment 2H (LTP Revision 3 Addenda).

1.7.30 Maine Yankee letter to NRC (MN-03-049), dated September 11, 2003, Proposed Change: Revised Activated Concrete DCGL and More Realistic Activated Concrete Dose Modeling - License Condition 2.B.(10), License Termination

1.7.31 NRC letter to Maine Yankee, dated February 18, 2004, Issuance of Amendment No. 170 to Facility Operating License No. DPR-36 - Maine Yankee Atomic Power Station (TAC No. M8000). Activated Concrete

1.7.31 NRC letter to Maine Yankee, dated September 30, 2005, Issuance of Amendment No. 172 to Facility Operating License No. DPR-36 - Maine Yankee Atomic Power Station (TAC No. M8000) Unrestricted Release



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Location Of Population Centers With Respect To Location Of Maine Yankee

Figure 1-1

ATTACHMENT 1A

Maine Yankee Decommissioning Supplementary Radiological Characterization and Analysis Agreement

Maine Yankee Decommissioning Supplementary Radiological Characterization and Analysis Agreement

Parties:

This is an agreement between Maine Yankee Atomic Power Company (Maine Yankee) and Friends of the Coast – Opposing Nuclear Pollution (Friends of the Coast).

Purpose:

The purpose of this agreement is to enhance public confidence in the decommissioning process through an independent, professional, comprehensive and scientifically valid radiological survey of the intertidal area adjacent to the Maine Yankee site.

Background:

Maine Yankee and Friends of the Coast agree that Maine Yankee has been lawfully allowed to discharge low levels of radioactive effluents through its licensed pathways. With that understanding, both parties entered into an agreement (copy attached) as a partial settlement of the 1998 FERC rate case, which included provisions for a survey of off-site marine sediments. The present agreement is in addition to the "FERC agreement" and supplements the License Termination Plan by explicitly recognizing, for the purposes of this agreement, the intertidal zone (defined below) as a separate and distinct element of an elective offsite survey

Substance:

Maine Yankee agrees to contract a radiological characterization of the intertidal zone (the present "supplemental agreement") supplementing and in addition to the radiological survey of offsite marine sediment (per the "FERC agreement). For purposes of economy and efficiency, Maine Yankee will seek a single contractor for both the offsite marine sediment survey and the intertidal zone survey through a single request for proposal (RFP). Nothing in this "supplementary agreement" alters the previous "FERC agreement".

The intertidal zone is that offsite area that lies between the site boundary (as described in the Maine Yankee license basis and the License Termination Plan) and the mean low tide mark of adjacent waters (or an outer bound drawn 100 feet from the high tide mark, whichever is closer). The extent of the intertidal zone to be characterized shall include the designated "non-affected" Eaton Farm location as well as Bailey Point (to an agreed upon point south of Ferry Road).

Dose pathways associated with the intertidal zone current and potential future uses will be identified and agreed upon between Maine Yankee and Friends of the Coast. Characterization results will be used to calculate an incremental intertidal zone dose which may be compared to the limiting "resident farmer" dose calculations in the License Termination Plan. Characterization results will also be reported in a form allowing comparison to on-site DCGLs (e.g., soil) in the License Termination Plan.

<u>Methods and Media</u>:

The intertidal zone characterization will be conducted using agreed upon methods and protocols. Upon request, Maine Yankee and Friends of the Coast will observe traditional split sampling protocols with interested parties.

The characterization will be accomplished via:

- Sampling and isotopic analysis of disturbed and undisturbed intertidal zone soils/sediments,

- Sampling and isotopic analysis of flora and fauna that may reasonably be considered contributors to an intertidal zone pathway dose (e.g., seaweed, shellfish, etc.), and

- Selected gamma scan employing high efficiency (e.g., sodium iodide) detectors, or best practical means, for the purpose of identifying discrete or "hot" particles.

<u>Conditions</u>:

Maine Yankee and Friends of the Coast will work together to define an RFP for a sampling and analysis plan for the intertidal zone, identify qualified independent contractors to receive the RFP, and select a contractor based on the bids received. Maine Yankee reserves the right to: 1) establish a reasonable ceiling on the cost of the supplemental study consistent with accomplishing the purposes of the study and re-bid as necessary to satisfy that constraint, and 2) void this agreement should issues associated with the intertidal zone, as the intertidal zone is defined in this agreement, become admissible contentions before the ASLB.

Maine Yankee and Friends of the Coast agree to develop the RFP by 12/31/2001 and implement the study following final liquid discharge from spent fuel pool operations (approximately 3/2003).

This agreement, if finalized in sufficient time, will be included in the revised License Termination Plan as an attachment to or in Section I and referenced wherever else Maine Yankee deems appropriate. If the agreement is not finalized before submittal of the revised License Termination Plan, a statement of intent will be placed in Section I and a later License Termination Plan supplement will provide the agreement when finalized.

If hot particles that would exceed remediation thresholds on-site are discovered in the "supplemental characterization", hot particle remediation will be undertaken following on-site methods and protocols.

Results of the "supplemental characterization" will be reported to Maine Yankee and Friends of the Coast. The written report will be publicly available and Friends of the Coast will receive sufficient copies to disseminate to interested parties and members of the public who request copies.

Friends of the Coast, assisted by Maine Yankee, will provide an annotated bibliography of historical records, studies, etc. to be included as an appendix in the "supplemental study" report.

Agreed by:

Original Signed by Wayne Norton for May 31, 2001

Maine Yankee Date

Original Signed by Ray Shadis for May 31, 2001

Friends of the Coast Date

paragraph (D), taken together with any overages funded by Maine Yankee under Part II-B(5)(B), exceed a total of $10 million.

(E) For purposes of this Part II-B(5), except as otherwise noted, all numbers shall be calculated in mid-1998 dollars (unless, in the alternative, the Settling Parties agree to nominal dollar calculations; for such purposes, the Incentive Budget is $ 488.8 million, utilizing a 3.8% escalation rate). Reconciliation of any amounts to, or to be paid by, Maine Yankee or ratepayers shall be calculated within sixty (60) days of the NRC license termination or site release date (*i.e.*, the date the NRC approves license termination or release of the site applicable to the balance of the plant site other than the ISFSI), regardless of whether such date occurs before, on, or after December 31, 2004. The terms and timing of flow-through of any such benefits or additional payments shall be determined by the Commission pursuant to proceedings commenced by Maine Yankee, and Maine Yankee shall commence such proceedings as promptly as possible after calculating such reconciliation.

C. **Site Restoration Issues.**

(1) Maine Yankee and Friends of the Coast agree that Maine Yankee has been lawfully allowed to discharge low levels of radioactive waste emissions through its licensed pathways. Maine Yankee and Friends of the Coast agree to use their best efforts to reach agreement on thresholds of radiation which would be expected to be in the environment as a result of discharges at licensed limits, and to educate the public through the Community Action Panel on those expectations. Maine Yankee agrees to work with Friends of the Coast to develop a request for proposal ("RFP") for competitive bids for an environmental field survey of off-site marine sediments and a full spectrum analysis following the presumed pattern of dispersal from the Maine Yankee outflow. The RFP

will request that bidders consider use of the computer modeling of projected disposition developed by Professor Thomas Hess (University of Maine at Orono), if possible. The purpose of this survey would be to develop an isotopic picture of how licensed discharges are distributed and accumulated in the environment. Maine Yankee and Friends of the Coast agree that this study will be performed for the sole purpose of measuring and developing such an isotopic picture, and the study will compare the findings to the expected levels of radiation. Maine Yankee shall pay for the costs of the study as part of its decommissioning cost. Maine Yankee and Friends of the Coast agree that the price Maine Yankee pays shall not exceed $165,000, and that the specific scope of the study shall be configured so that the costs of the study do not exceed $165,000.

(2) Maine Yankee agrees to investigate with environmental organizations that are tax exempt under Section 501(c)(3) of the Internal Revenue Code, including, without limitation, colleges and universities, the development of an environmental center at the location of the so-called "Eaton Farm" property on the Maine Yankee site in Wiscasset, Maine. Maine Yankee agrees to consult with Friends of the Coast regarding organizations from which to solicit proposals. Maine Yankee agrees to convey to such an organization the Eaton Farm property, comprised of approximately 200±acres. Such donation will be provided for the purpose of creating a nature preserve and an environmental education center, and to provide public access of coastal lands in the mid-coast region of Maine. The center shall foster stewardship of the Sheepscot estuarine environment and provide a center for dialogue on environmental policy issues. Prior to making its decision and donating property, Maine Yankee agrees to seek the input of the so-called Community Advisory Panel regarding the donation. Maine Yankee's

MAINE YANKEE

LTP SECTION 2

SITE CHARACTERIZATION

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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, an overall 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 all phases of 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 initial 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 initial characterization results (ICS[1]) were provided to MY in the "Characterization Survey Report for the MY Atomic Power Plant," developed by GTS Duratek. After review of this initial characterization report, it was determined that 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, which is generally referred to as Continuing Characterization Surveys (CCS), is discussed in Section 2.5. As additional data is required (such as concrete cores, forebay sediment, etc.), characterization samples will be obtained; thus, CCS is an ongoing activity and is included as part of the FSS process.

This section summarizes the key findings of the HSA and characterization survey results, as supplemented by continuing characterization. The initial characterization report and the detailed results of continuing characterization, are maintained at the MY site and are available for NRC review. Data from the CCS effort, due to its ongoing nature, is filed with the appropriate characterization package associated with the system, structure, component, or area being surveyed (or sampled). These packages are maintained in the Plant Technical File System. The level of detail provided in this summary demonstrates that the overall 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

[1] "ICS," as used in the LTP refers to the initial characterization performed by GTS Duratek, as documented in the "Characterization Survey Report for the MYAPP," 1998. It may also simply be referred to as the "GTS Duratek report."

status of the site, MY also interviewed terminating employees for any recollection of such events.

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, as supplemented, is a compilation of the approximately 140 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.
18. Two large volume spills in the Containment Spray Building

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	Estimated Residual Activity
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.[2]	6 mCi
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.	5.9 mCi
Silt spreading area	1992, 1993 Outages	Land adjacent to and south of ballfield.	1250 ft^3	Residual contamination evaluated and entered into	12 µCi

[2] 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 overall 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 of 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, or transferred to the ISFSI. 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 contained approximately 33,660 gallons of slightly contaminated water. An additional 320,000 gallons added to the refueling cavity for shielding during reactor component removal, have been processed as radwaste.

MY does not expect any primary systems to remain after decommissioning. It is expected that the diffuser will remain in place. Characterization of the diffuser is described in Section 2.5.3 and Attachment 2H. Demolition of structures to 3 feet below grade will remove the majority of embedded or buried piping. Remaining embedded or buried piping will be classified and surveyed in accordance with Sections 2 and 5.

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 in the partial site release applications (Maine Yankee Letters dated August 16, 2001 (MN-01-034) and November 19, 2001 (MN-01-044) Early Release of Backlands (Combined) Proposed Change 211, Supplements 1 and 2 respectively). The NRC granted the request license amendment in its letter to Maine Yankee, dated July 30, 2002. (See Attachment 2A and References, Section 2.7.)

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 includes 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 are 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 continues to play a part in the removal of various components and building materials. Section 3.6 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 MDEP has been delegated authority, by the EPA, to administer the National Pollutant Discharge Elimination System (NPDES) permit program as authorized by the Clean Water Act. Maine Yankee maintained an NPDES permit during operation.

2.3 Site Characterization Survey Methods

As discussed in Section 2.1, the site's initial characterization survey work (ICS) was performed by GTS Duratek and its subcontractor. Continuing Characterization Surveys (CCS) were, and continue to be, performed by Maine Yankee, initially supported by the former Decommissioning Operations Contractor (DOC), Stone & Webster (SWEC), and its subcontractor, Radiological Services, Inc. (RSI). The FSS plan was based on this information. These site characterization efforts used 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 (CCS) 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 initial site characterization (ICS) 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 7,900 measurements in unaffected areas and approximately 6,400 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 represent the key remaining structures upon license termination, an additional 51 concrete core samples were obtained to improve nuclide data. (See Section 2.5.3 and Attachments 2F and 2G for additional detail on these concrete cores and results.)

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 were limited to 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

All phases of the 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 initial radiological characterization survey (ICS) 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 any of 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 2B. All plant areas are included in one of the survey groups/packages. The five groups are listed below.

- 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 (CCS) 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 (ICS) included testing for PCBs, RCRA metals, semi- volatile organic compounds and volatile organic compounds.

- Group E-Hazardous Materials on Structures, Systems or Surfaces
- 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 (ICS and CCS) 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 (ICS) 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 (CCS) 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 (ICS) and continued characterization (CCS) samples, which included HTD nuclides. The lower values were for gamma spec analyses. When characterization soil samples (ICS and CCS) 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 a screening value of approximately 3-4 pCi/g for a 10 mrem/yr annual dose. 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/MY (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 (ICS) 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 initial characterization data (ICS). 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 (ICS) 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.

The NUREG-1575/NUREG-1507 method was used to calculate scan sensitivities in the continuing characterization work (CCS). 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 (ICS) 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 (ICS) 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 (CCS) with scan MDCs of approximately 5 pCi/g (Cs-137 source). This nominal MDC value of 5 pCi/g was based on a background of 10,000 c/m, an index of sensitivity (d') of 1.38, a surveyor efficiency factor of 0.707, and a conversion factor of 1200 c/m per microR/hr, as stated in the manufacturer's literature. The exposure rate of soil for 5 pCi/g was determined by Microshield and was the same value of 1.3 microR/hr, as given in Section 6.7.2.1 of NUREG-1575.

e. Instrument Calibrations

Analytical and field instruments for both ICS and CCS 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 (ICS). Approved procedures were employed to specify on-site instrumentation calibration requirements for continuing characterization (CCS). 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 (CCS) in order to detect discrete Co-60 particles.

2.3.5 Quality Assurance

Quality Assurance plans were developed for characterization work (ICS and CCS). 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 initial site characterization survey (ICS) . 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 (CCS) was performed 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 initial characterization (ICS) and continuing characterization (CCS) , 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 (ICS) 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 (ICS) 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 (ICS) 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 (CCS) under the requirements of DESEL Manual 100, "Laboratory Quality Assurance Plan."

The methods used by the off site laboratory for analysis of hazardous materials (ICS) 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 (ICS) 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 overall 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 (ICS) 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 (ICS) 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 (ICS) tracked and corrected nonconformances identified by these surveillances according to approved procedures.

During continued characterization (CCS), 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.

2.3.6 Data Quality Objectives

Initial site characterization (ICS) 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 (CCS) 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 initial characterization surveys (ICS) are reported by survey group and package number as identified below.[3] 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.

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[3] Additional survey packages were developed (and are discussed in this section) as necessary to support data collection for continued characterization. These later packages are not listed here in Section 2.3.7.

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
D3200	Hydrogen and Nitrogen System
D3300	Turbine Building Sumps and Drains

PACKAGE NUMBER	GROUP "D" Unaffected Plant Systems Survey Packages
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 (ICS) 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 (ICS) Results

The operational history and the range of contamination determined during initial site characterization (ICS) 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 2B.

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 2B 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 (CCS) 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 contained 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. Initial characterization surveys (ICS) showed elevated readings in one hotside shower drain. In the two years following shutdown, 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 (CCS) plan to properly evaluate the level and extent of contamination. These measurements 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/g 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 initial characterization (ICS) 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 operational 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. Total embedded piping includes approximately 800 feet of primary and secondary component cooling water pipes. Based on inventory estimates made in 2002, the total embedded piping expected to remain on site is approximately 940 linear feet, representing slightly over 150 m^2. A detailed listing of the embedded piping inventory is provided in Attachment 6-7.

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 describes the principal sections of buried and 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 68 ft. (C0300): During plant operation, the system was filled with reactor coolant water. Initial site characterization surveys (ICS) 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) during ICS. Higher levels of contamination have been found in subsequent surveys. This 16 inch embedded piping makes up a surface area of 26.5 m^2.

b. Containment Foundation Drains-approximately 378 feet.(C2000)[4]: 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 initial site characterization (ICS); 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. See Section 2.4.12. No removable surface contamination or direct surface measurements have been made. This combination of 2 inch and 6 inch embedded piping makes a surface area of 30.2 m^2.

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

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4 As noted in Section 2.3.7, additional survey packages were developed for data collection during continued characterization (i.e., not part of ICS) and, thus, are not listed in Section 2.3.7. Survey Packages C2000, D3500, and D3700 are examples of packages developed for CCS and/or for FSS using the same numbering system as was used for ICS.

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 initial site characterization (ICS) , 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 (D3500): 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 (D3700) (411ft): 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 20 linear feet of up to 1" piping
- Approximately 35 linear feet of 1.5" piping
- Approximately 50 linear feet of 2" piping
- Approximately 35 linear feet of 3" piping
- Approximately 55 linear feet of 4" piping
- Approximately 100 linear feet of 6" piping
- Approximately 45 linear feet of 8" piping
- Approximately 5 linear feet of 10" piping
- Approximately 25 linear feet of 16" piping
- Approximately 10 linear feet of 24" piping
- Approximately 20 linear feet of 30" piping
- Approximately 11 linear feet of 40" Fuel Transfer Tube piping

Each of these penetration, except for the Fuel Transfer Tube, consists of a five foot length of pipe penetration through the containment foundation wall. The calculated surface area of this embedded piping is approximately 78 m^2.

i. The Primary Auxiliary Building and Spray Building Penetrations (60ft). Several non-containment piping penetrations through the Primary Auxiliary Building and Spray Building will remain in the respective building foundations following demolition of the above grade structure. Each of these penetrations consists of a 2 to 3 foot length of pipe penetration through the building foundation wall. The calculated surface area of this embedded piping is approximately 19.5 m^2.

j. The spent fuel pool liner leak detection system (24ft). Four 1 inch lines embedded in the spent fuel pool structure will remain following demolition of the above grade structure. The calculated surface area of this embedded piping is approximately 1 m^2.

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 (ICS) (A9900, A9901, A9902) and continuing characterization (CCS) . In 1998, GTS Duratek took seven (7) concrete core samples that were later subjected to analysis by Stone and Webster to determine HTD nuclides at low MDC's. In 1999, forty-three (43) additional concrete core samples were obtained and analyzed by gamma spectrometry. In 2000, an additional eight (8) concrete cores were collected and analyzed for HTD nuclides at low MDC's. Table 2C-2 lists the original 43 cores (1-1A through 11-2A) taken during continuing characterization plus the 8 additional cores (12-1A through 13-3A) collected in 2000 for a total of 51 cores. Three of the cores (3-1A through 3-3A) were activated concrete and are labeled as "activation samples" in Table 2C-2. Four samples (5-6A, 6-5A, 6-6A, and 7-2A) had no reported activity. Section 2.5.3a discusses the establishment of the nuclide mixture for contaminated concrete surfaces. See Attachment 2F for a description of the process used to evaluate the concrete surface nuclide mixture. See Attachment 2G for additional discussion of concrete core sample collection and processing.

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 1 to 2 inches into the concrete, and dropping off at greater depths (A9902). 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 fluence. 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 2B 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. Initial characterization activities (ICS) 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 8.6.2.

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 2B.

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 (ICS) 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 initial characterization survey (ICS) 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 initial characterization survey (ICS) . 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 initial characterization survey (ICS) .

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
Wood & Scrub Land Exposure (NaI_2)	5.9 μR/hr	8.3 μR/hr	7.2 μR/hr
Open Land Exposure (NaI_2)	10.0 μR/hr	13.6 μR/hr	11.6 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	<400 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 (CCS)

The site's initial characterization work (ICS) left a few survey areas unresolved with respect to the nuclides present and the extent or boundaries of contamination. Those areas were characterized during the Continuing Characterization Survey (CCS) effort, which included obtaining the following data:

- 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
- Concrete cores
- Forebay/diffuser media
- Groundwater

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.

As noted in Section 2.1, characterization samples (CCS) 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 (ICS) and the methods employed for Continuing Characterization (CCS) 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 during CCS 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 (CCS) are summarized below. Detailed data including mean, maximum, and standard deviation are presented by survey package in Attachment 2D. 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. Stone & Webster Review of the GTS Report (ICS)

Upon review of the GTS Duratech report (ICS), Stone & Webster identified areas requiring additional characterization as follows:

1. Determine the extent of soil contamination at the Southwest fence (CR0200, CR1000[5]) - The East/West boundaries of

[5] Note: Survey package numbers, as initially established for characterization, are listed in Section 2.3.7. To distinguish a given package's data from the characterization phase to the Final Status Survey (FSS) phase, a convention was adopted. A preceding "C" was added (to the package number) to indicate the "characterization" and a preceding "F" would be used to denote the "FSS" phase of the project. Thus, "CR0200" in the LTP text refers to the survey package containing characterization data for survey package R0200.

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, pending final status survey.

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.

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 2C tables.

Data obtained during all phases of 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 (ICS). 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 1.0 pCi/g. The actual soil nuclide profile is provided in Section 2.5.3.

The samples from each area were analyzed by gamma spec. If the gamma spec results were consistent with reported values, between 240 and 800 g were removed from the sample containers and added to the composite. The amount removed depended on the total number of samples available from each location. The composites were well mixed and counted again to ensure expected results were achieved. The composites were then sent for HTD analysis except for H-3. Tritium was not analyzed because the samples had been in storage for a long time and were exceptionally dry.

Samples for H-3 analysis were taken from locations adjacent to the original sample locations. K-40 and Th were not reported because they were not plant-derived nuclides.

During characterization (CCS) 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.

For additional discussion on soil samples and nuclide fraction see Attachment 2I.

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 (both ICS and CCS), 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. Nearly all of the embedded pipe consists of the through-wall stubs of 1 to 4.5 feet in length. Since the embedded pipe contributes approximately 2 tenths of one percent of the total annual dose rate, it was decided to assume the small lengths of embedded pipe were contaminated with the same source term as the concrete surfaces through which they passed. Buried pipe is considered to be contaminated with the same source term as other contaminated surfaces, and the activity is released into the surrounding soil upon pipe degradation. Buried pipe contributes less annual dose than embedded pipe.

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).

f. 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 (ICS) 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. See Attachment 2G for discussion of concrete core sample collection and processing.

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 (i.e, variance of activity with depth; see Table 2-10). 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 6.6.2.

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. See Table 2C-3 for results of reflecting pond samples. (Fire pond water and sediment results are not included since the fire pond will be demolished.)

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.) See Section 2.5.3.d for additional information regarding site hydrogeology and groundwater sampling, and the establishment of the groundwater nuclide fraction used for dose assessment.

As part of both initial and continuing site characterization, forebay sediment was sampled. To gain additional insight regarding the spatial distribution of contamination and to support further characterization and remediation planning, additional sampling efforts were undertaken. The principal campaign was in Spring 2001 and included the sampling of: (1) sediment around the protective rip-rap (inside the forebay), (2) underwater sediment on the structure floors, (3) exposed material on the forebay ledges near the weir wall, and (4) dike soil material beneath the rip-rap. Diver operations and inspections of the diffuser also provided an opportunity for the sampling of sediment inside the diffuser piping, as well as piping coupons. The characterization of the forebay and diffuser system is summarized in 2.5.3e and described in more detail in Attachment 2H.

Section 6.6.9 discusses the associated dose assessment related to these contaminated media.

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 (both ICS and CCS) 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 (Including "Special Areas")

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 buried and embedded pipe, may also remain. The nuclide profile determined for contaminated concrete is assumed to apply to all concrete surfaces. 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 original set of 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 on 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. Additional concrete cores were taken and analyzed, revealing other areas in the plant warranting a separate nuclide fraction. See discussion below related to "special areas."

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 2-7 provides the nuclide fraction for the "balance of plant" contaminated concrete surfaces.

Table 2-8 gives the nuclide fraction for special areas in the plant. These areas include the containment outer annulus trench, the PAB pipe tunnel, and the letdown heat exchanger cubicle. These were separated from the "balance of plant" contaminated concrete surfaces and were chosen based on operating conditions and the presence of TRU contamination. The dose consequences and DCGL for this collection of areas are described in Section 6.7.2.

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
Nuclide Fractions
Contaminated Concrete Surfaces
("Balance of Plant" Areas)

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 Surfaces
"Special Areas"

Nuclide	Nuclide Fraction (1/04)
Mn-54	4.03E-04
Fe-55	2.24E-02
Co-60	3.64E-01
Ni-63	3.02E-01
Sr-90	6.87E-03
Sb-125	4.52E-03
Cs-134	2.82E-03
Cs-137	2.89E-01
Pu-238	1.17E-04
Pu-239	8.75E-05
Pu-240	8.75E-05
Pu-241	6.71E-03
Am-241	5.93E-04
Cm-243	4.65E-05
Cm-244	4.45E-05

b. Activated Concrete / Rebar

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. As noted in Section 2.5.2f, one portion of activated concrete included embedded rebar. The rebar sample was also 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 fractions for the activated concrete and rebar 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.

Based on the higher dose contributions from activated concrete, in comparison to the rebar, the nuclide fraction for activated concrete was used in the Section 6 dose assessment. See Section 6.6.2.

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	677*	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 - 4.0	824	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		

*Adjusted to remove Cs-137 surface contamination from the total activity

**Note that the depth column represents a "label" for each sequential slice and is not intended as an exact measurement. The slices were generally ½" to $^{3}/_{4}$" but were not uniform in thickness. Therefore, while Table 2-10 presents the profile out to 20 inches, this represents all of the data available for the entire 22 inch core.

***Measured activity provided in this table includes gamma detectable activity from the nuclides listed in Table 2-9.

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.[6] 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.

For additional discussion on soil samples and nuclide fraction, see Attachment 2I.

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

6 Regarding buried and embedded piping and its impact on soil contamination, the most significant of buried/embedded piping within the industrial area are the HPCI and LPCI lines. These contained the same fluid as the RWST and would be well represented by the RWST and the subsequest RWST related soil samples used in part of the soil nuclide fraction.

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 contained relatively low levels of 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. Additional information regarding background tritium in and around the Maine Yankee site is provided in a comprehensive report on site hydrogeology (Stratex; February 2002, Reference 2.7.19).

The February 2002 Stratex report (referenced above) summarized and discussed radioactivity in site groundwater and its relationship to site history regarding releases of contamination.[7] In general, while relatively low levels of Co-60 and Cs-137 have been sporadically detected in the containment foundation sump and other site wells, the primary, consistently detected nuclide is tritium. The nuclide fraction for groundwater (used as an initial condition for the dose assessment) consists of tritium only. See Section 6.6.6 for additional discussion, activity levels, and the use of this nuclide fraction in the dose assessment.

An additional groundwater re-sampling program consisting of fifteen wells was implemented in spring of 2002. The results of this effort, which included the analysis of twelve of the fifteen well samples for "hard to detect" nuclides, were reported in Maine Yankee's letter to the NRC, dated August 28, 2002 (Reference 2.7.20). This submittal included an addendum to the February 2002 Stratex report (August 2002). This sampling effort included not only the containment foundation sump but also numerous wells in the industrial area as well as several new wells, as recommended in the February 2002 Stratex report. (Additional groundwater exploration of the Primary Auxillary Building "PAB" test pit area, as recommended by Stratex in February 2002, was not pursued. See discussion below.)

Consistent with prior well sampling in the industrial area, the results of this site groundwater re-sampling effort showed relatively low levels of groundwater contamination. Two wells reported relatively low levels of either Co-60 and Cs-137. Tritium levels were above background in several wells; however, they were consistent with previously detected concentrations and well within the conservative levels assumed for dose modeling. Hard to detect analyses (including transuranics) detected no

[7] Stratex, February 2002, Section 3.7 (LTP Reference 2.7.19).

other nuclides, also consistent with prior sampling. (See References 2.7.20 and 2.7.25.) The nuclide fraction for both ground and surface water is given in Table 2-12.

Special consideration and assessment was given to the isolated detection (1999) of contamination in the PAB test pit, as discussed in the February 2002 Stratex report. Additional study of the fate and transport of relevant nuclides was performed by Stratex, supported by Brookhaven National Laboratory (reported in the August 2002 Maine Yankee submittal to the NRC [8]). Based on the additional study, including consideration of recent sampling of the test pit and the containment foundation sump and site hydrogeology, Maine Yankee concluded that no additional field investigations or groundwater exploration were necessary to further study the fate and transport of the historical PAB test pit contamination. In that the PAB test pit is a structure to remain post-decommissioning, it will undergo any necessary remediation and final status surveys to demonstrate compliance with surface contamination release criteria. (See Reference 2.7.20.)

Samples from the containment foundation sump and the PAB test pit will be routinely obtained and analyzed until the final status survey is commenced for these two plant areas. See Section 6.6.6. Furthermore, as noted in Section 6.6.6, future groundwater sampling data obtained prior to unrestricted release will be considered for its impact on the dose assessment. Should such consideration require additional groundwater information, Maine Yankee will take appropriate action, which may include sampling of existing wells, if available, or the installation and sampling of new wells at appropriate locations.

Table 2-12
Ground and Surface Water Nuclide Fraction

Nuclide	Fraction
H-3	1.000

e. Forebay and Diffuser Contaminated Media

A detailed discussion of the characterization of the forebay and diffuser system is provided in Attachment 2H. The characterization effort and resulting nuclide fraction for forebay/diffuser media are summarized below.

8 Reference 2.7.20, as corrected by Maine Yankee letter to the NRC, MN-02-045, dated October 3, 2002 (Reference 2.7.24)

The forebay (and seal pit) characterization consisted of sampling efforts that identified the following contaminated media:

1. Rock floors and walls of the forebay/seal pit, as well as a limited amount of concrete surfaces at the northern and southern ends of the forebay basin;
2. Rip-rap, contaminated surfaces;
3. Marine sediment deposited on the floors of the forebay/seal pit and around the rip-rap; and
4. Dike "soil," i.e., that material beneath the rip-rap, interior to the dike walls.

Sampling and assessment of the diffuser system identified two contaminated media, namely, sediment entrained inside the diffuser discharge piping and contaminated surface film deposited on the inside surfaces of diffuser piping. This surface contamination was noted to be very similar to that on the rip-rap covering the interior forebay dike walls.

As the results of several sampling campaigns (including diving operations), each of the above media were sampled, analyzed, and evaluated regarding nuclides present, activity levels, and relative fractions. The evaluation included three sets of sediment samples analyzed for HTD nuclides. The overall assessment concluded that a single nuclide fraction was appropriate and conservative for application to these media. The nuclide fraction for forebay and diffuser related media is presented in Table 2-13. See Attachment 2H for additional discussion on the principal construction features of the forebay and diffuser system, the sampling campaigns, results, and conclusions. See also EC 041-01 for supporting technical bases and analyses.

Table 2-13
Forebay/Diffuser Material Nuclide Fractions

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

f. Future Sampling

The radionuclide profiles for contaminated concrete, activated concrete, soil, ground water, surface water, and sediment listed in Tables 2-7 and 2-8, 2-9, 2-11, 2-12, and 2-13 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 (ICS) 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 2A & 2B). 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 (ICS) 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 both ICS and CCS. Background values for structural materials using these detectors are shown in Table 2-14.

Table 2-14
Structural Material Backgrounds

Background Counts per Minute (reflects beta count rate)		
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 (both ICS and CCS) 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 (ICS and CCS) 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 (ICS and CCS) support dose assessment and FSS design.

2.7 References

2.7.1 NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual, (MARSSIM), Revision 1 (June 2001)

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

2.7.3 Continuing Characterization (CCS) Plan (PMP 6.8).

2.7.4 CCS Quality Control (PMP 6.8.4).

2.7.5 Corrective Action Program

2.7.6 Document Control Program (0-17-1).

2.7.7 Radiation Protection Performance Assessment Program (PMP 6.0.8).

2.7.8 Selection, Training and Qualification of Radiation Protection Personnel, (PMP 6.9).

2.7.9 Maine Yankee Atomic Power Co. (MY), *RCRA Quality Assurance Project Plan for Maine Yankee Decommissioning Project, Revision 1.* (June 28, 2001)

2.7.10 NUREG-1507, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions.(June 1998)

2.7.11 NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans. (April 2000)

2.7.12 Regulatory Guide 1.179, Standard Format and Content of License Termination Plans for Nuclear Reactors. (January 1999)

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 1-9, 1998 (ICS).

2.7.15 Dr. Chabot letter to P. Dostie, dated November 12, 1998, discussing determination of MDC

2.7.16 Maine Yankee letter to the NRC, MN-02-002, dated January 16, 2002, transmitting special report from the Technical Issue Resolution Process, entitled "Transuranic and Other Hard to Detect Radionuclides in Maine Yankee Sample Media."

2.7.17 NRC letter to Maine Yankee, dated July 30, 2002, Issuance of Amendment No. 167, license amendment approving partial release of site lands.

2.7.18 Maine Yankee Engineering Calculation, EC-041-01 (MY), Revision 0

2.7.19 Maine Yankee letter to the NRC, MN-02-010, dated February 20, 2002, "Maine Yankee Response to NRC RAI #16 (dated December 18, 2001) Addressing Site Hydrogeology," (included submittal of Stratex, LLC, report, *Site Hydrogeology Description, Maine Yankee, Wiscasset, Maine,* February 2002).

2.7.20 Maine Yankee letter to the NRC, MN-02-037, dated August 28, 2002, "Maine Yankee Addendum Report Regarding Site Hydrogeology," (including Stratex, LLC, report *Site Hydrogeology Addendum, Maine Yankee, Wiscasset, Maine*, August 2002).

2.7.21 Maine Yankee letter to the NRC, MN-02-011, dated March13, 2002, "Response to NRC Request(s) for Additional Information for the Maine Yankee Atomic Power Station LTP"

2.7.22 MYAPC Historical Site Assessment (HSA), transmitted by MN-01-038 dated October 1, 2001

2.7.23 Maine Yankee letter to the NRC, MN-02-015, dated April 11, 2002, "Revised Maine Yankee Response to NRC RAI #5 (dated December 18, 2001) - Supplementary Historical Site Assessment (HSA) Data"

2.7.24 Maine Yankee letter to the NRC, MN-02-045, dated October 3, 2002, "Minor Changes to Maine Yankee Responses to NRC Request for Additional Information"

2.7.25 Maine Yankee Engineering Calculation, EC-006-01 (MY), Revision 2

2.7.26 Maine Yankee letter to the NRC, MN-02-063, dated December 12, 2002, "Update on Forebay Dike Coring Results and Associated Changes to LTP Attachment 2H (LTP Revision 3 Addenda)"

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 641 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, as supplemented, documented approximately 140 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 Backlands Report

On August 16, 2002, Maine Yankee submitted an application[1] for amendment to its license to release these backlands from the jurisdiction of the license. This application was supplemented[2] on November 19 2001. In the supporting justification attached to the application, Maine Yankee reviewed the soil sample Cs-137 results of the Initial Characterization Survey (ICS) to determine if the residual radioactivity, if any, in the backlands is indistinguishable from background and thereby support the classification of "non-impacted".

Demonstrating indistinguishability from background employs MARSSIM Scenario B. In Scenario B, the null hypothesis is that the survey unit meets the release criterion (indistinguishable from background). Under Scenario B, the comparison of measurements in the reference area and survey unit is made using two nonparametric statistical tests: the Wilcoxon Rank Sum (WRS) test and the Quantile test. The WRS and Quantile tests are both used because each test detects different residual contamination patterns in the survey units. Because two tests are used, the Type I error rate, α, (normally set at 0.05) is halved, and set at 0.025, for the individual tests. Using the NUREG-1505 recommended α of 0.025 allows for the use of the look-up tables in NUREG-1505, for r and k values used in the Quantile test.

The WRS test is designed to determine whether or not a degree of residual radioactivity remains uniform throughout the survey unit. The Quantile test is designed to detect a patchy

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1 Maine Yankee Letter to USNRC dated August 16, 2001, "Early Release of Backlands (Combined), Proposed Change No. 211, Supplement No. 1, (MN-01-034)

2 Maine Yankee Letter to USNRC dated November 19, 2001, Early Release of Backlands (Combined), Proposed Change No. 211, Supplement No. 2, (MN-01-044)

contamination pattern.
Table 2A-8 contains the soil sample Cs-137 results for the background reference area. The background reference area consisted of area surrounding the Marrymeeting Airfield located approximately 10 miles from the site and was representative of site characteristics. The Kruskal-Wallis test was used to confirm that there was no significant difference in the mean background concentrations among potential reference areas.

Table 2A-9 summarizes the results of the soil sample Cs-137 results for the backlands areas and compares them to the results for the background reference area. For each of the backlands areas, the results of the WRS test, where applicable[3], and the Quantile test successfully demonstrated that the residual radioactivity, if any, in the areas was indistinguishable from background.

2A.7 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, the results of the site characterization surveys, and the demonstration of indistinguishability from background described in the Backlands Report, 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 may be performed in this reference area, but no systematic sampling as required by MARSSIM for impacted areas will be performed.

[3] For area R-1500 Ash Rd. Rubble Piles, the maximum Cs-137 reading was less than the value known as the Upper Boundary of the Grey Region; therefore, the application of the WRS test was not necessary to demonstrate indistinguishability from background.

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	0.45; 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					0.42; 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	0.45; 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
Shoreline		0.20; n=30
Mean Value	0.56	0.32

Table 2A-8 Reference Area Soil Sample Cs-137 Results pCi/g			
Reference Areas - Merrymeeting Airfield	**Mean (Ave.)**	**Std. Dev. (1 σ)**	**Number of Samples**
Combined (wood, open & scrub)	0.42	0.21	50
Wood Land	0.47	0.24	10
Open Land (Hay Field)	0.38	0.12	30
Scrub Land	0.48	0.34	10

Table 2A-9 Soil Sample Cs-137 Results					
Area Description	Minimum Cs-137 pCi/g	Median Cs-137 pCi/g	Average Cs-137 pCi/g	Maximum Cs-137 pCi/g	Number of Measurements
Reference Area R2200	0.09	0.38	0.42	1.40	50
Survey Unit R1500* Ash Rd. Rubble Piles	0.02	0.06	0.07	0.21	30
Survey Unit R1600 Eaton Farm	0.05	0.39	0.45	1.43	60
Survey Unit R1700 North of Old Ferry R.	0.04	0.30	0.39	1.55	60

* Disturbed open land area within R1700 North of Ferry Rd.

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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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 cm²			Removable Beta DPM/100 cm²			Removable Alpha DPM/100 cm²			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	0.07	0.21	N/A	N/A	11.34	12.63	0.63
R1600 Eaton Farm Land	60	0	0	0	59	0.045	1.43	N/A	N/A	12.07	17.8	2.06
R1700 Land North of Ferry Rd	60	0	0	0	50	0.39	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

* Includes twelve marine sediment samples taken the New Meadows River and the Damariscotta River.

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

* Activity consisted, in part, of discrete particles

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	MDA pCi/L	Plant Derived Gamma Activity ?
R2200	Eaton Farm Well	685	743	No
	Bailey Farm Well	-1689	3126	No
	Days Ferry (private well)	1220	2255	No
	Average	635	2042	

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)	1331 0	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 (geometry corrected except as noted (1))

Sample #	Net CPM 43-68 (2)	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	157	28	36	280	33	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'

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

Concrete Core Samples (geometry corrected except as noted (1))

Sample #	Net CPM 43-68 (2)	Co-60 pCi/g	Cs-134 pCi/g	Cs-137 pCi/g	Eu-152 pCi/g	Eu-154 pCi/g	Area
6-4A	0	0.4		2			PAB 11'
6-5A	0	0		0			PAB 11'
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		14.7			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	1179			Cntmt El-2'

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

Concrete Core Samples (geometry corrected except as noted (1))

Sample #	Net CPM 43-68 (2)	Co-60 pCi/g	Cs-134 pCi/g	Cs-137 pCi/g	Eu-152 pCi/g	Eu-154 pCi/g	Area
12-5A	143651	664	56	11914			Cntmt El-2'
13-1A	1193	7		61			PAB El-11'
13-2A	14383	86	10	192			PAB El-11'
13-3A	5273	52	2	47			PAB El-11'

(1) Activation Samples (not geometry corrected)
(2) Net Count Rate. For additional discussion, see Attachment 2G.

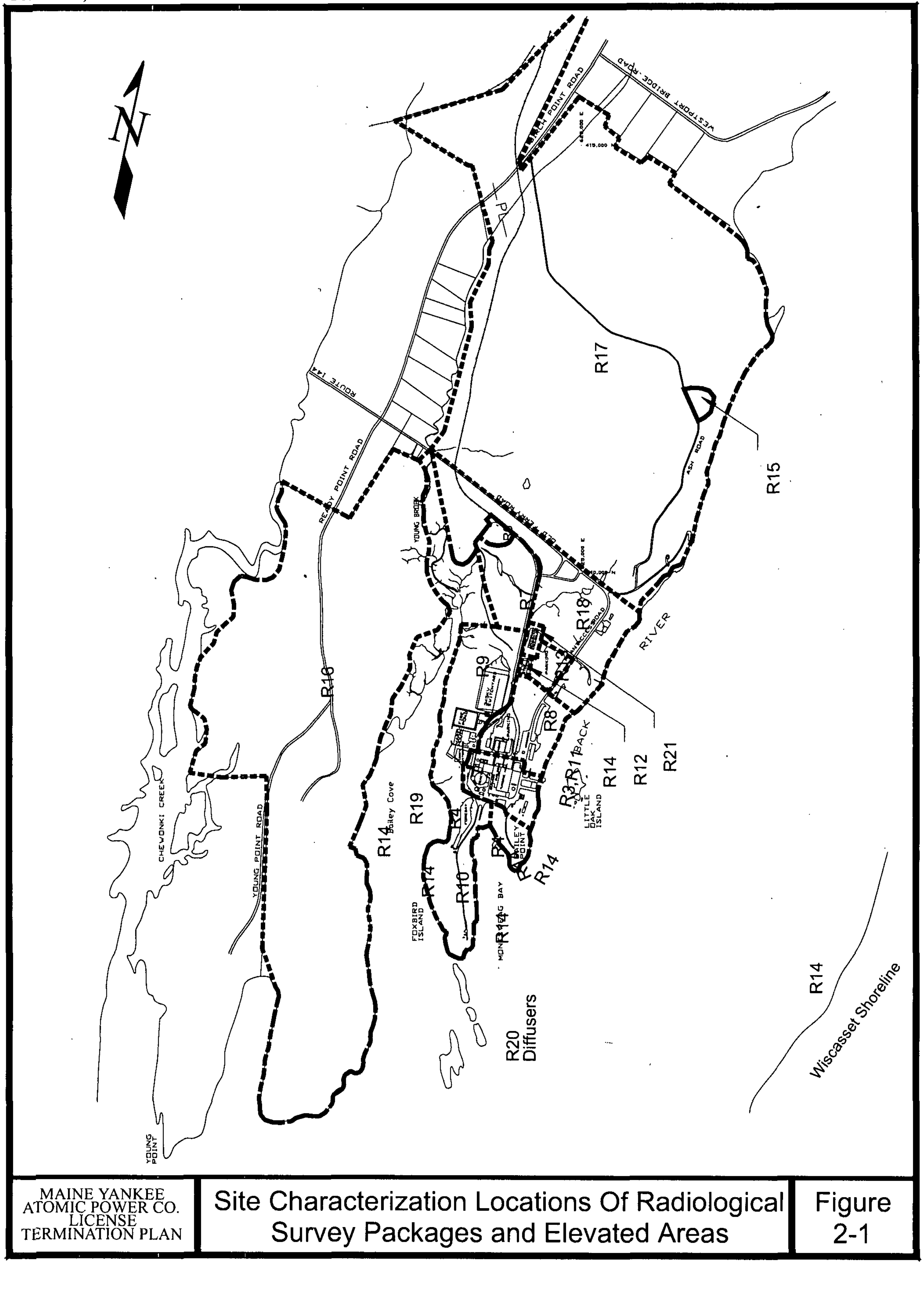
Table 2C-3
Continued Characterization Results for Water and Sediment Samples

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 (1)	Fe-55: 13.6 pCi/g Ni-63: 8.9 pCi/g Co-60: 31.7 pCi/g Sb-125: 0.4 pCi/g Cs-137: 1.2 pCi/g

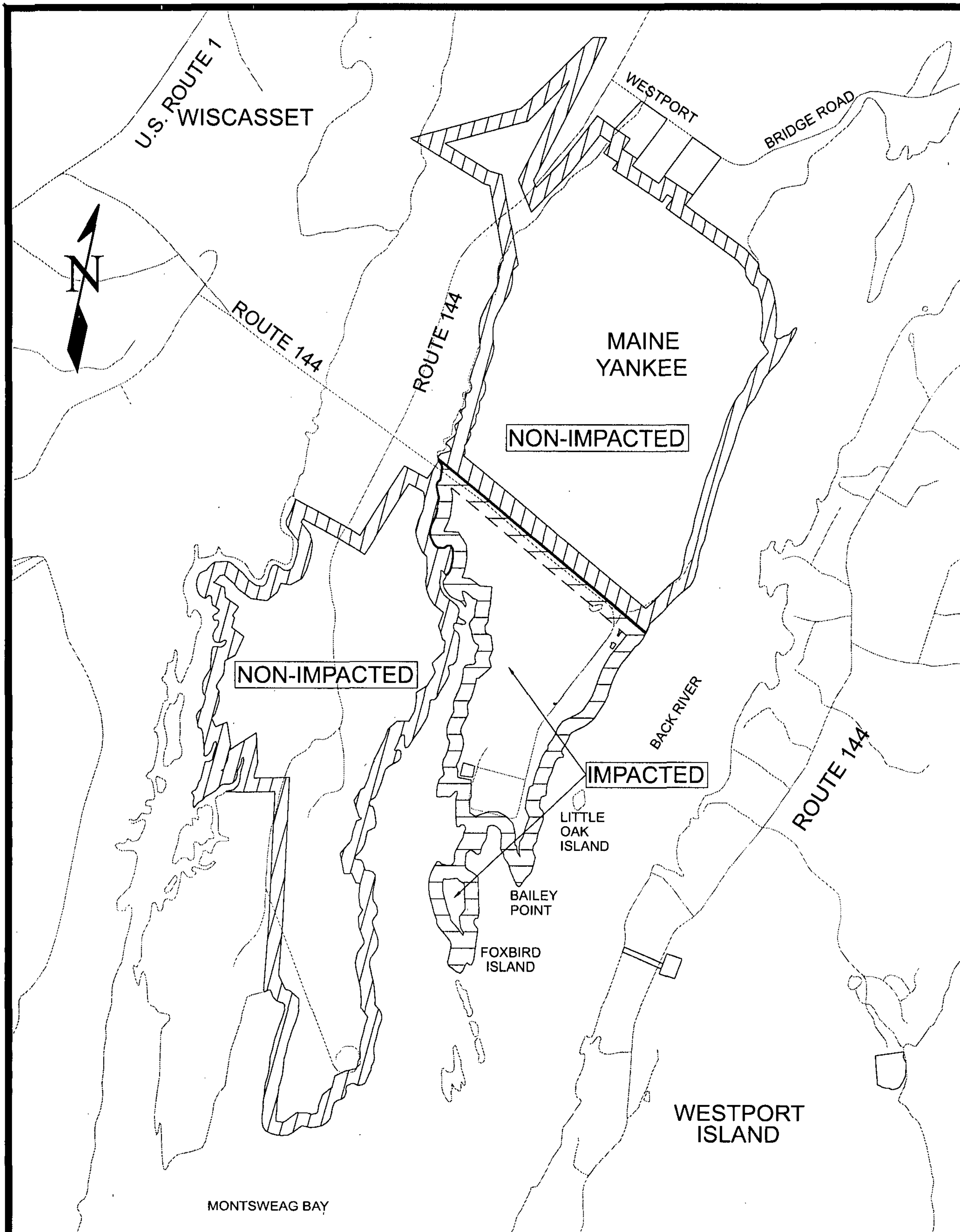
(1) Results are from the 2000 composite forebay sediment sample. For additional information, see Attachment 2H regarding forebay and diffuser characterization.

ATTACHMENT 2D

Maine Yankee Site Characterization
Locations of Radiological Survey Packages



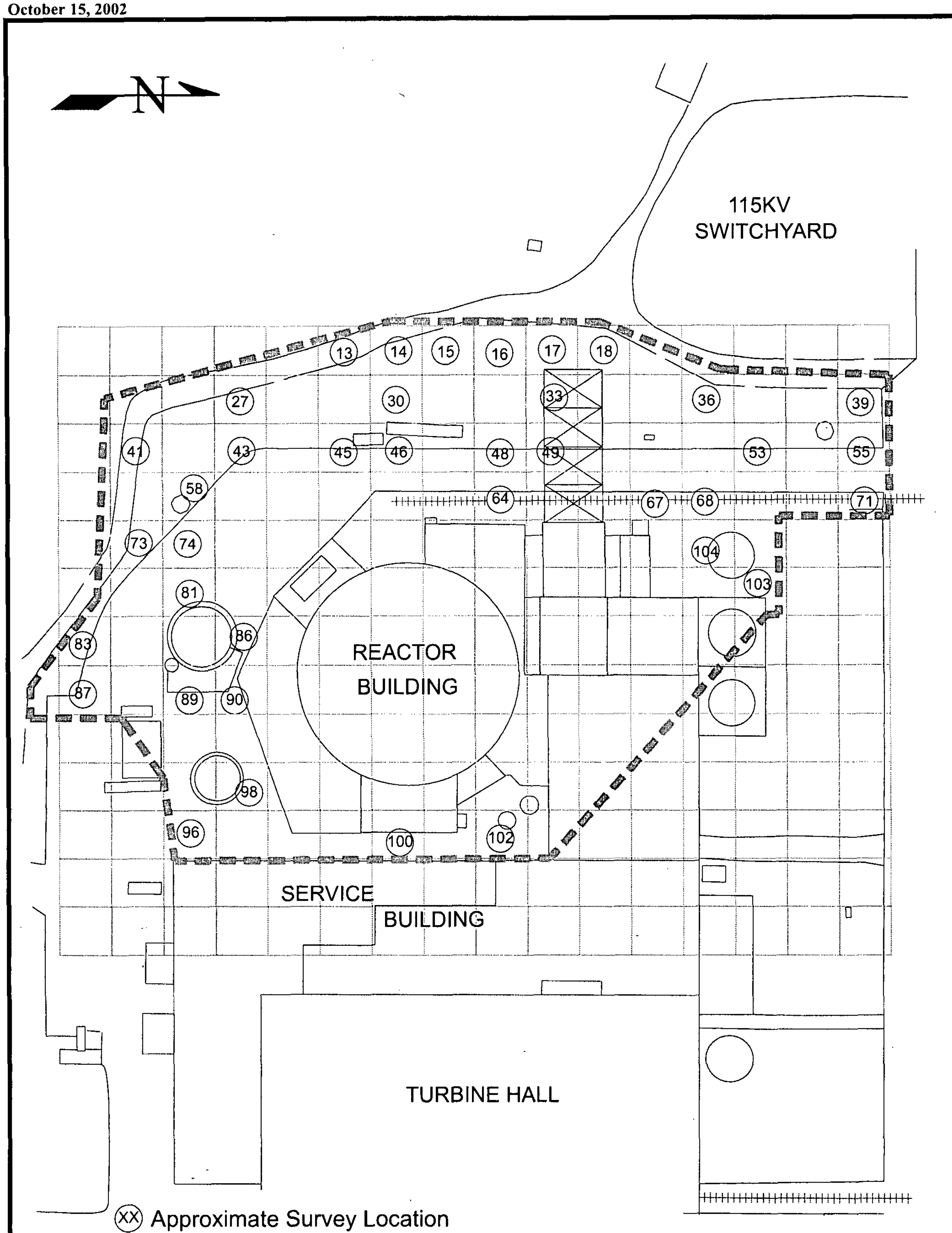
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Locations Of Radiological Survey Packages and Elevated Areas	Figure 2-1



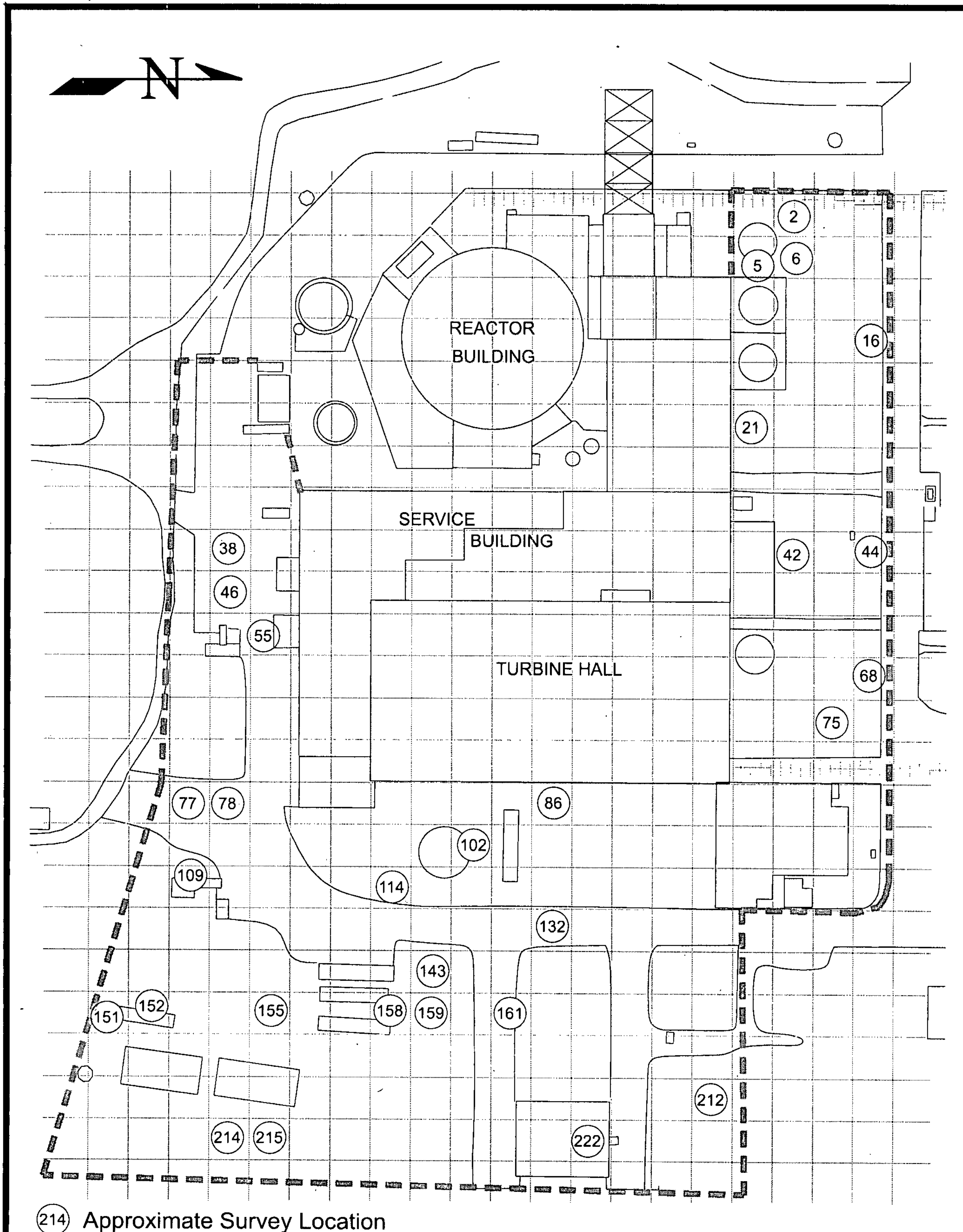
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Impacted / Non- Impacted Areas	Figure 2-2

ATTACHMENT 2E

Site and Survey Area Maps



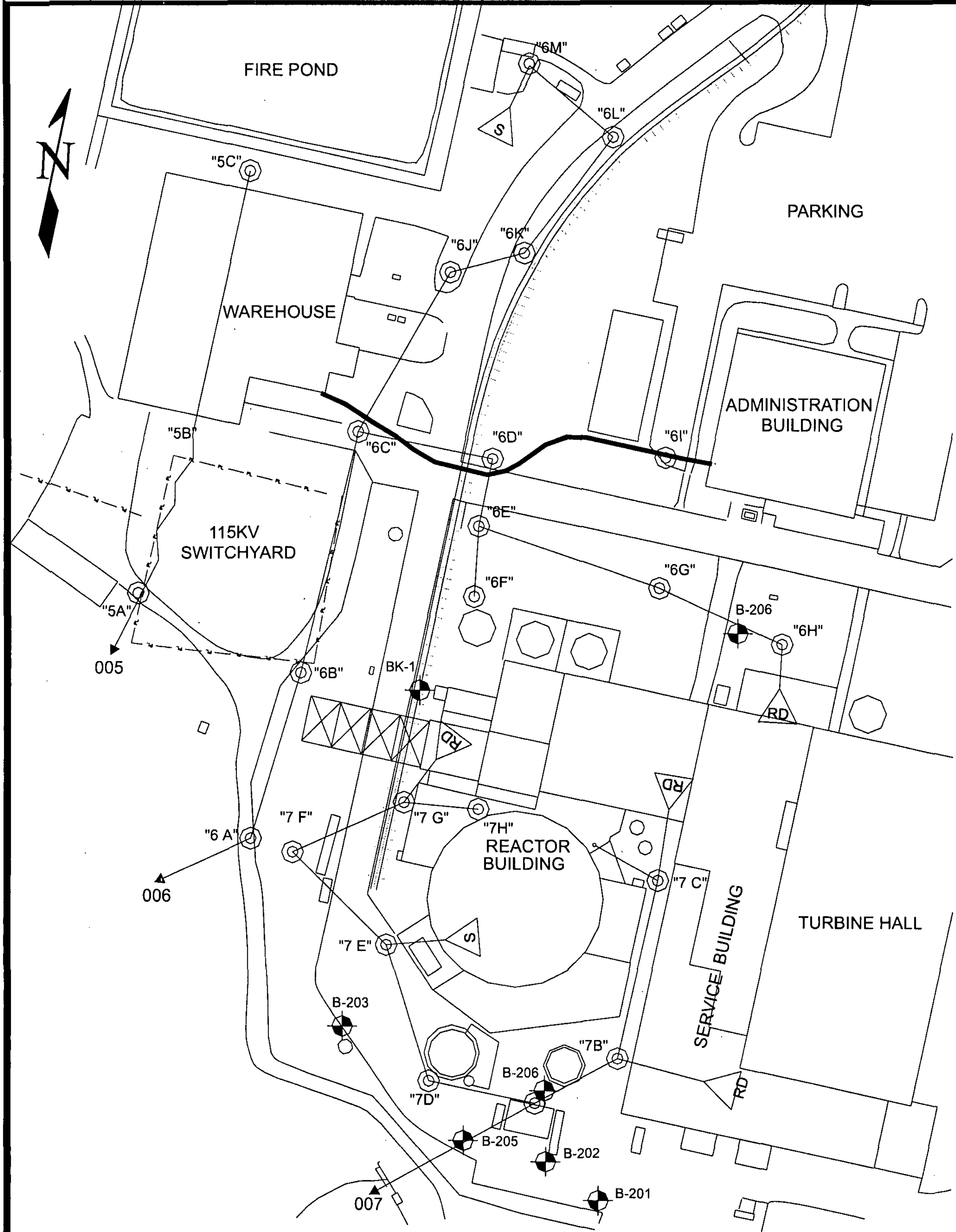
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization West Side Yard Survey Package R0100	Figure 2-3



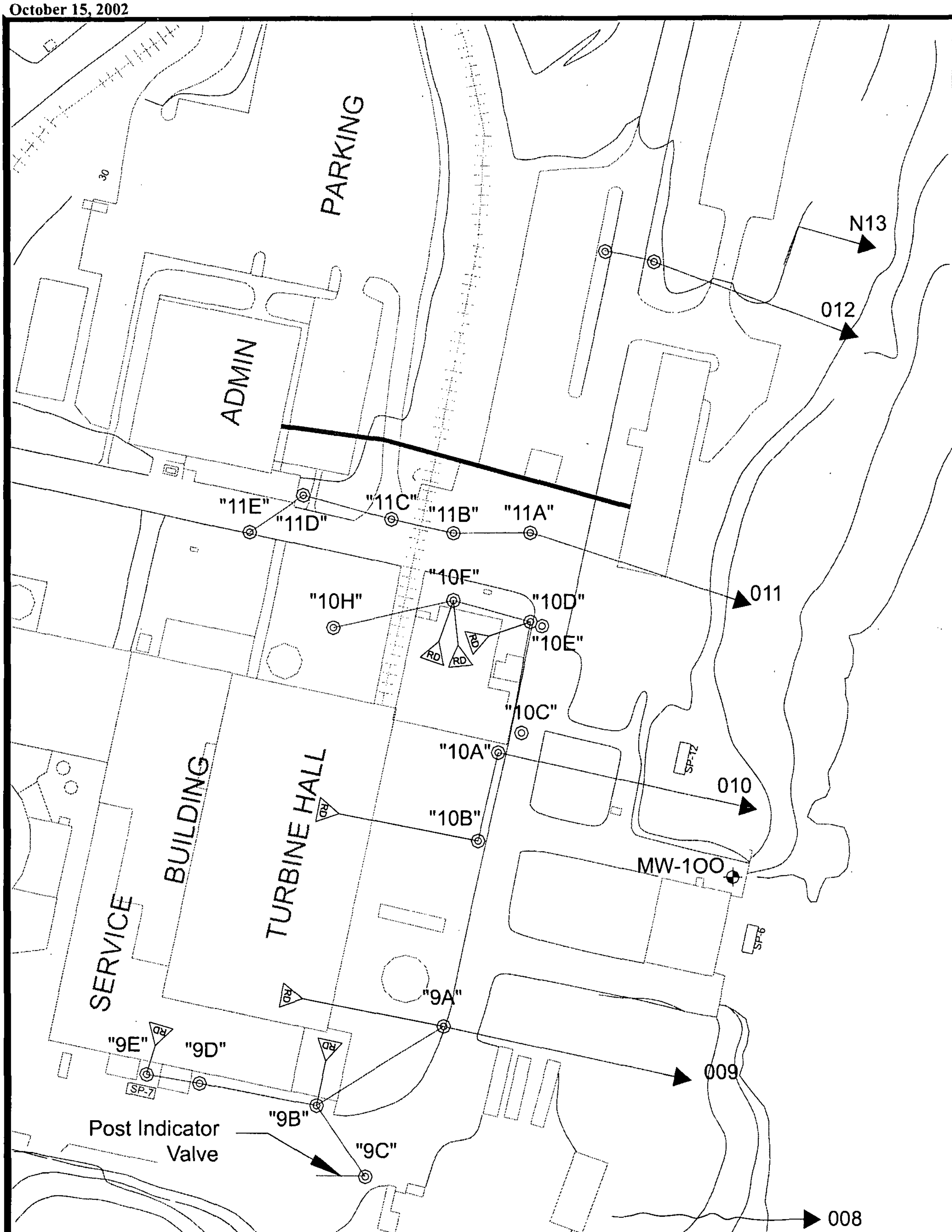
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization East Side Yard Survey Package R0200

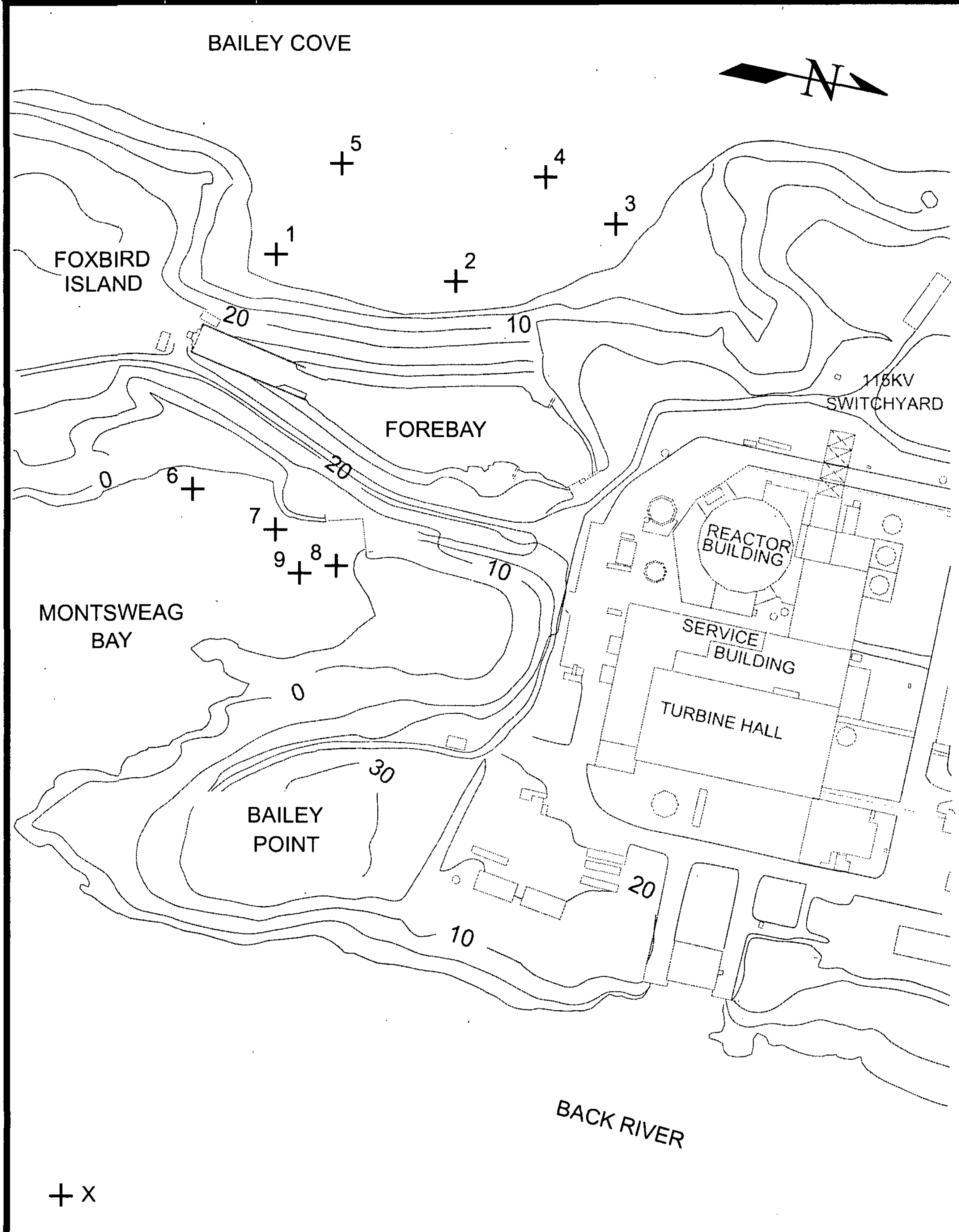
Figure 2-4



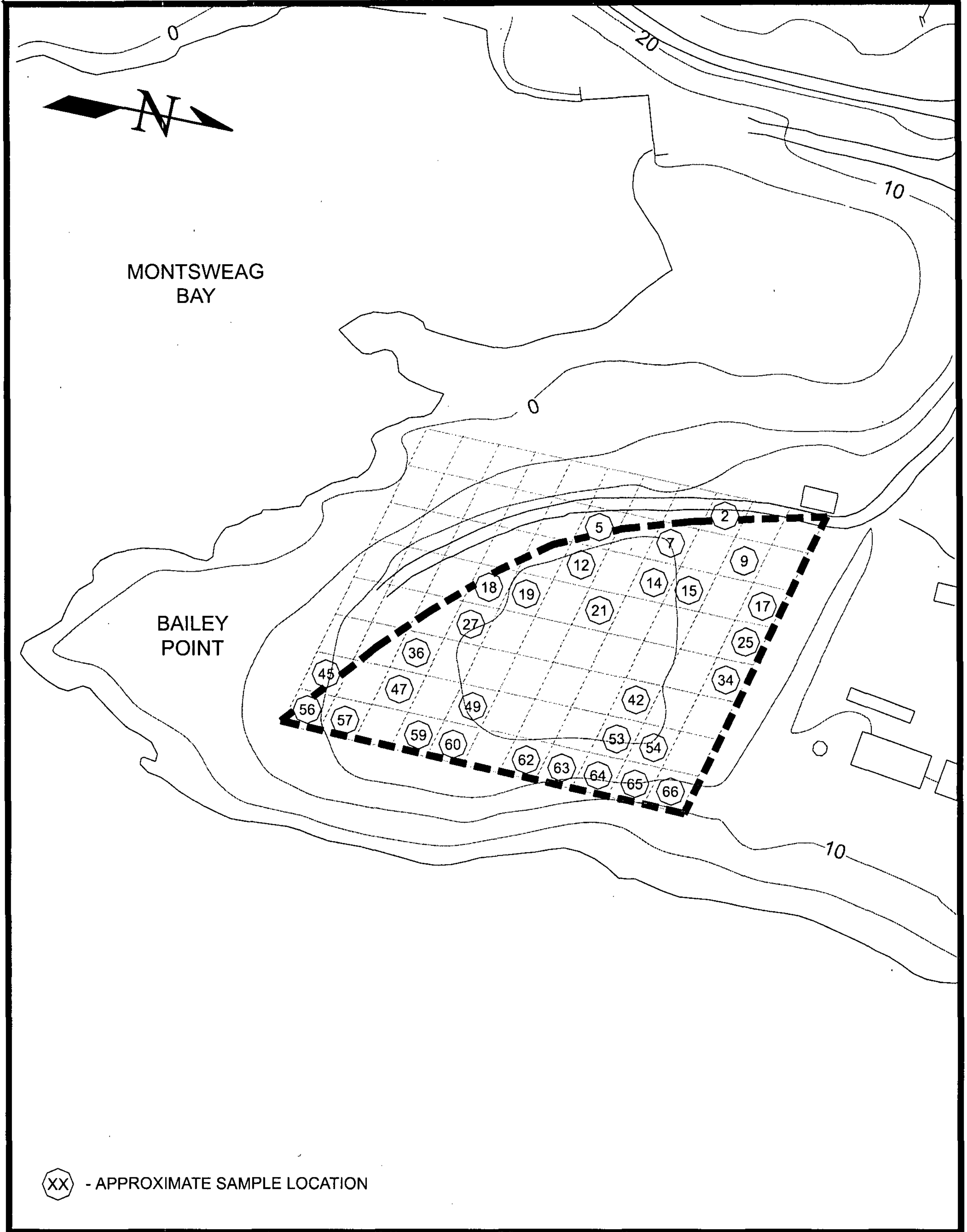
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Roof and Yard Drains #005, 006, 007; Survey Package R0300-1	Figure 2-5



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Roof and Yard Drains #008, 009, 010, 011, 012, N12; Survey Package R0300-1	Figure 2-6



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Forebay Shorelines Survey Package R0400-1A	Figure 2-7



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Bailey Point Survey Package R0500-2	Figure 2-8

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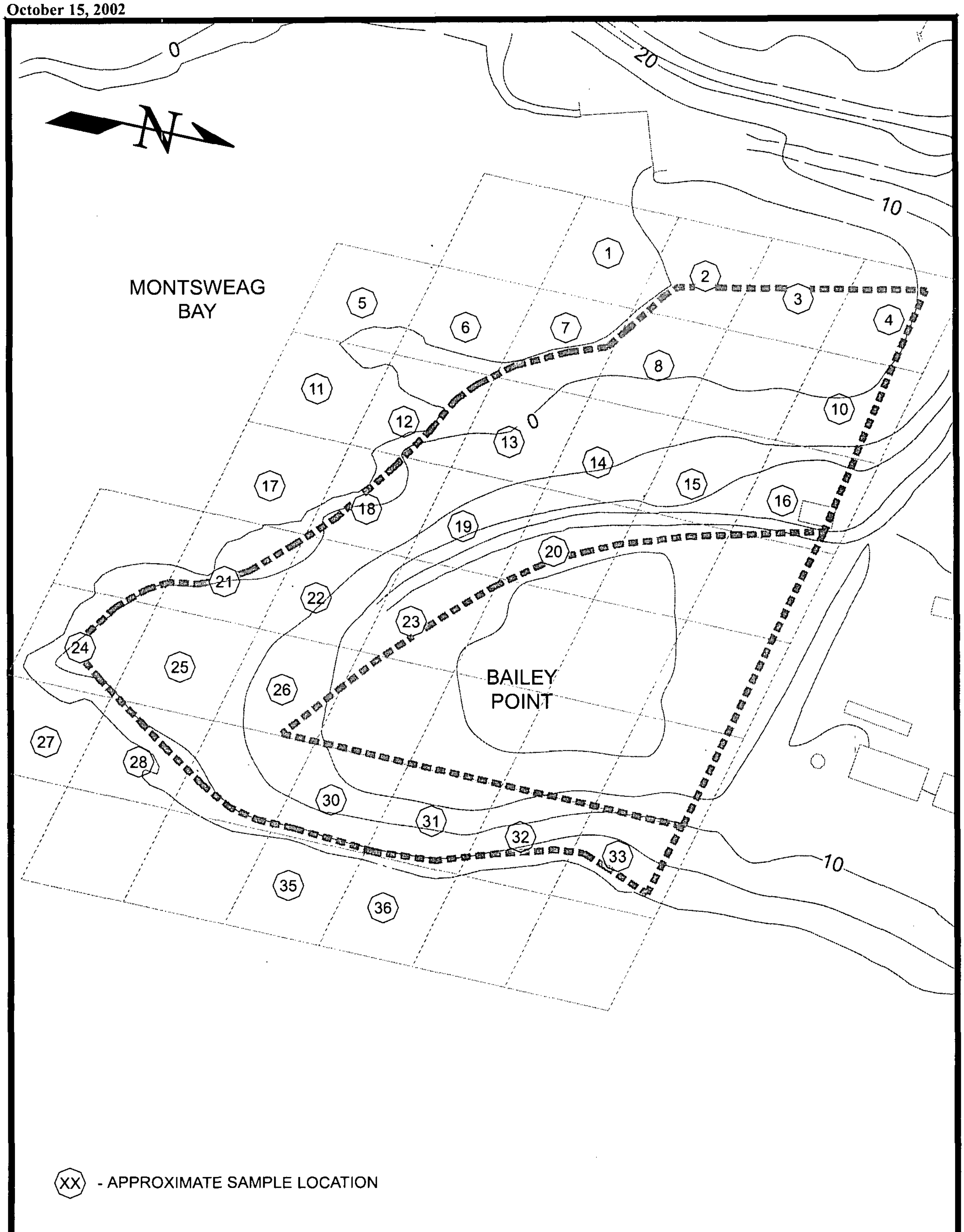
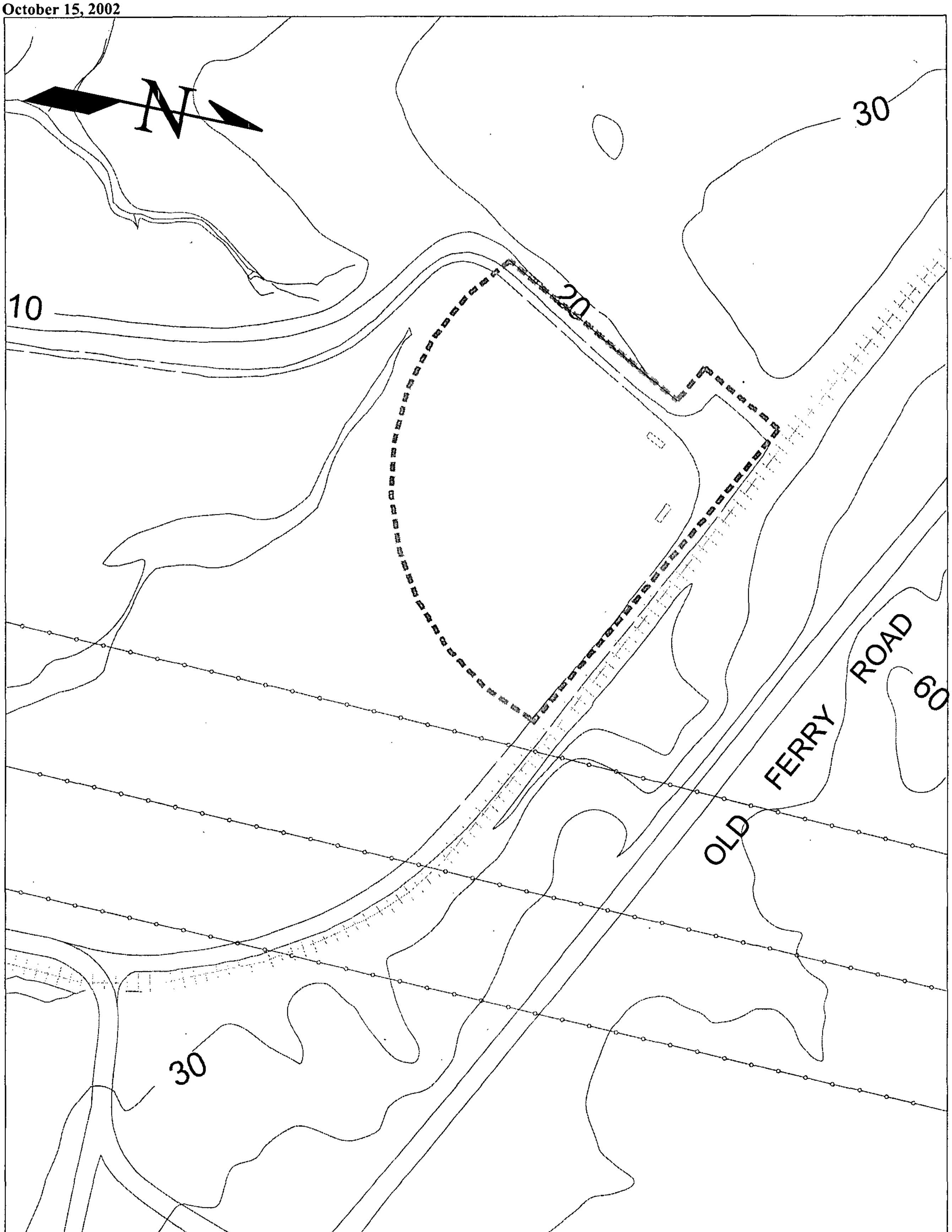
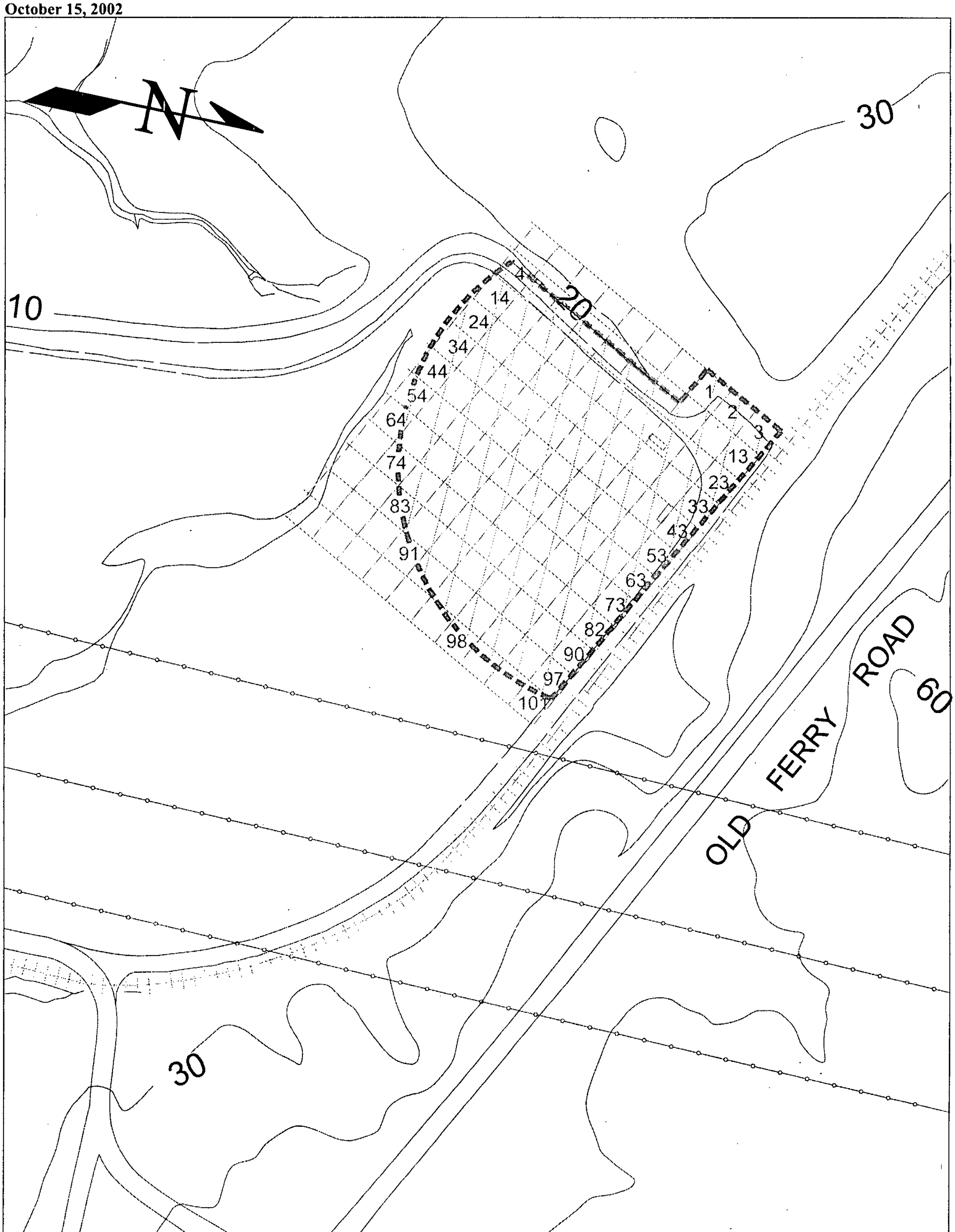
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Bailey Point Survey Package R0500-2	Figure 2-9



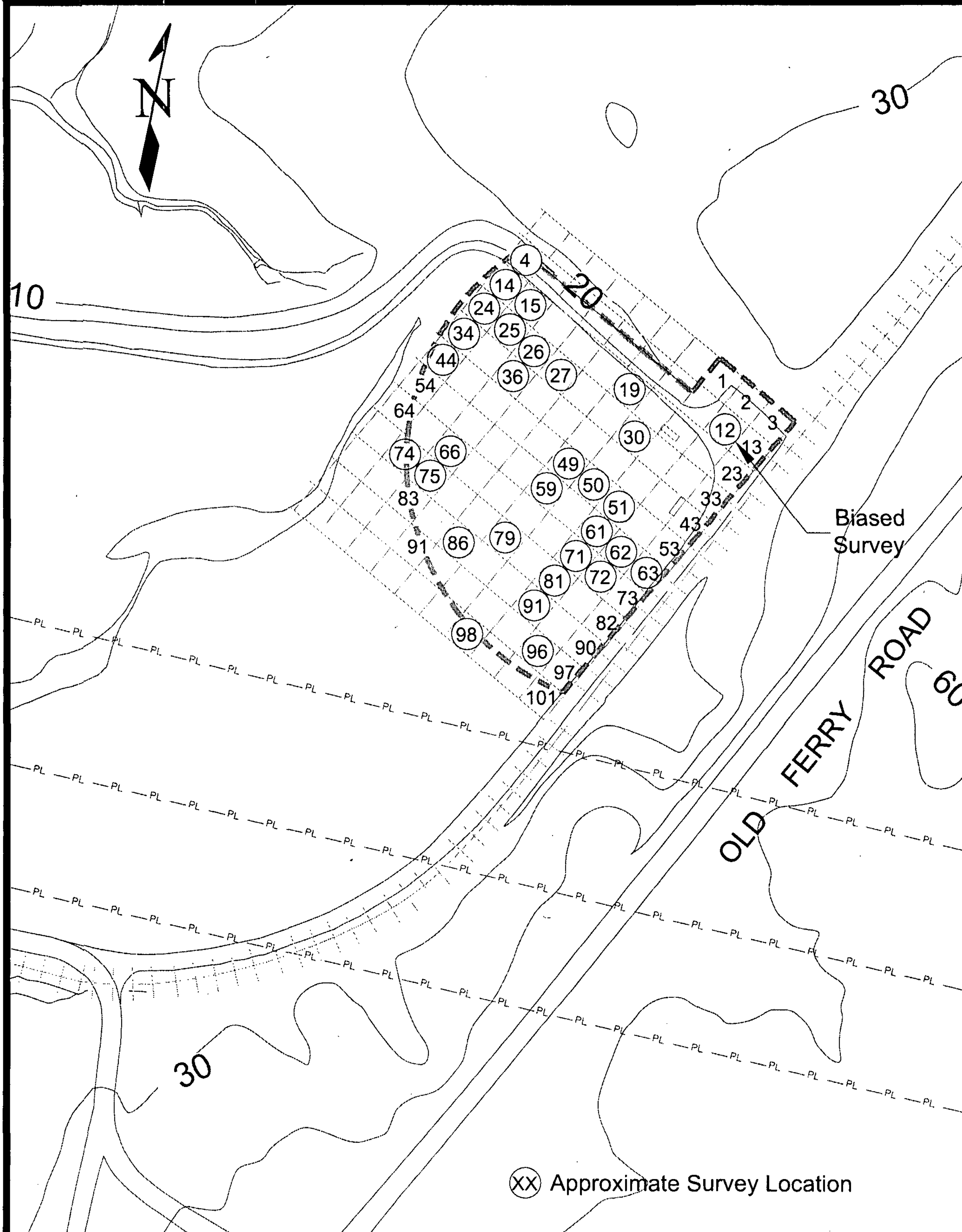
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Bailey Point Survey Package R0500-4	Figure 2-10



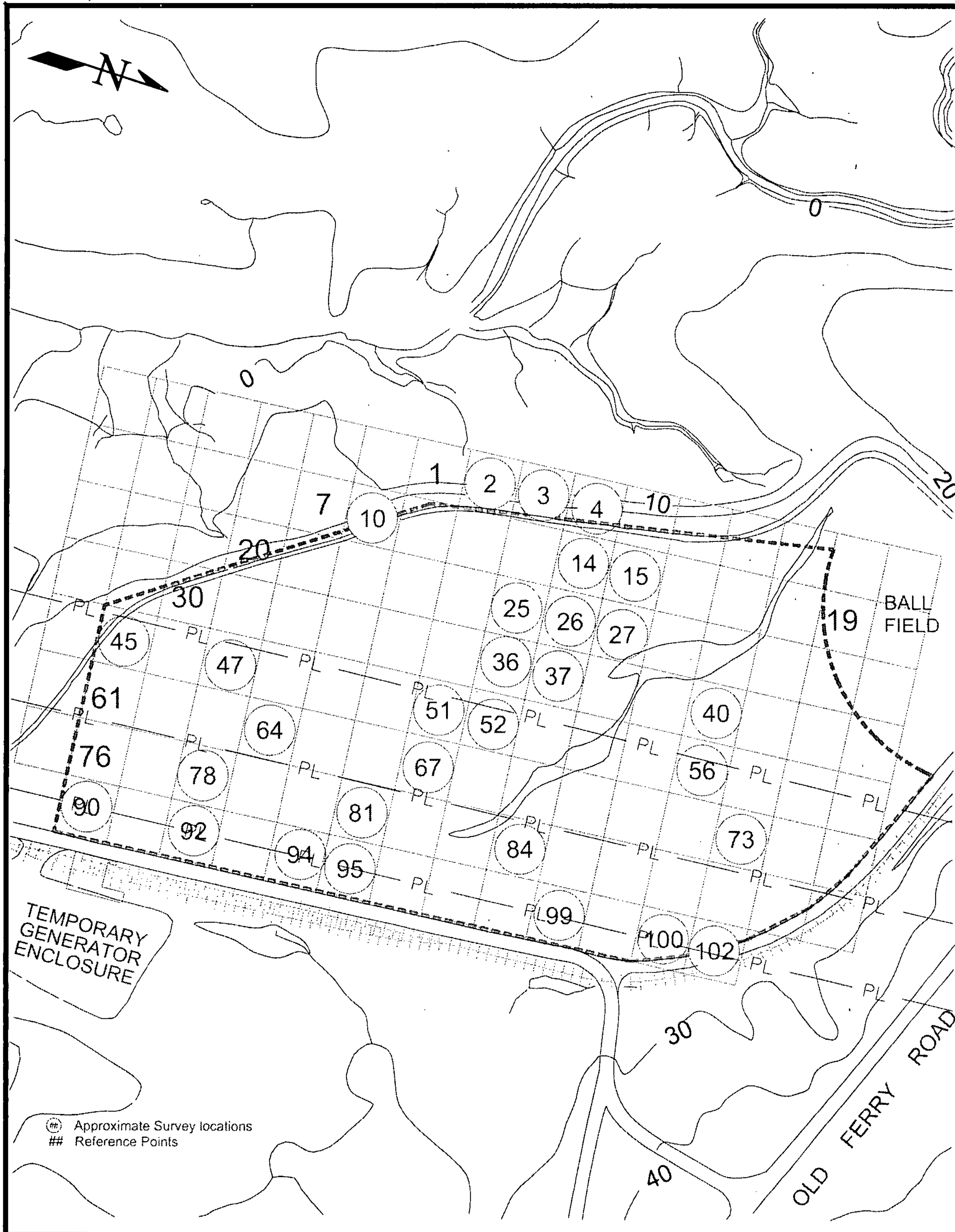
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Ball Field Survey Package R0600-1	Figure 2-11



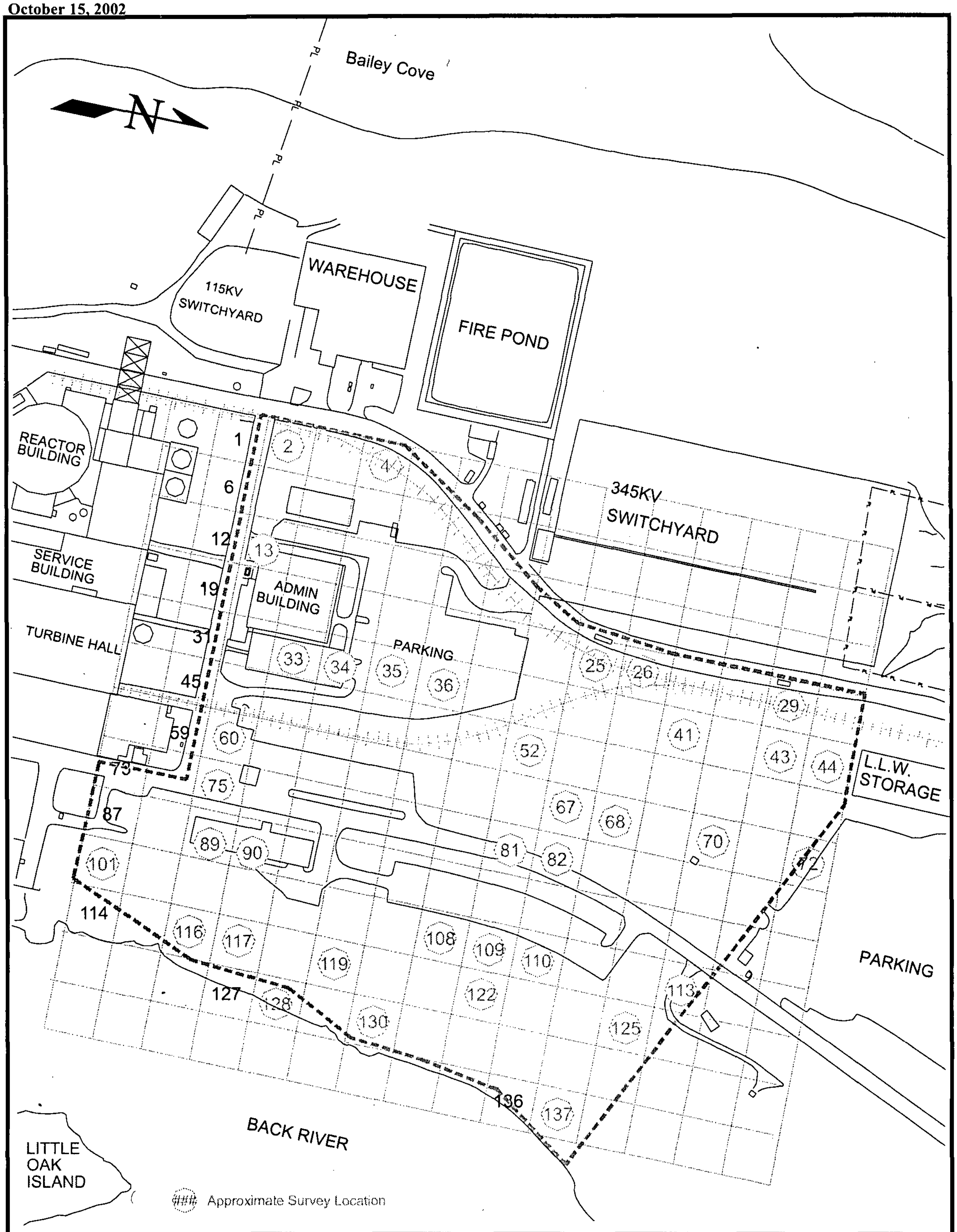
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Ball Field Survey Package R0600-2	Figure 2-12



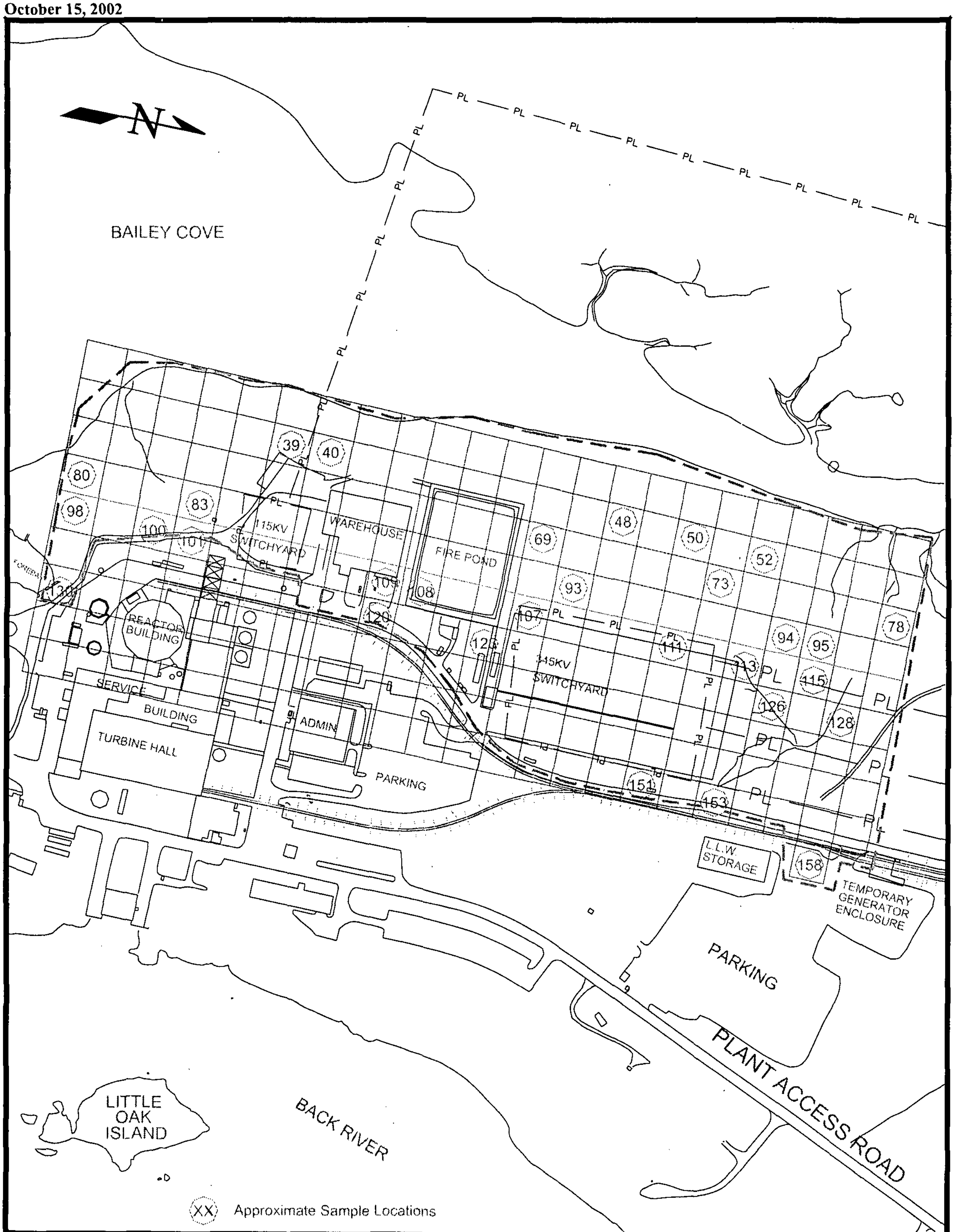
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Ball Field Survey Package R0600-2	Figure 2-13



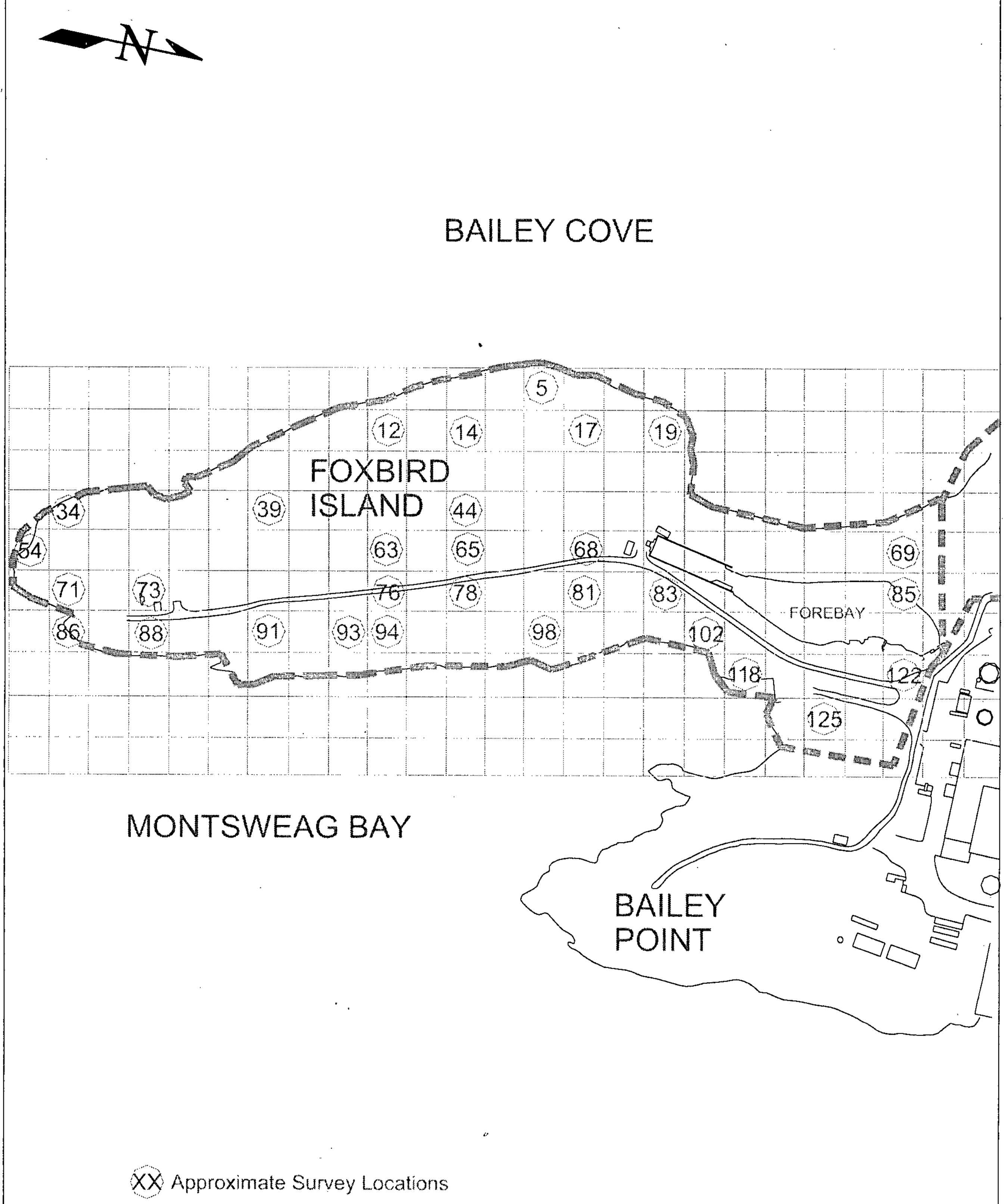
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Construction Debris Landfill Survey Package R0700-2	Figure 2-14



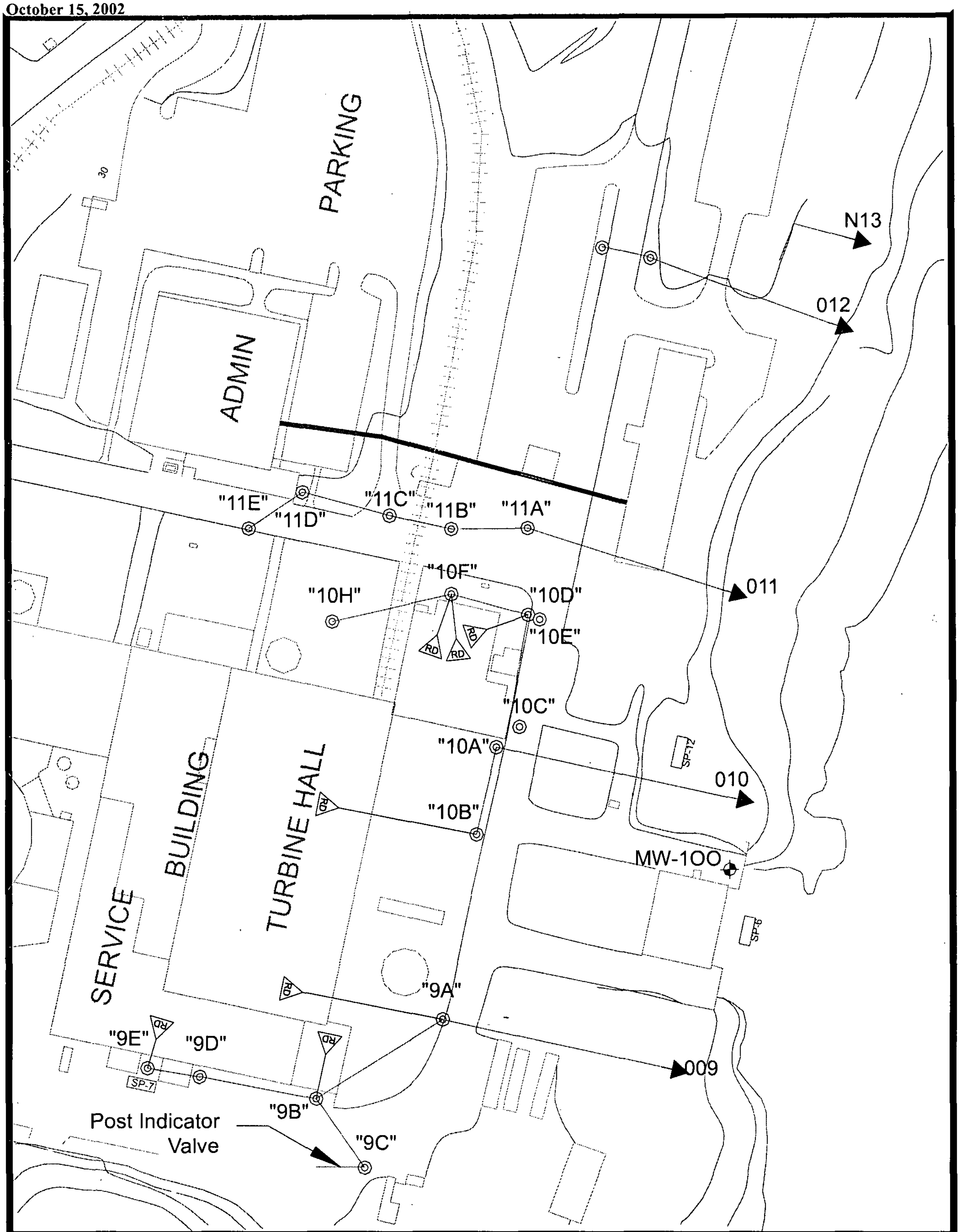
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Admin. and Parking Areas Survey Package R0800-2	Figure 2-15



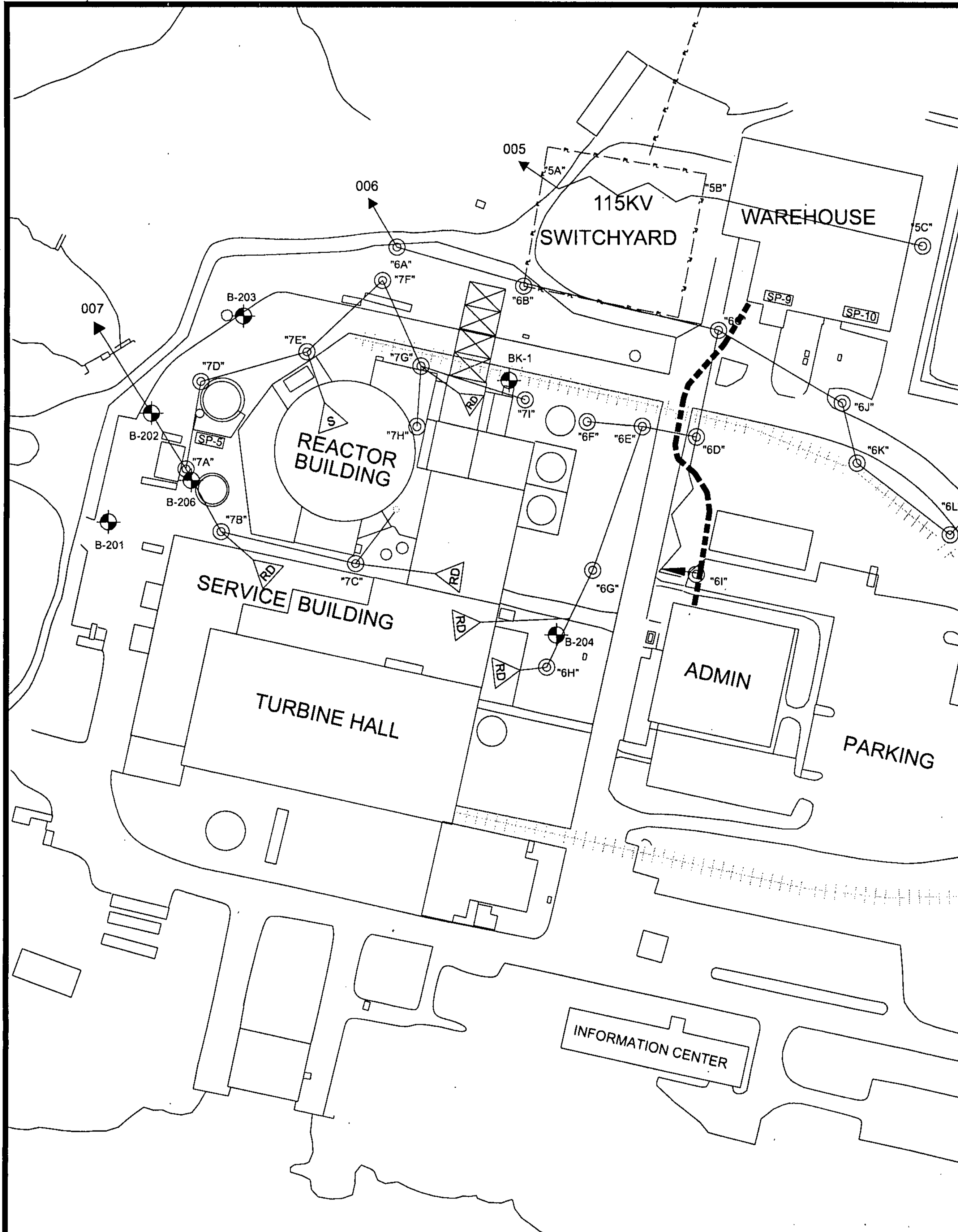
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Balance of Plant Areas Survey Package R0900	Figure 2-16



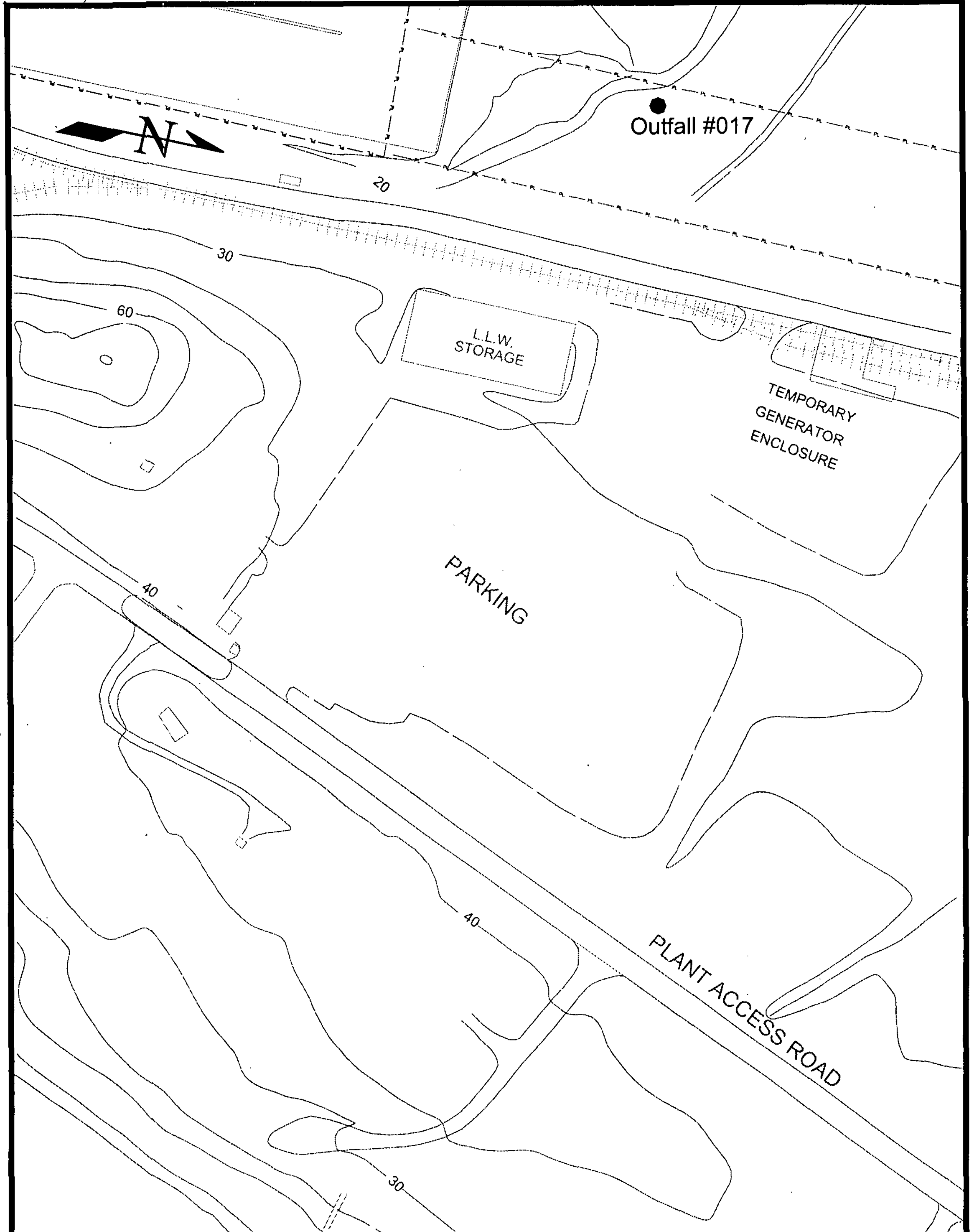
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Foxbird Island Survey Package R1000	Figure 2-17



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Roof and Yard Drains #005, 009-12; Survey Package R1100-1	Figure 2-18



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Roof and Yard Drains #005, 006, and 007 Survey Package R1100-1	Figure 2-19



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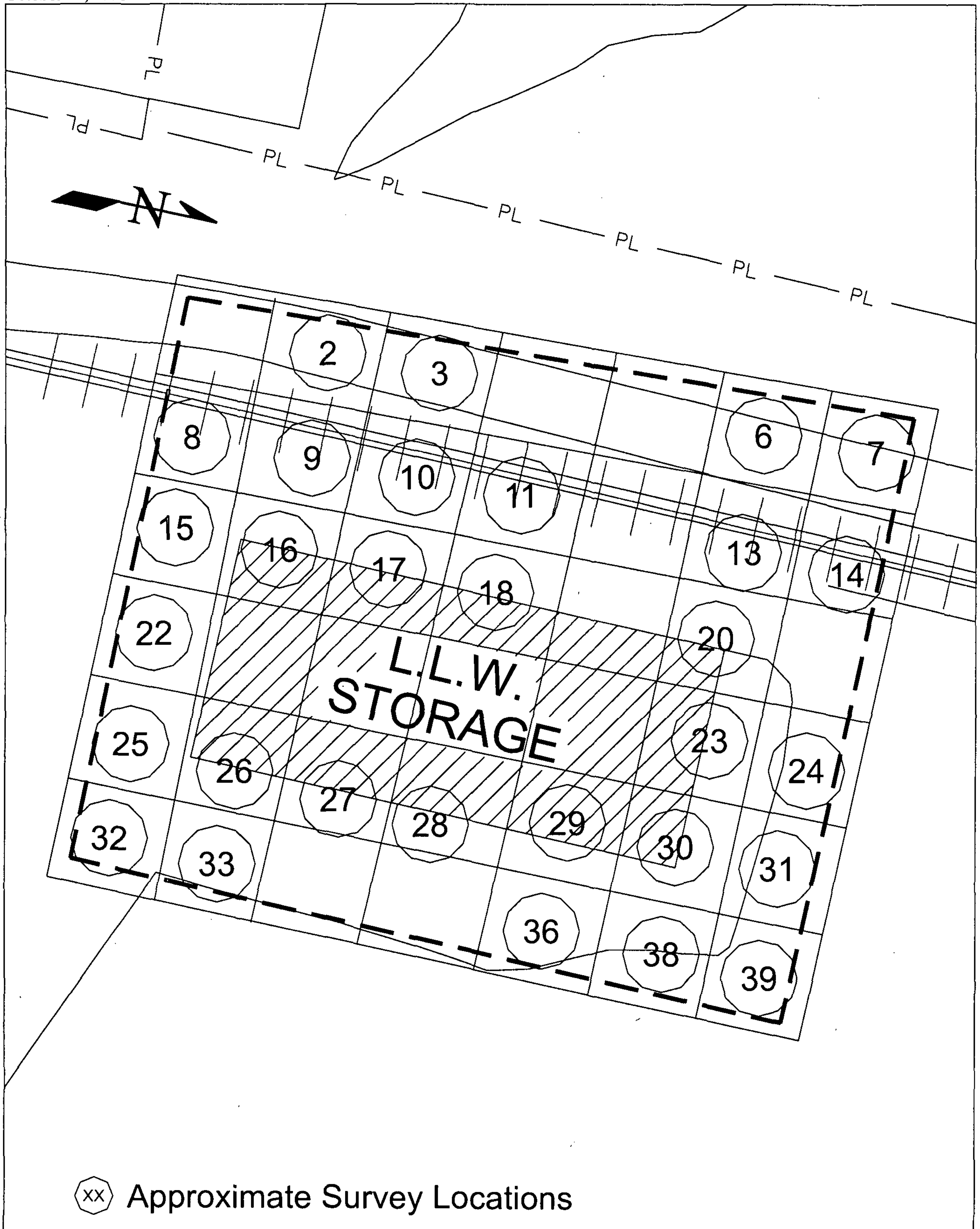
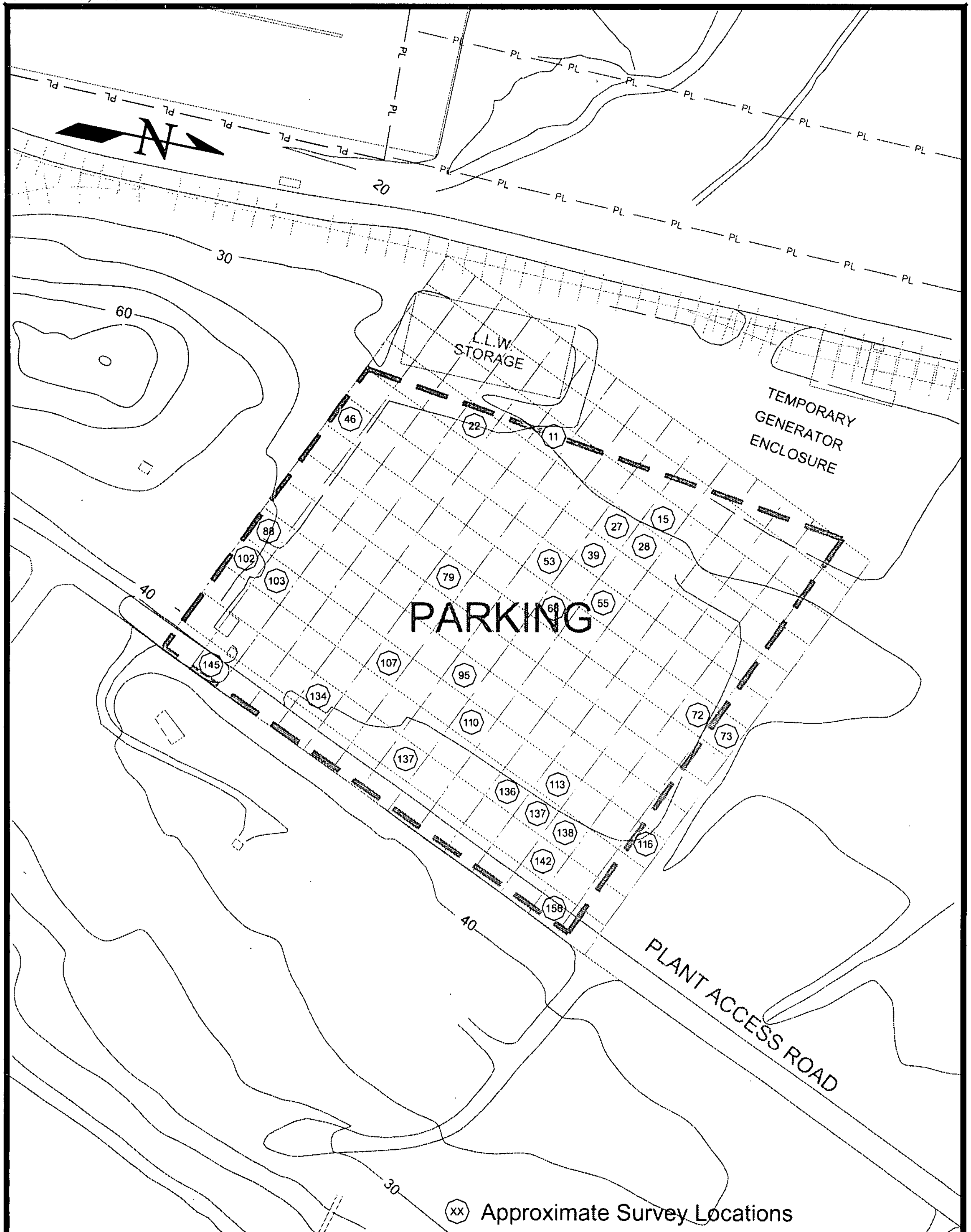
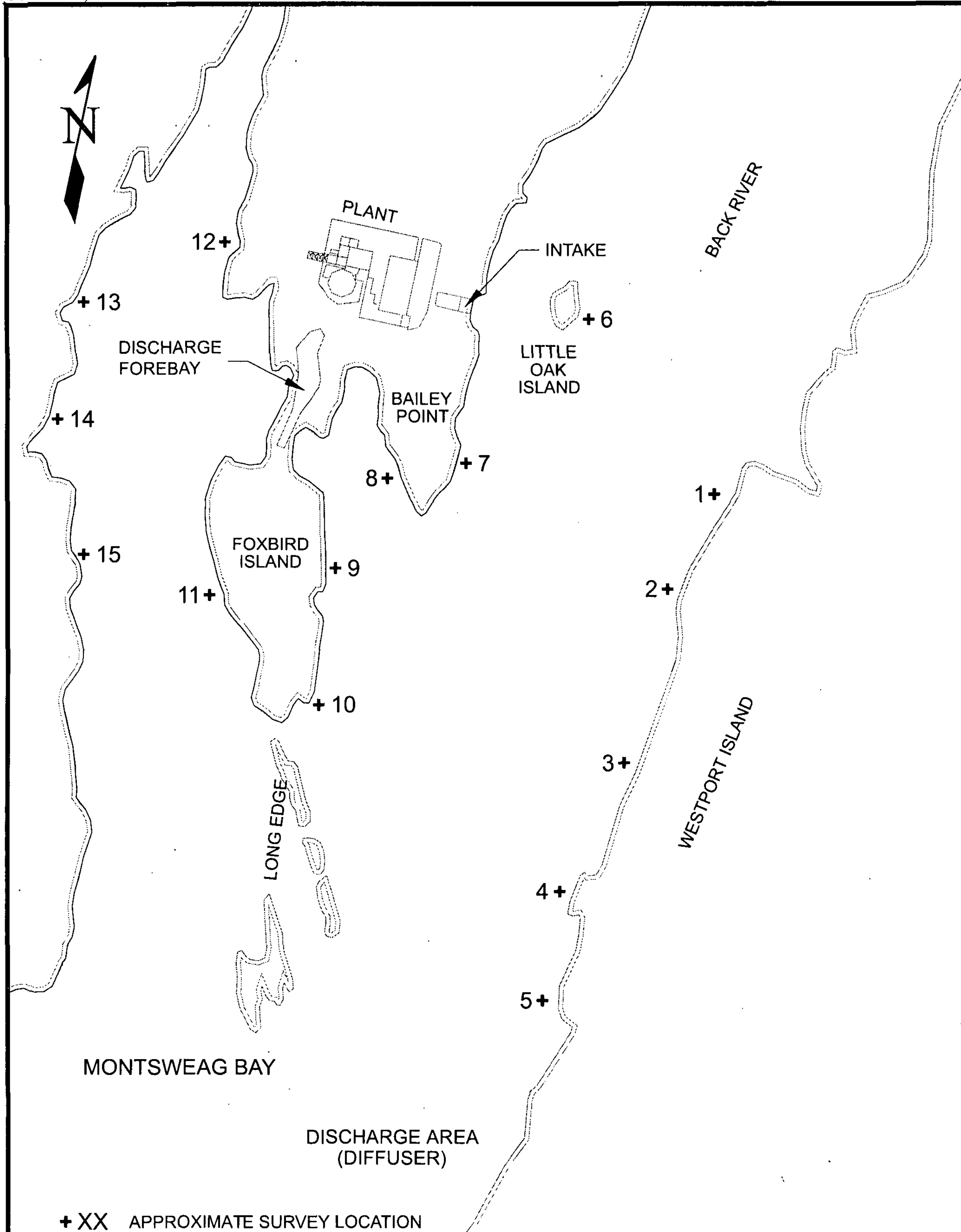


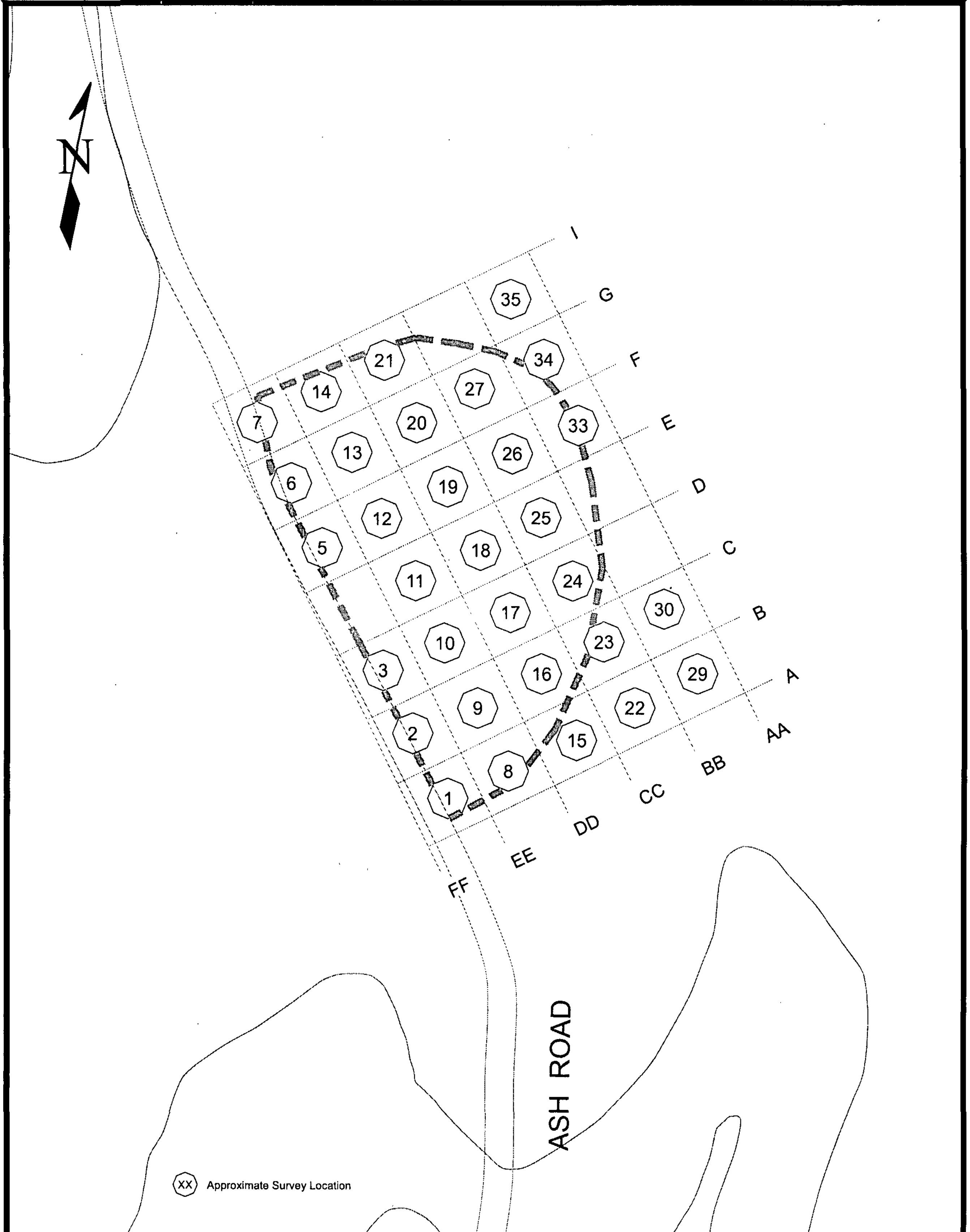
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Low Level Radioactive Waste (LLRW) Storage Yard Survey Package R1200-2	Figure 2-21



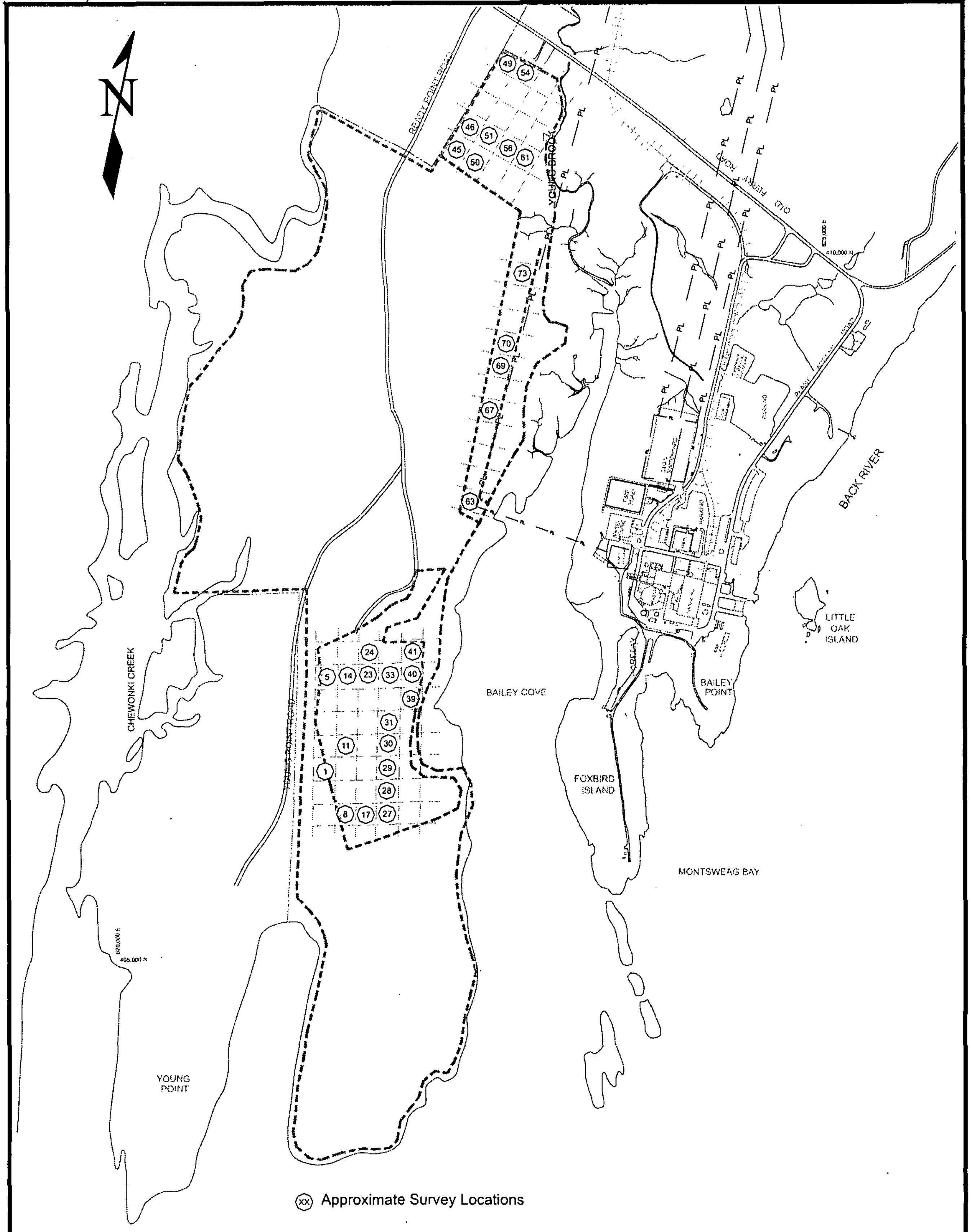
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Dry Cask Storage Area Survey Package R1300-2	Figure 2-22



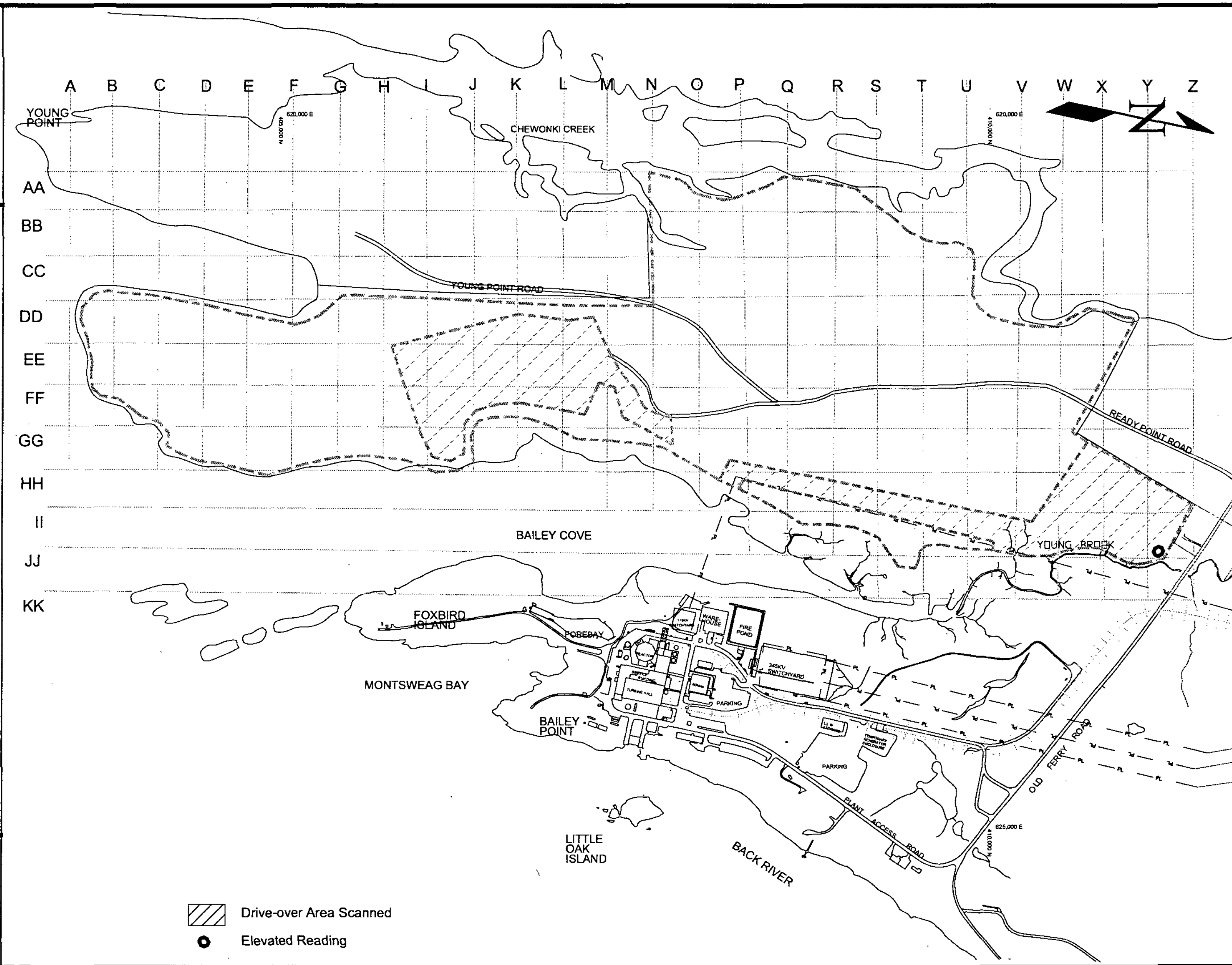
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Westport, Montsweag Bay, Bailey Point and Cove, and Plant Area Shorelines Survey Package R1400-1	Figure 2-23



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Ash Road Rubble Piles Survey Package R1500-2	Figure 2-24



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Owner Controlled Area Survey Package R1600-4	Figure 2-25



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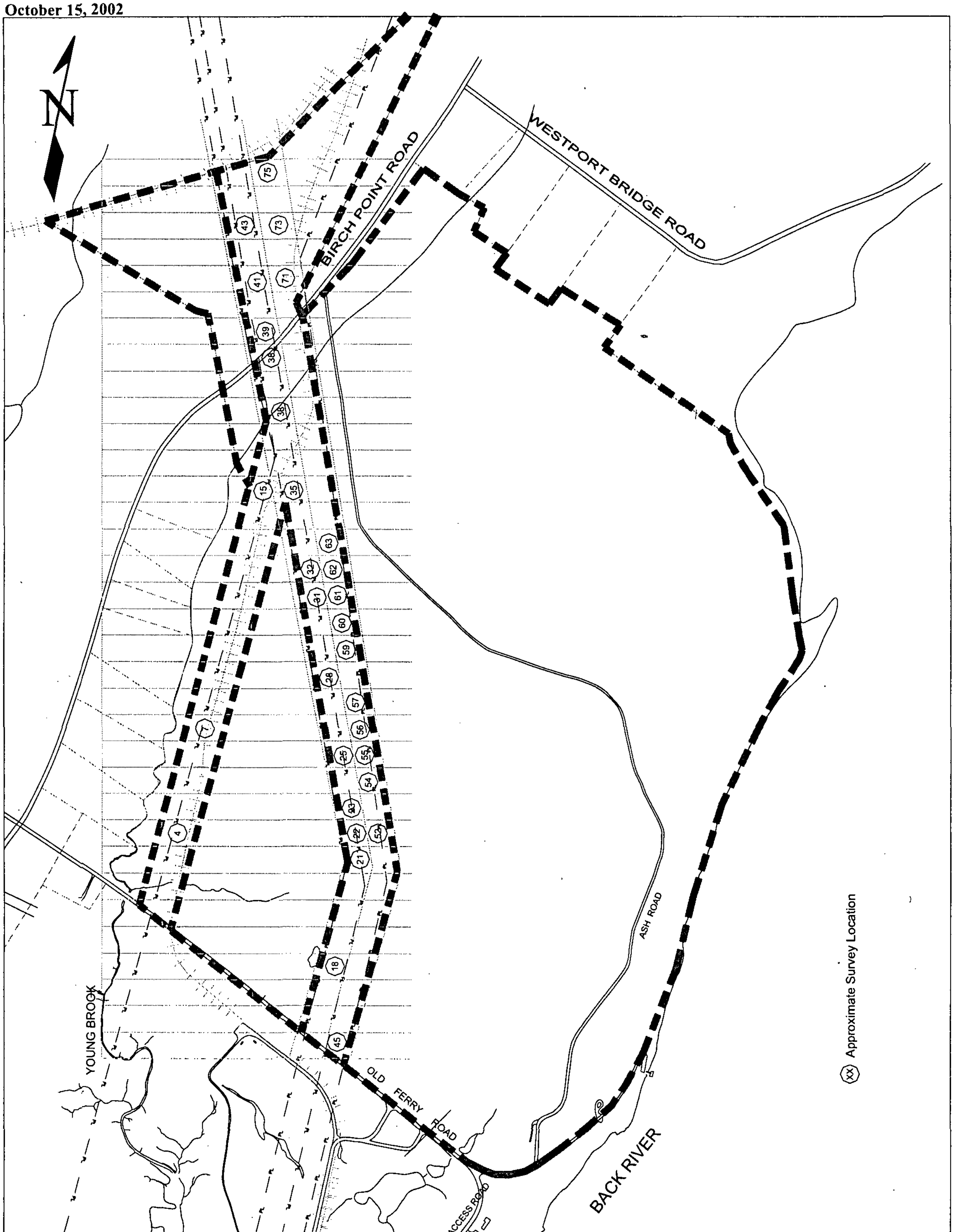
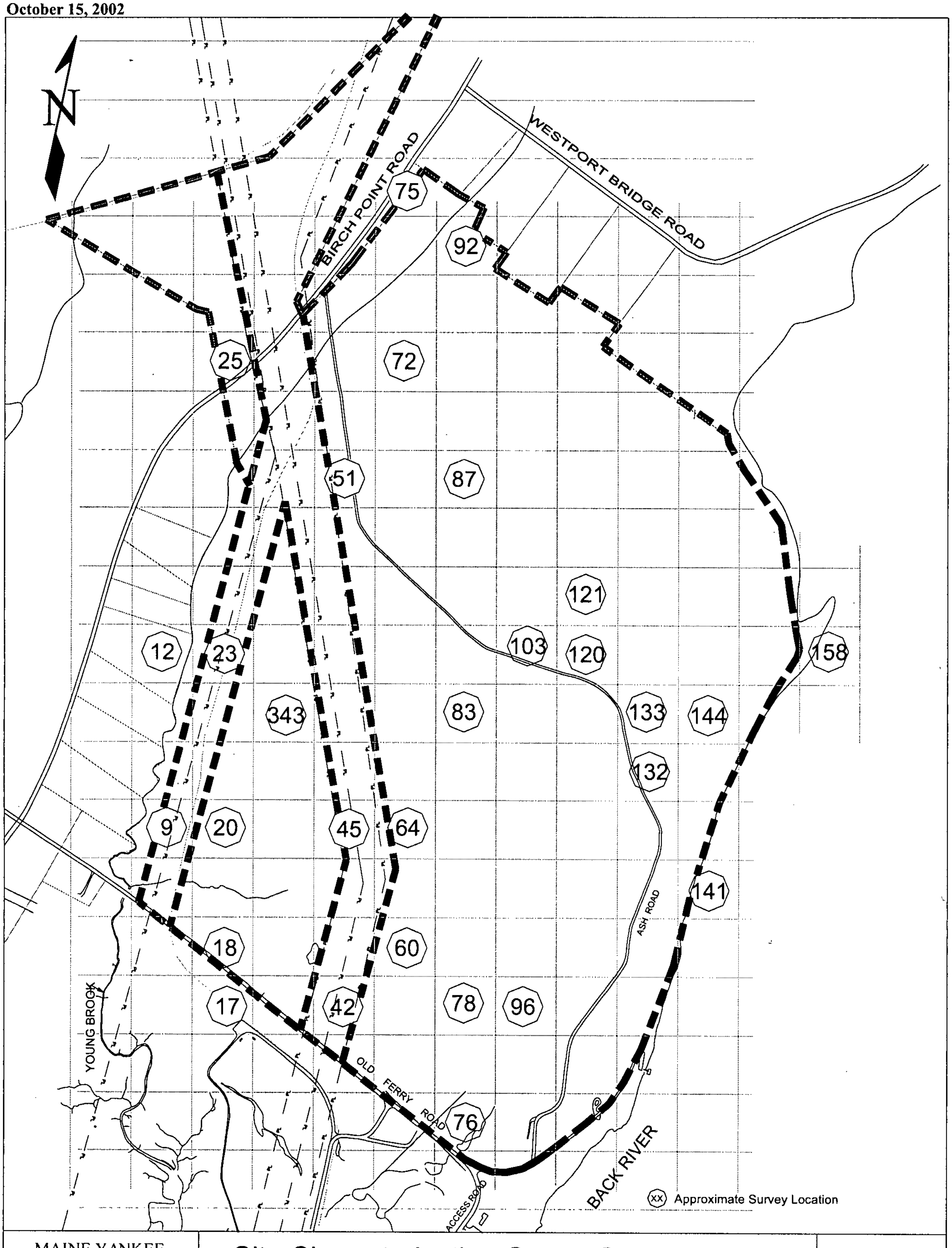
Site Characterization Owner Controlled Area Survey Package R1600-4

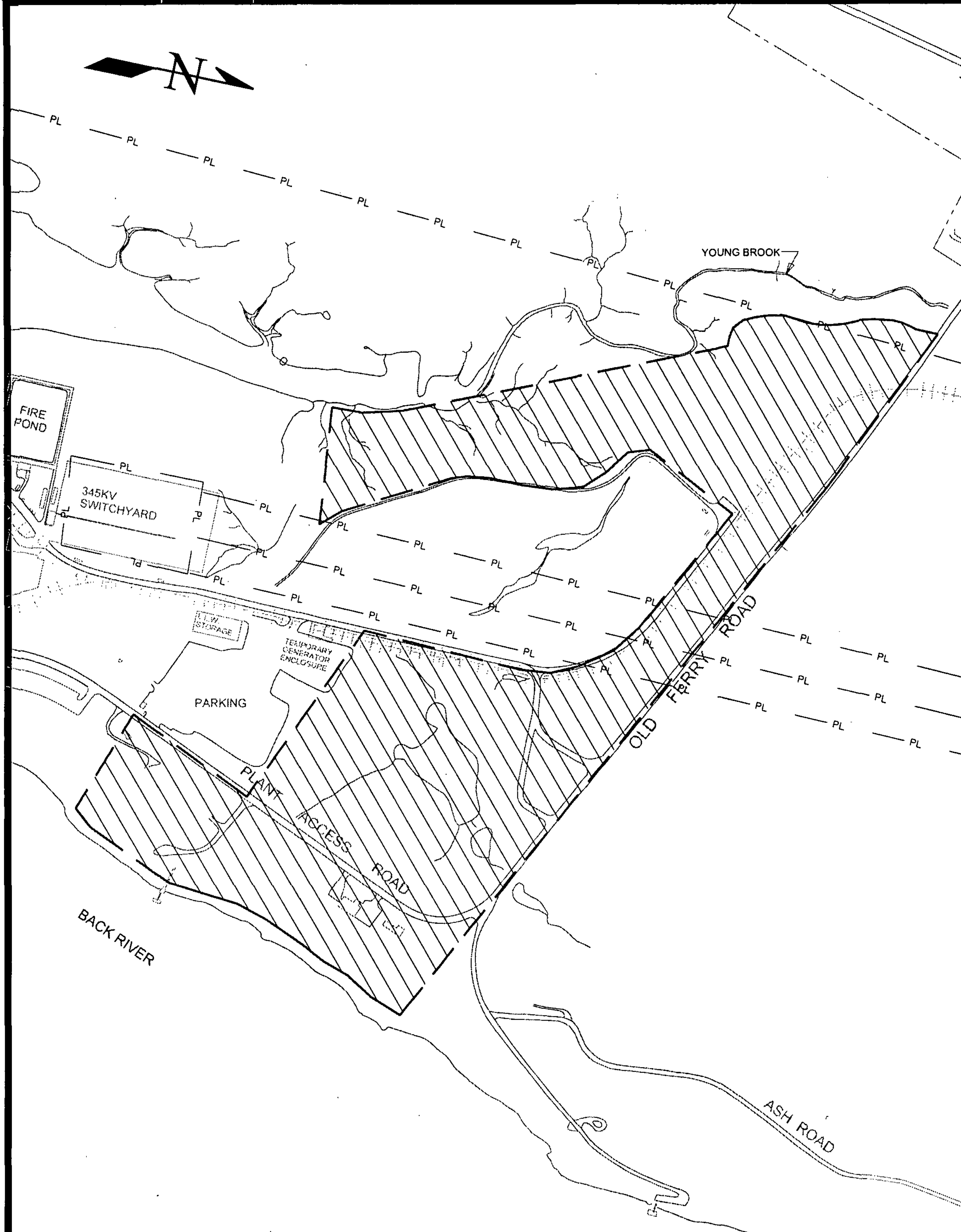
Figure 2-26



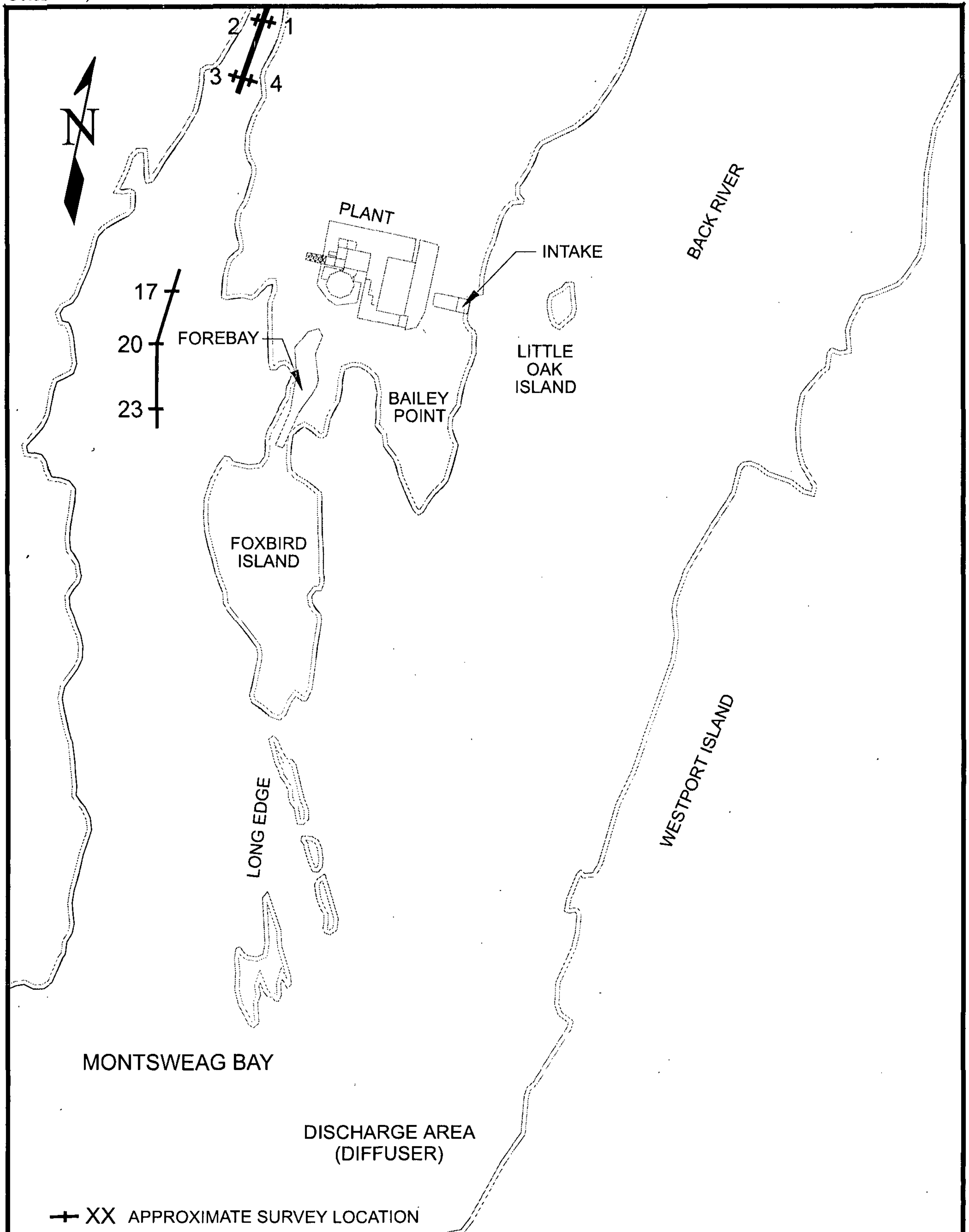
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Owner Controlled Area North of Old Ferry RD Survey Package R1700-1	Figure 2-27



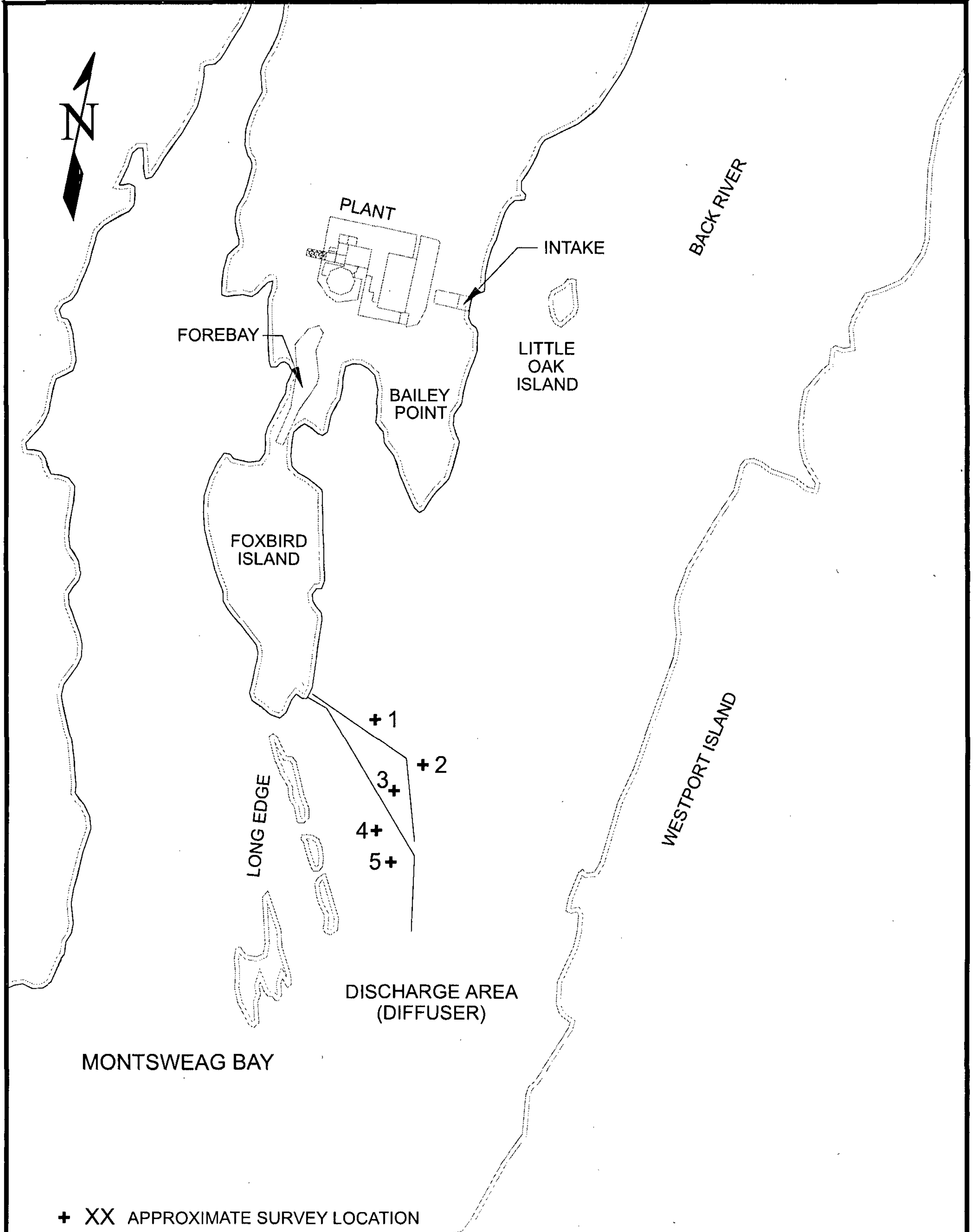
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Owner Controlled Area North of Old Ferry RD Survey Package R1700-4	Figure 2-28



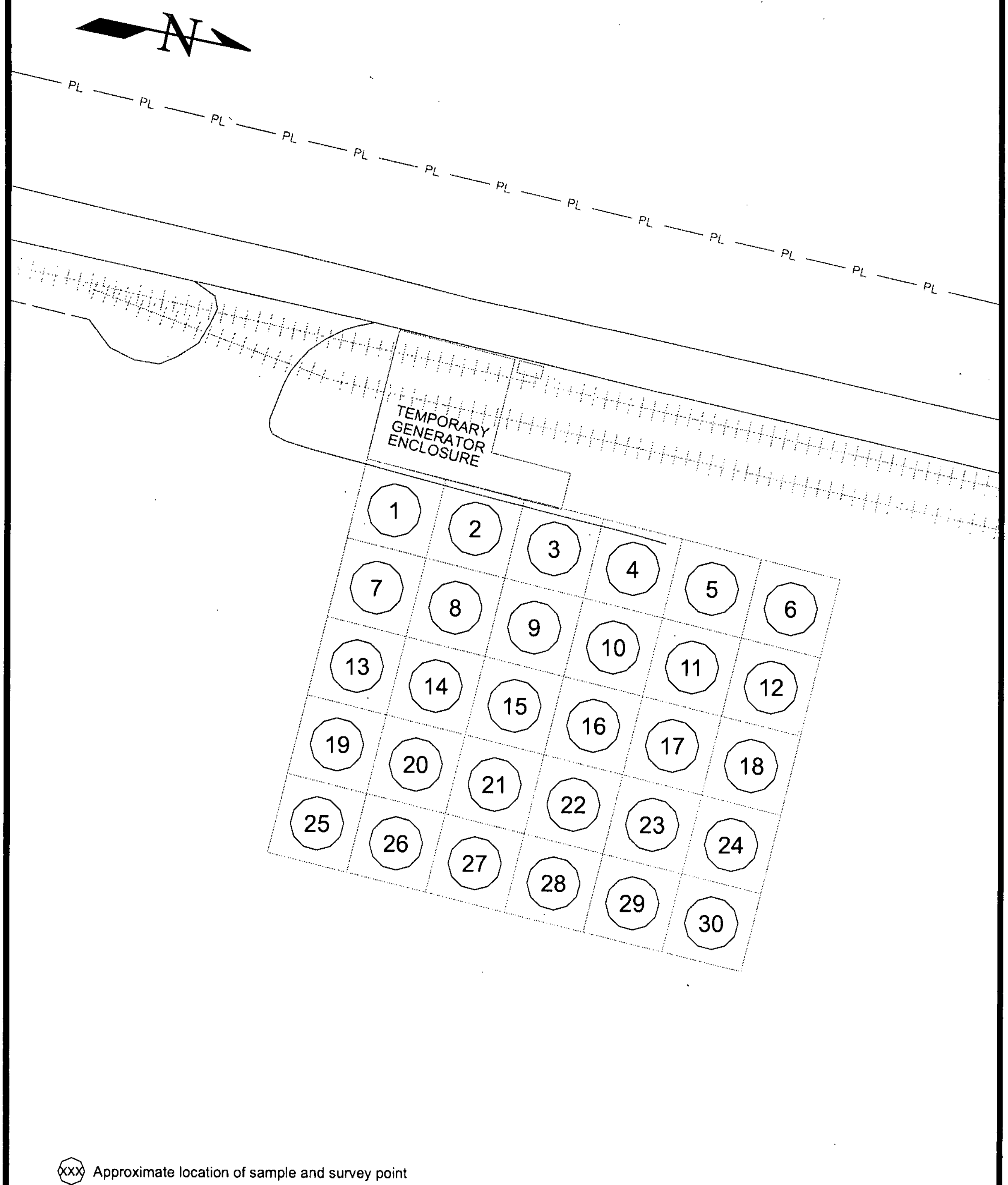
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Bailey House Area Survey Package R1800-1	Figure 2-29



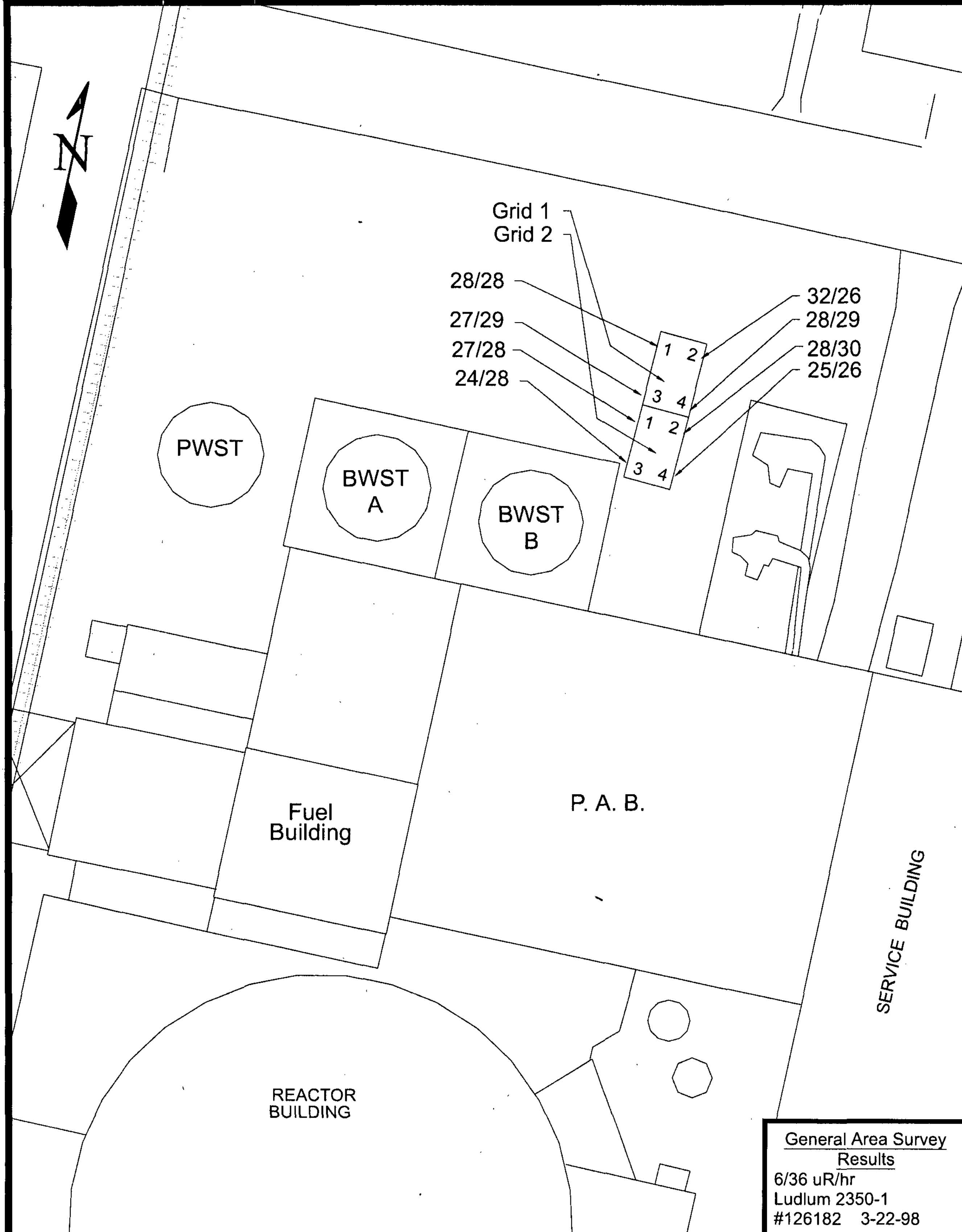
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Bailey Cove Survey Package R1900	Figure 2-30



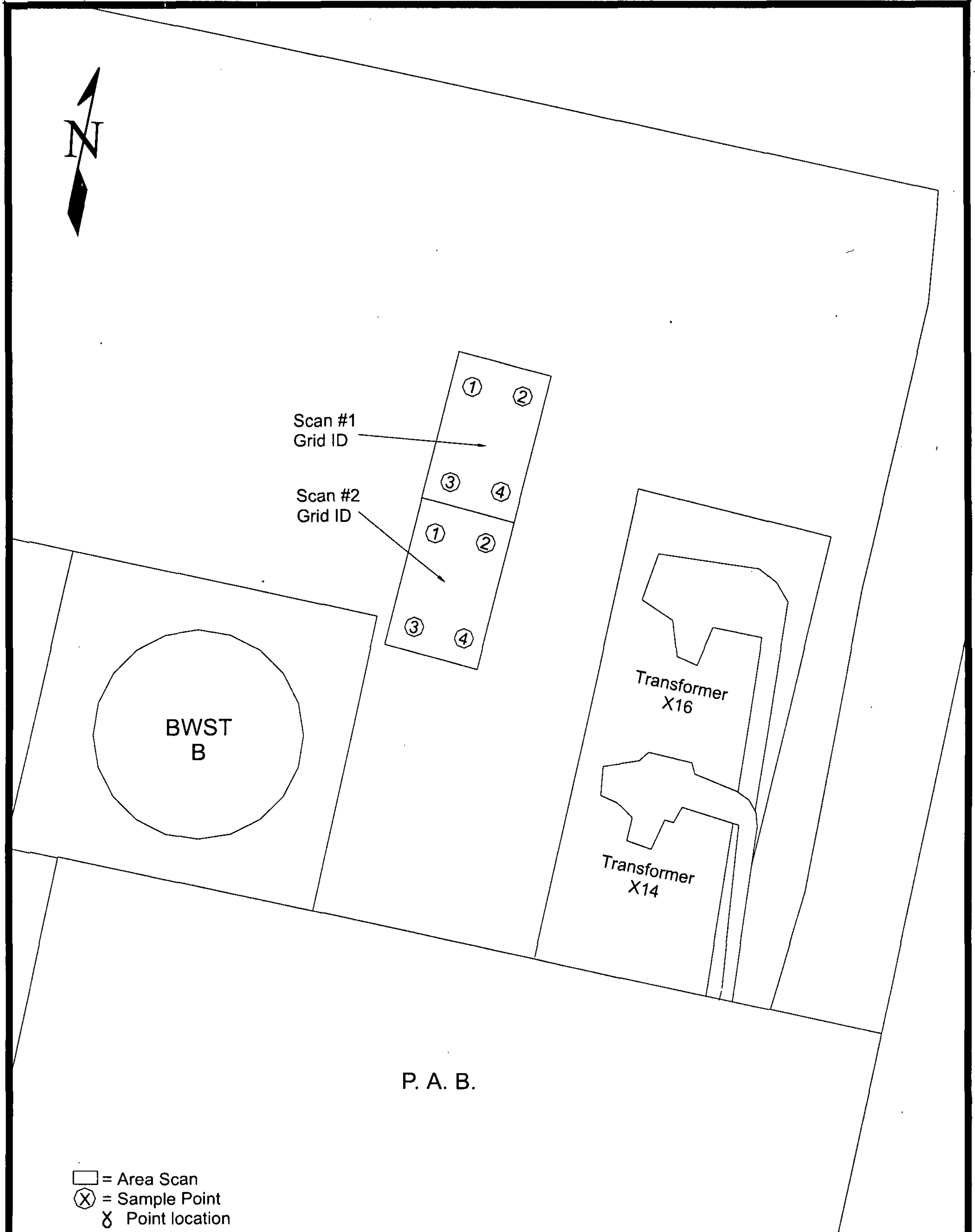
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Diffusers Survey Package R2000	Figure 2-31



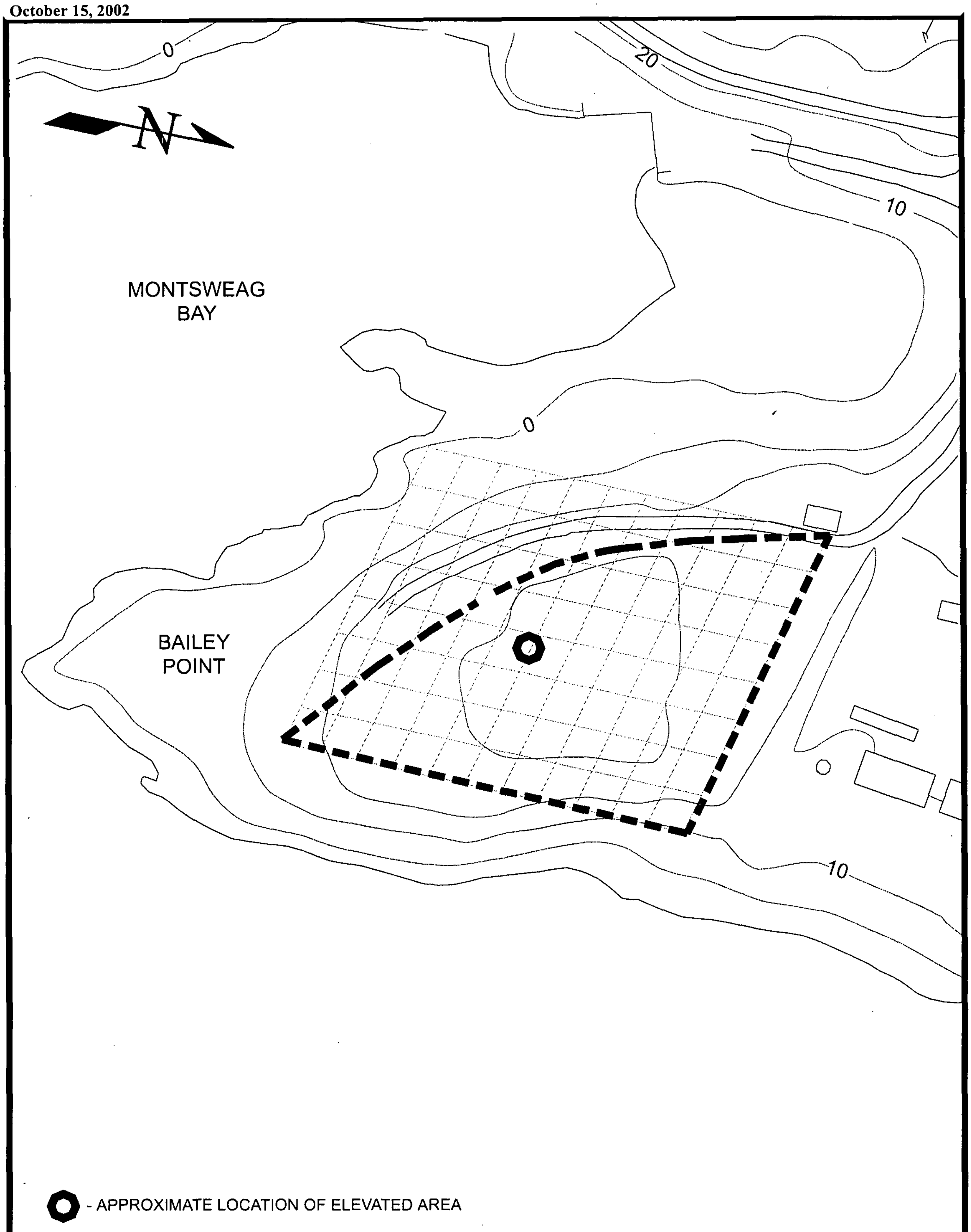
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Maintenance Yard (Stockyard) Survey Package R2100	Figure 2-32



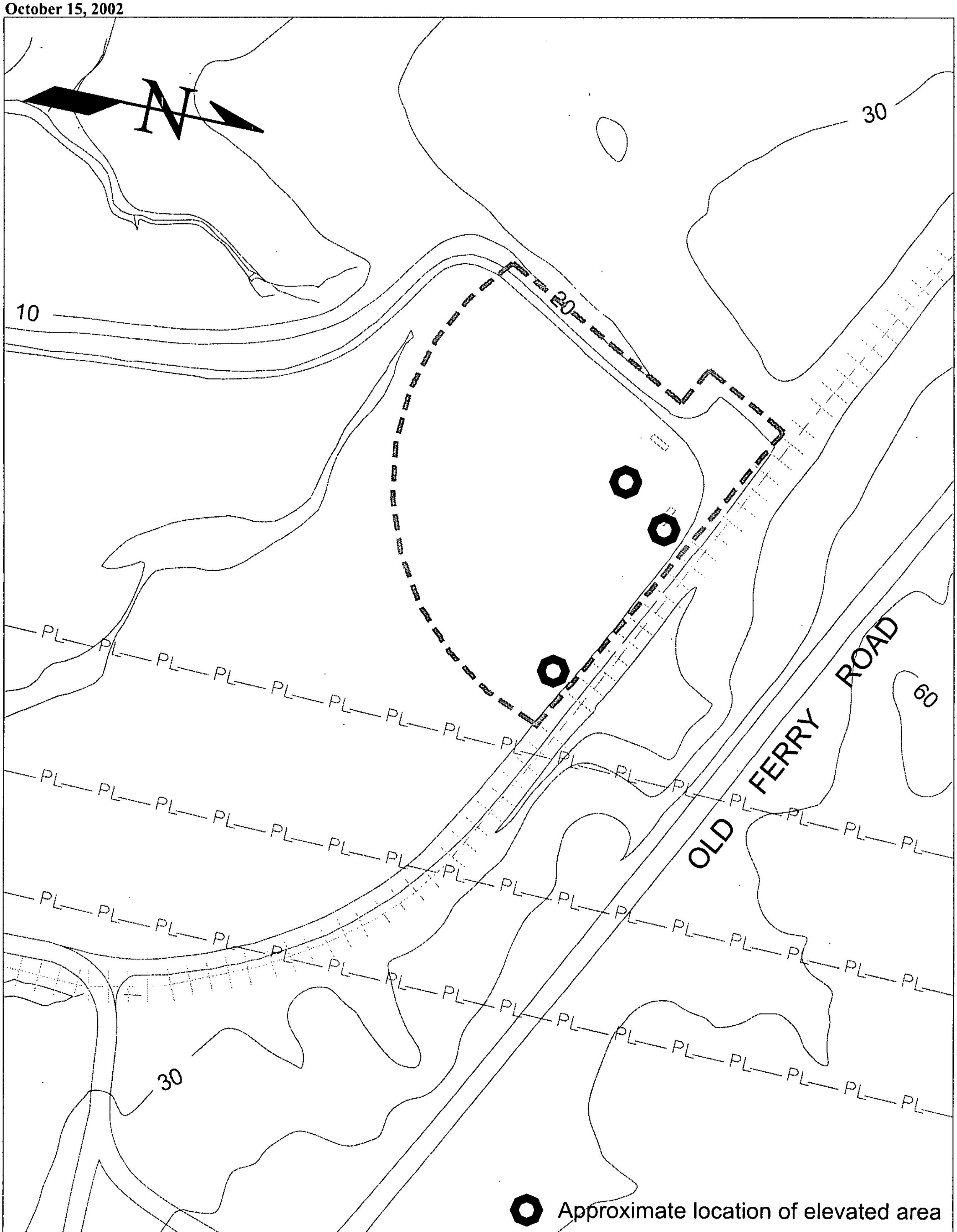
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Roof SFPI Substation Slab Survey Package R2300-1	Figure 2-33



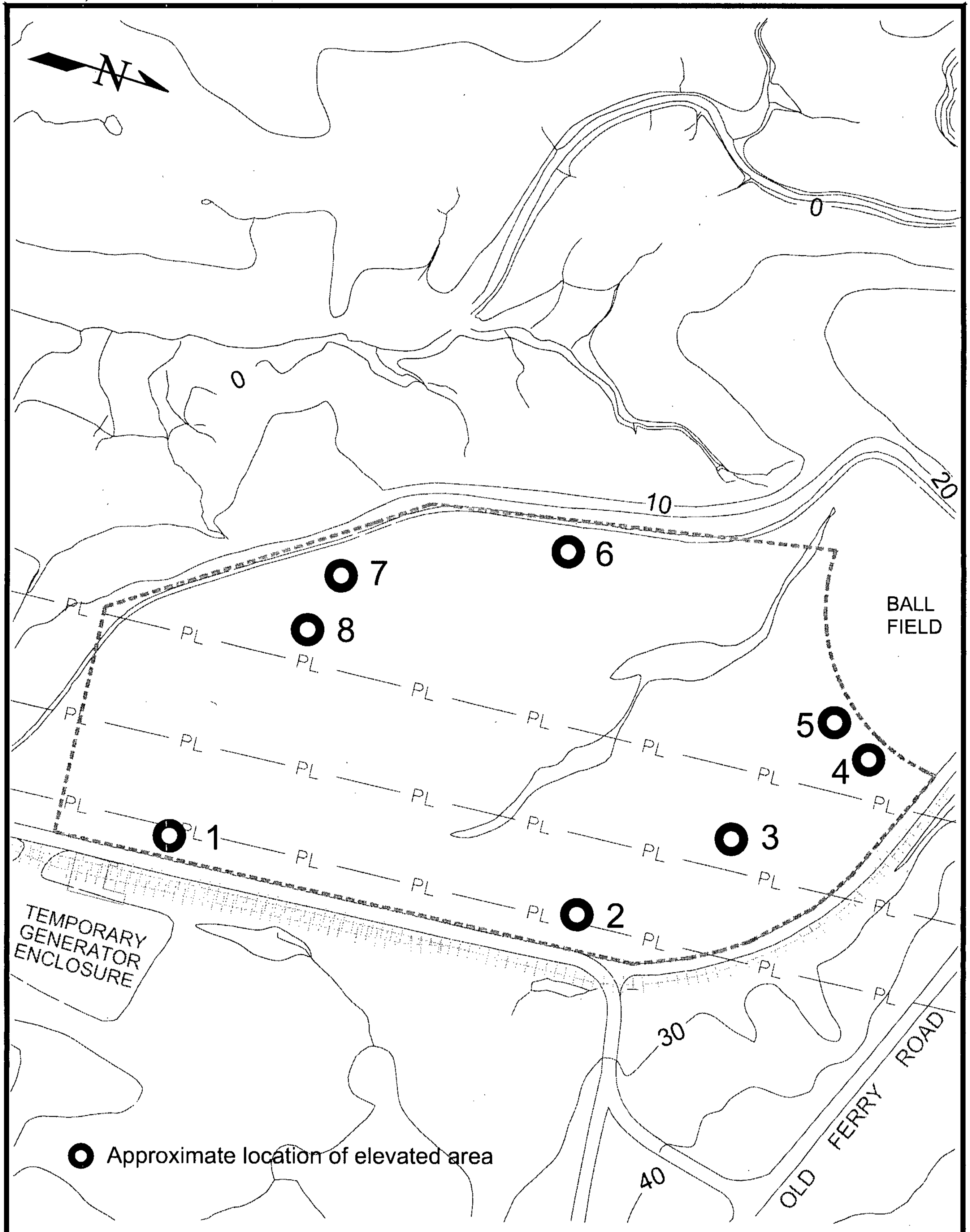
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Roof SFPI Substation Slab Survey Package R2300-2	Figure 2-34



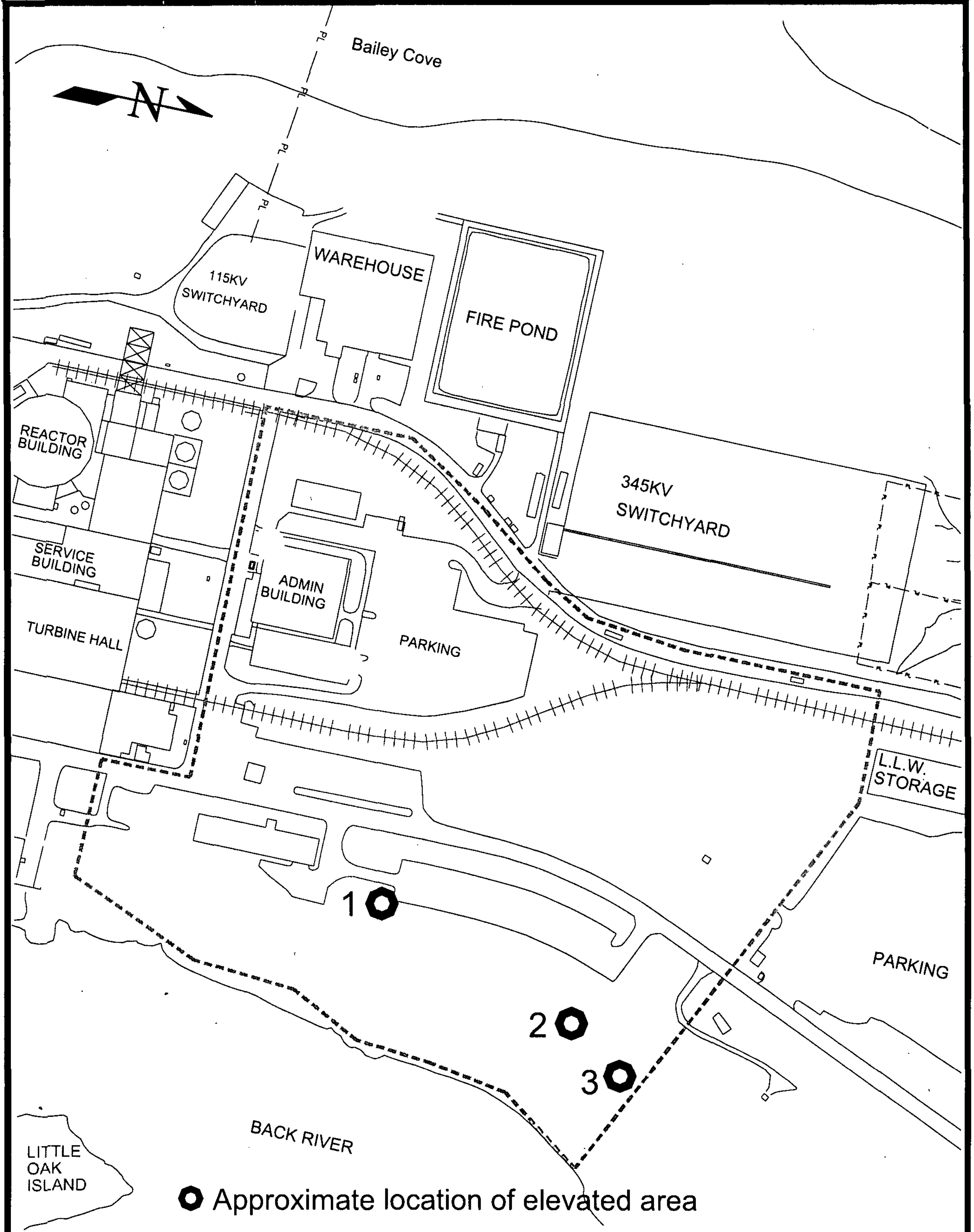
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Drive Over Elevated Areas Survey Package R2500-1	Figure 2-35



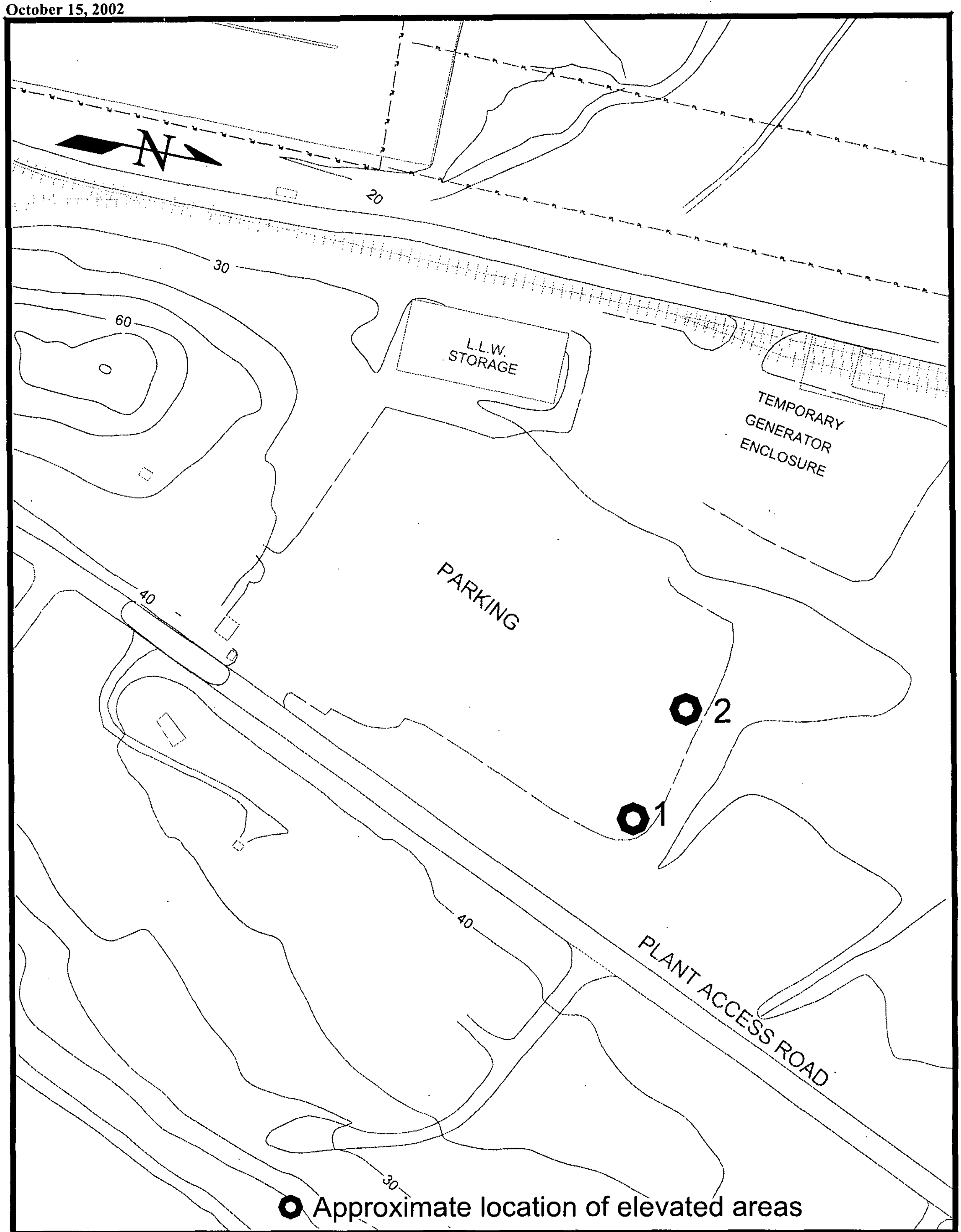
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Drive Over Area Survey Package R2500-2	Figure 2-36



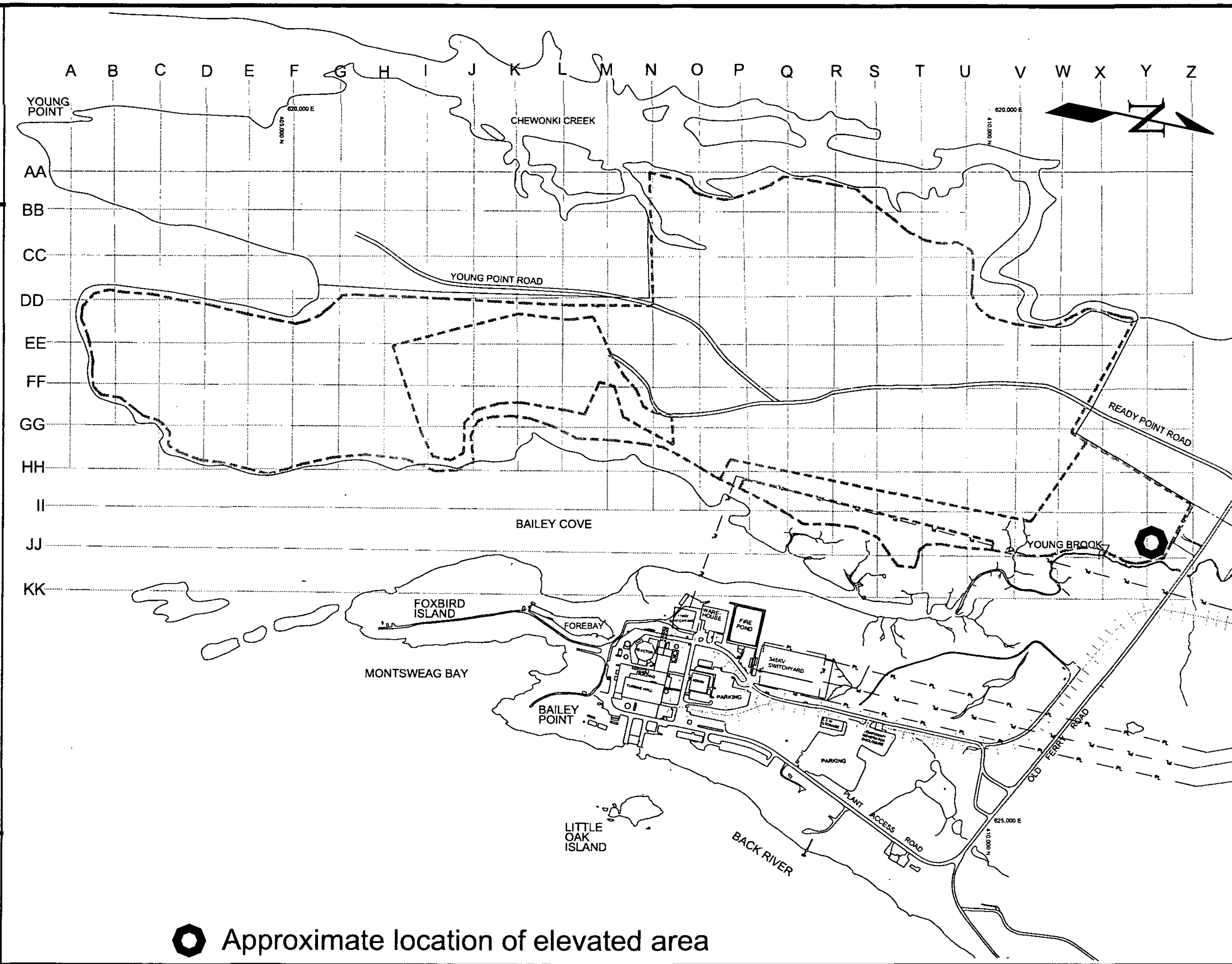
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Drive Over Elevated Areas Survey Package R2500-3	Figure 2-37



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Drive Over Areas Survey Package R2500-4	Figure 2-38



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Drive Over Elevated Areas Survey Package R2500-6	Figure 2-39



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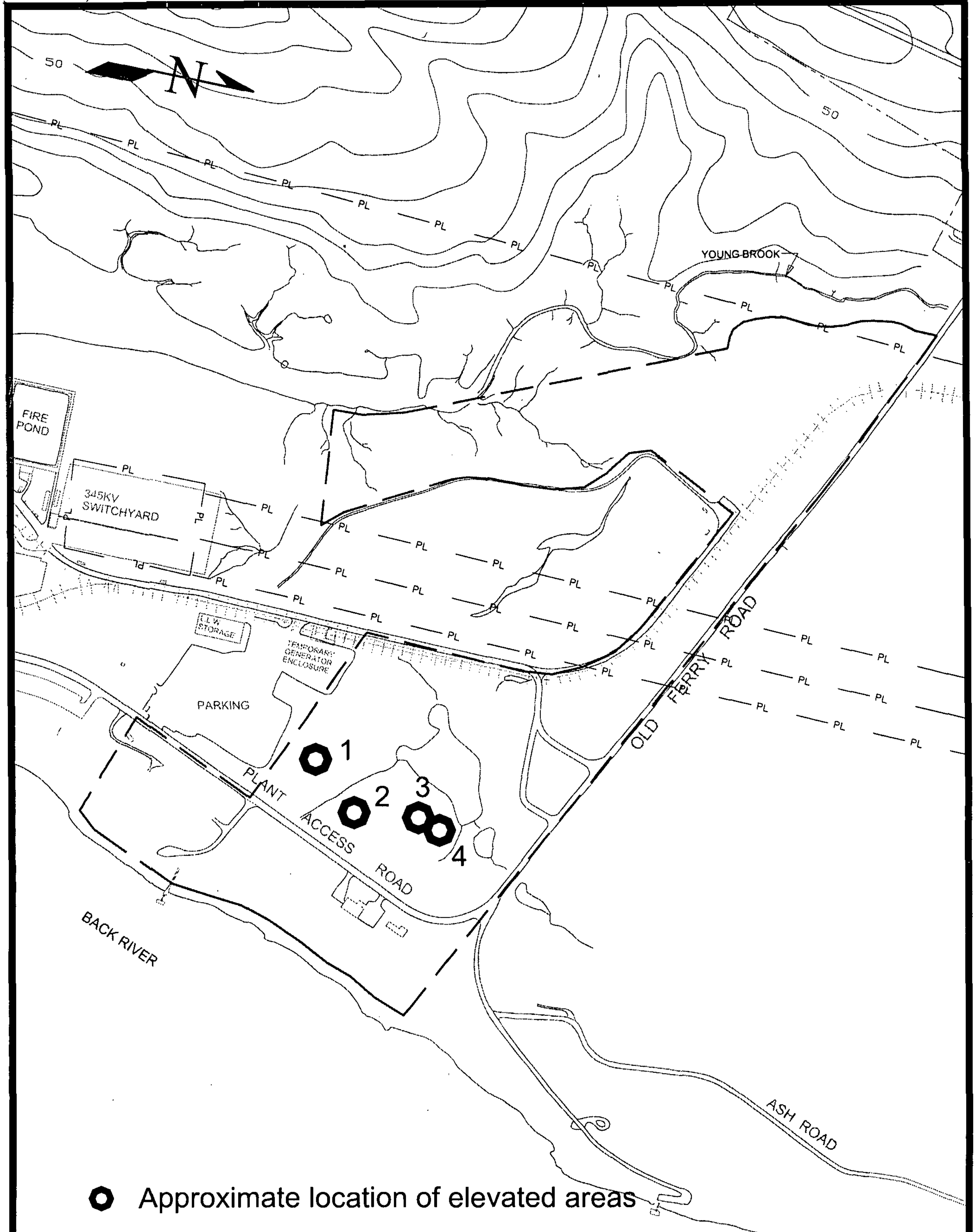
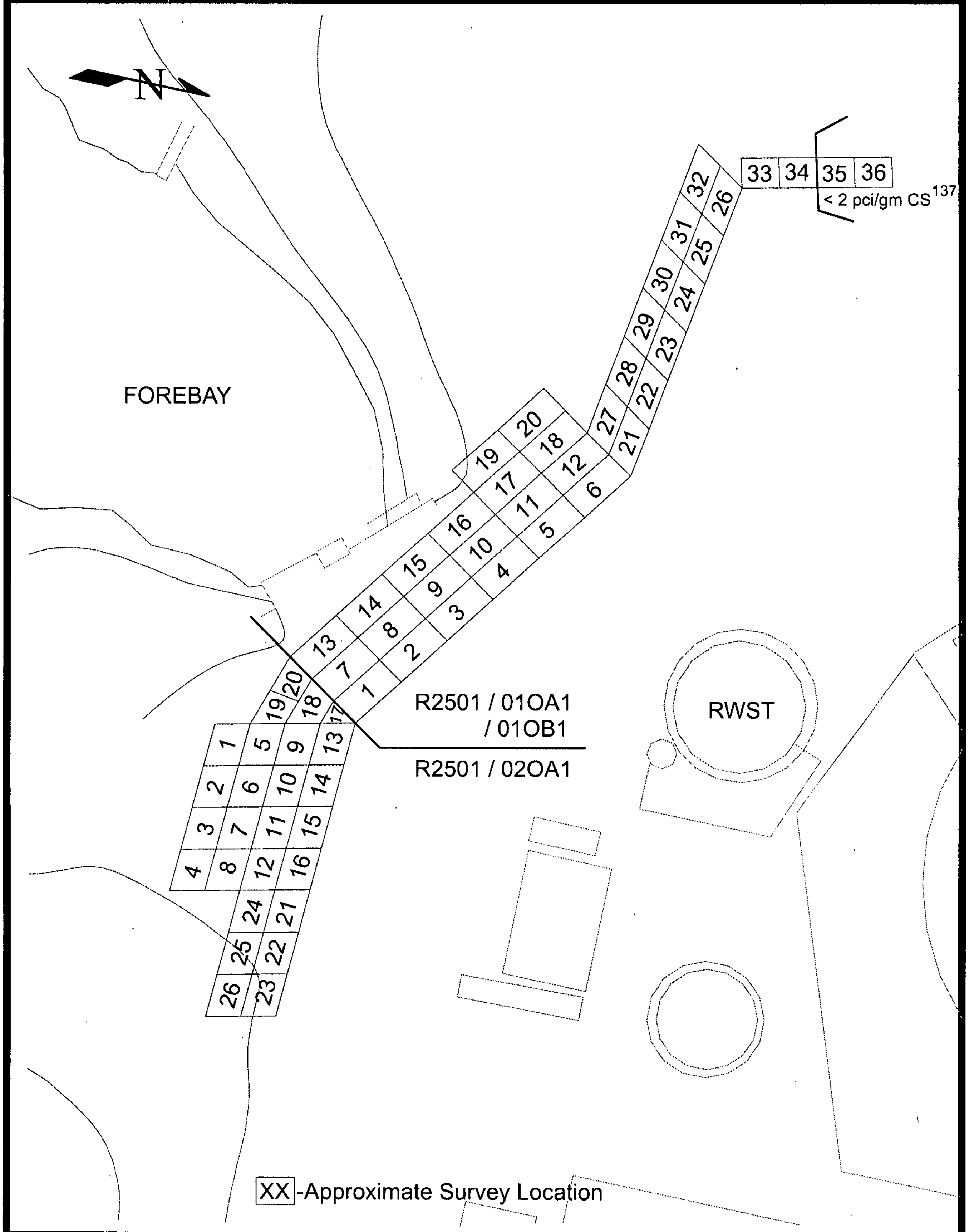
Site Characterization Drive Over Elevated Areas Survey Package R2500-7

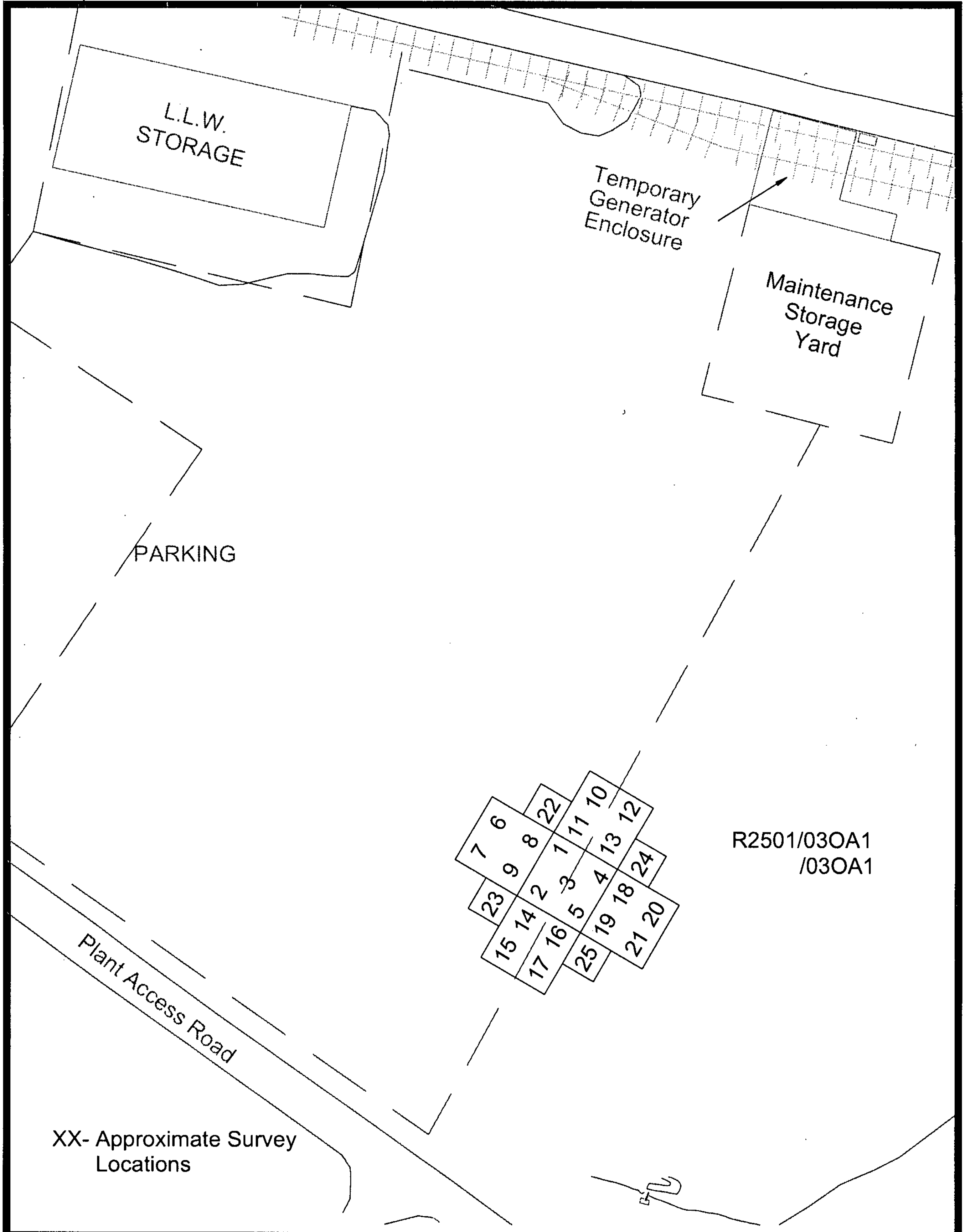
Figure 2-40



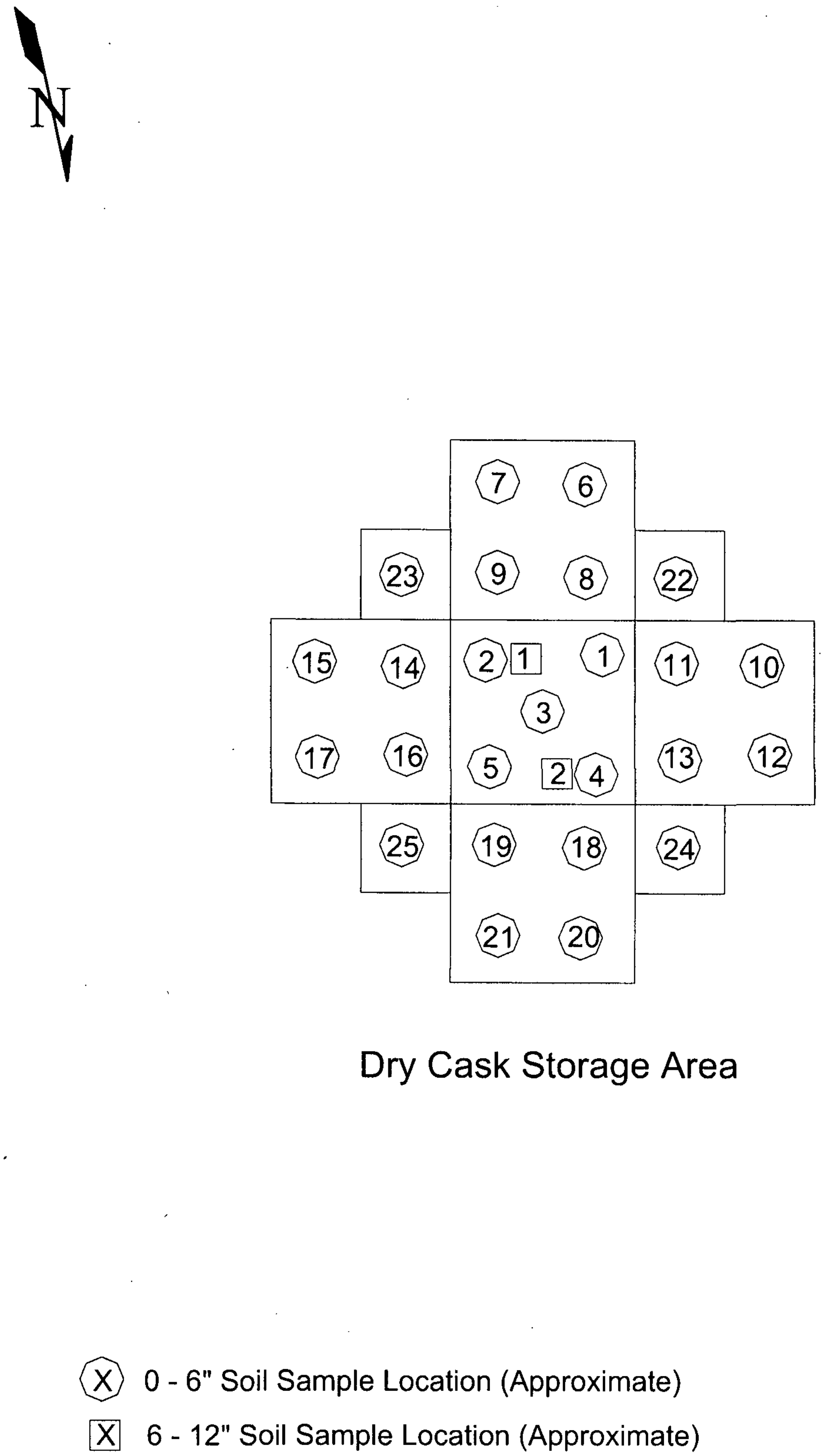
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Drive Over Elevated Areas Survey Package R2500-8	Figure 2-41



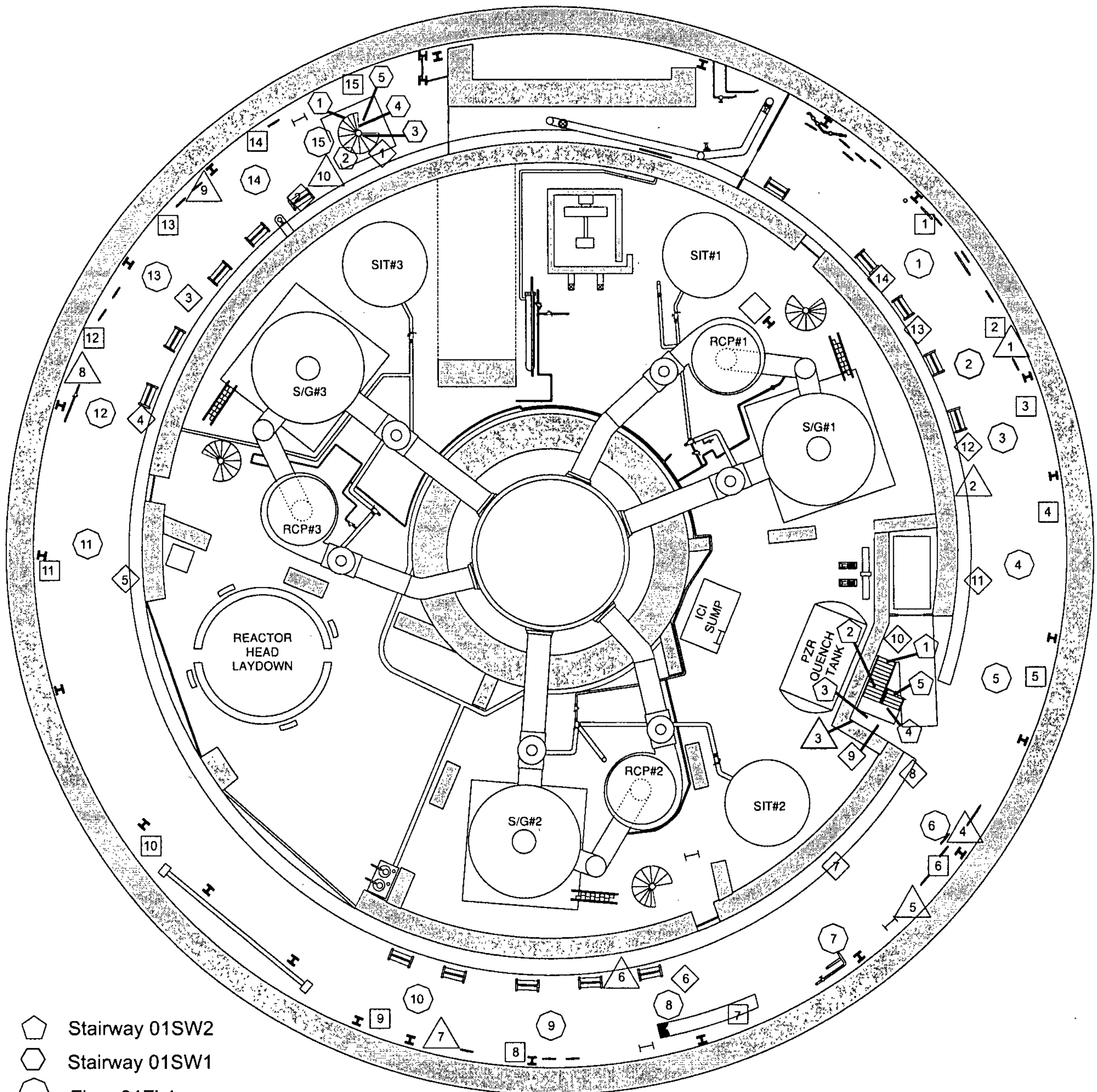
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Forebay Area Survey Package R2501	Figure 2-42



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Dry Cask Storage Area Survey Package R2501	Figure 2-43



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Follow-up Sampling At Elevated Soil Sample Locations Survey Package	Figure 2-44



Stairway 01SW2

Stairway 01SW1

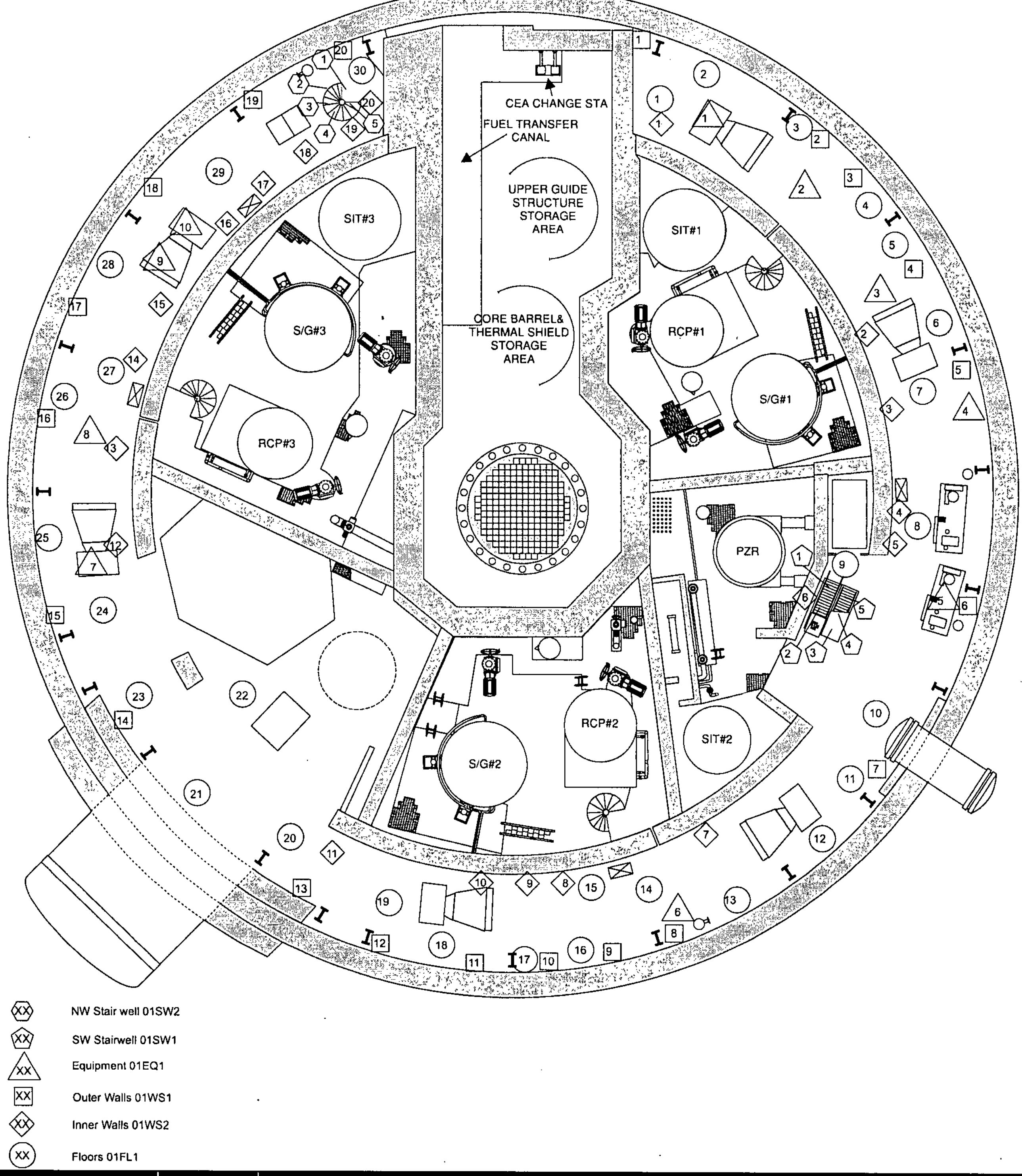
Floor 01FL1

Walls 01WS2

Walls 01WS1

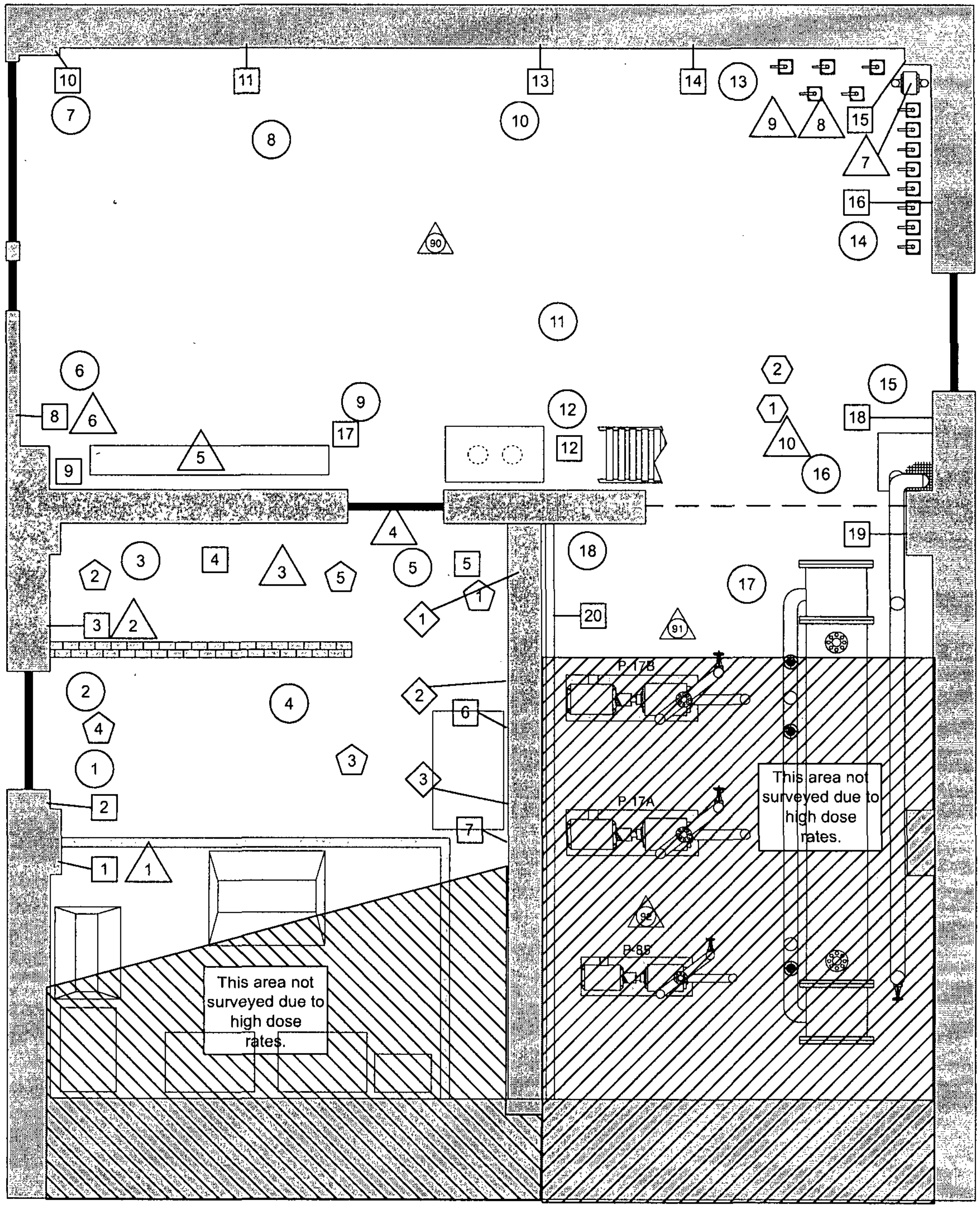
Equipment 01EQ1

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Containment Building -2ft Elevation Survey Package A0100	Figure 2-45



Site Characterization 20 ft Elevation
Survey Package A0200

Figure 2-46



XX (hexagon)	Stair well
XX (pentagon)	Floor Drains
XX (triangle)	Equipment 01EQ1
XX (square)	Walls
XX (diamond)	Floor Drain
XX (circle)	Floors

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Fuel Building 21 ft Elevation Survey Package A0400	Figure 2-47

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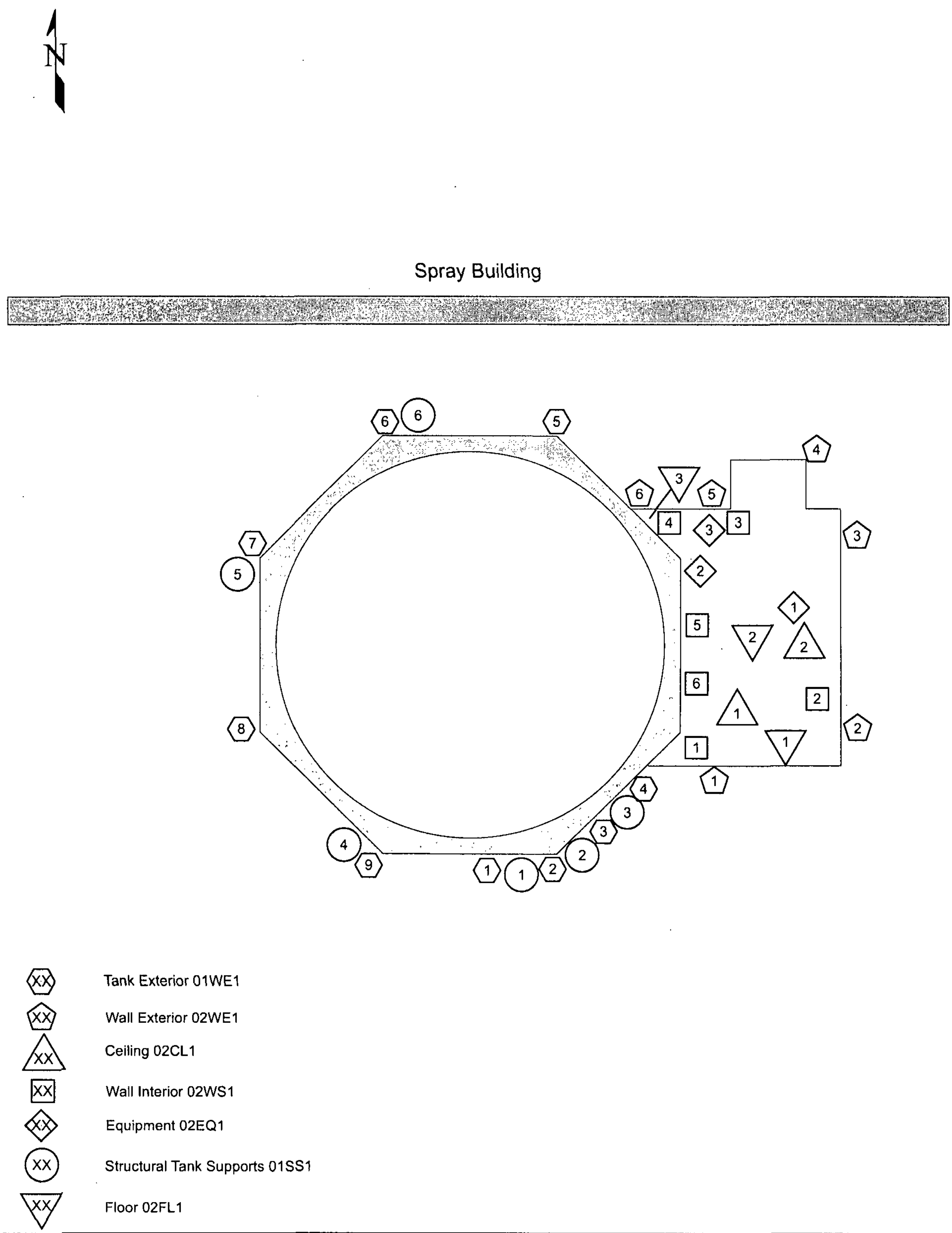
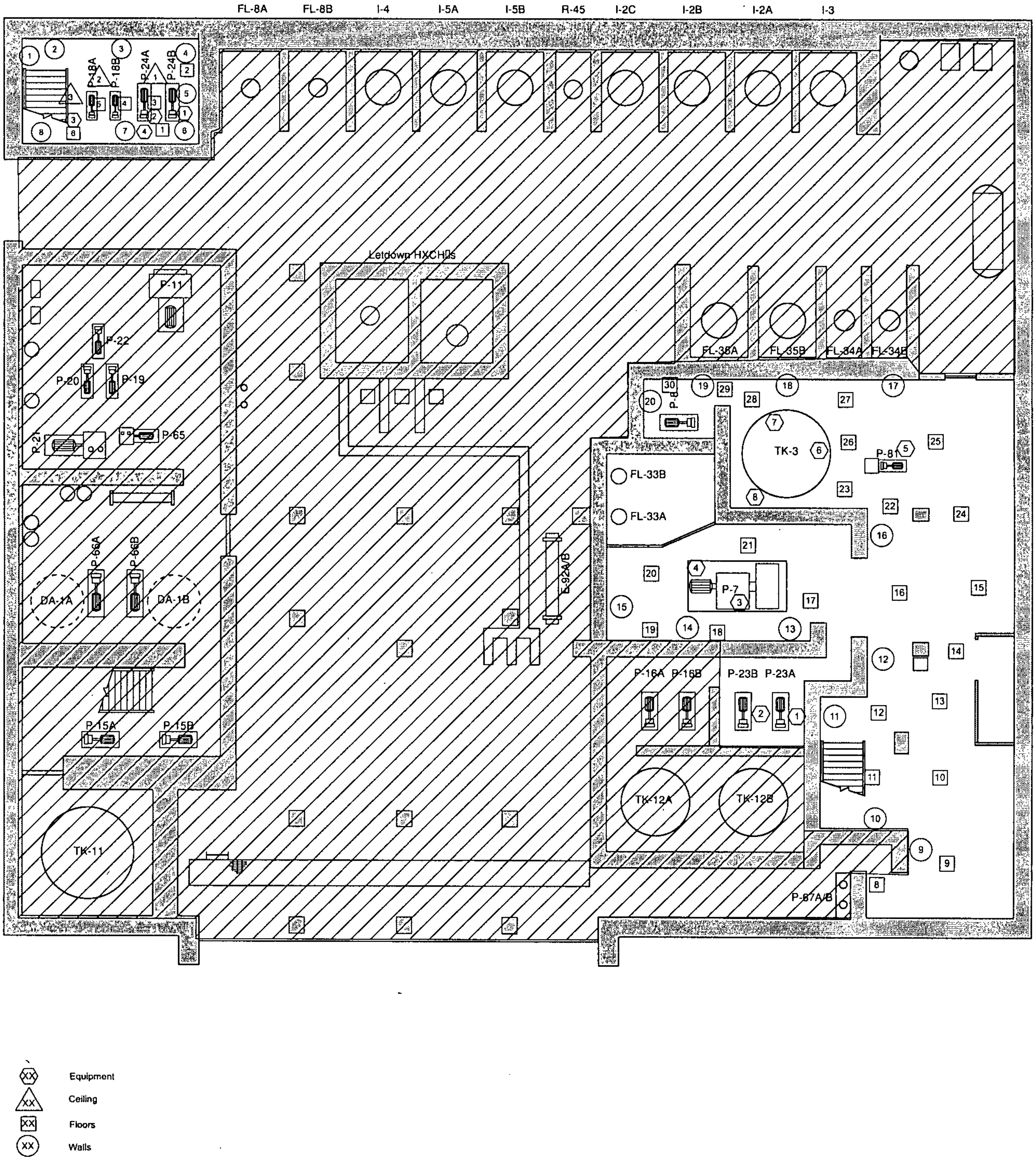
Revision 3
October 15, 2002



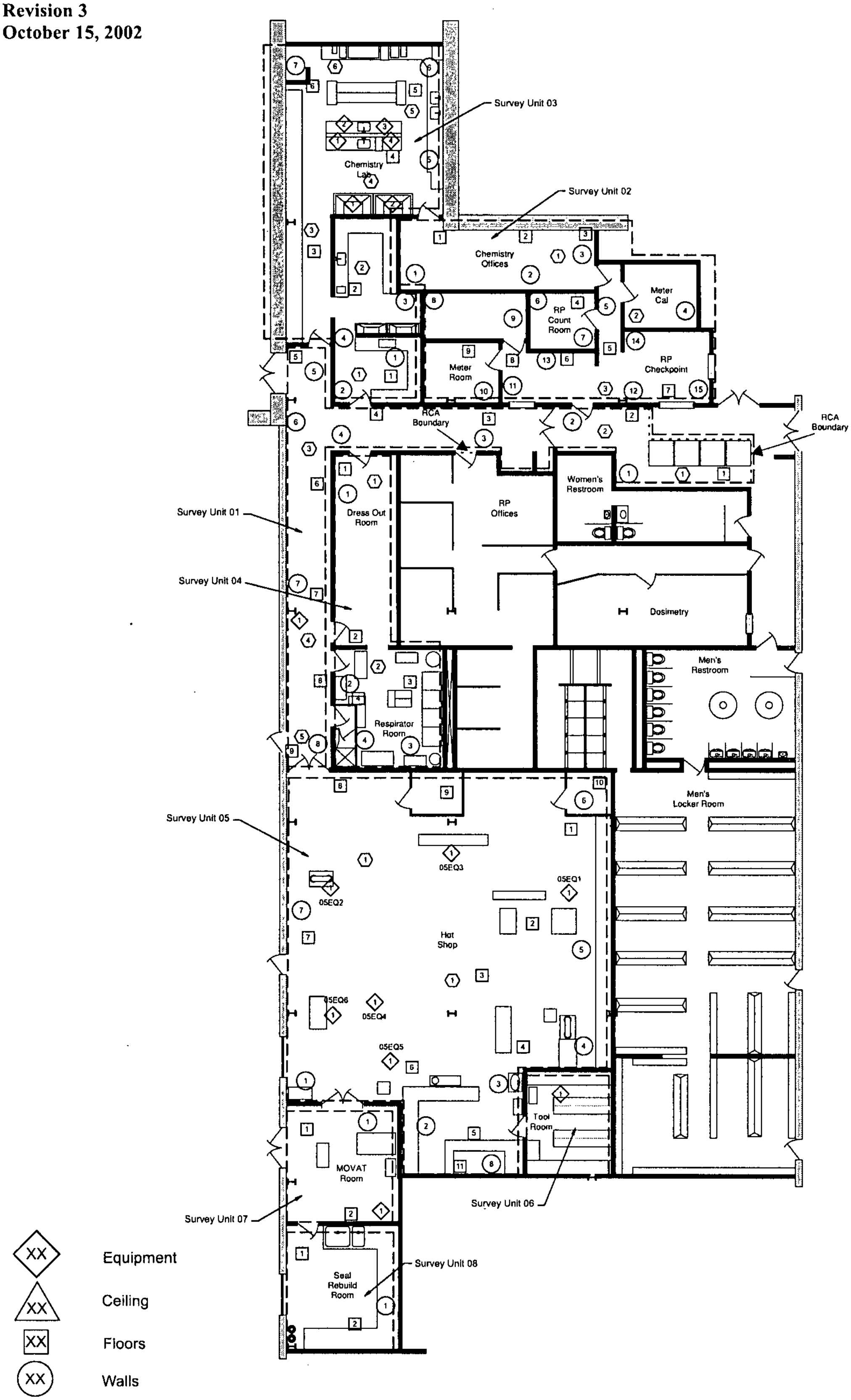
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization Demineralized Water Storage Tank (DWST) Survey Package A0500

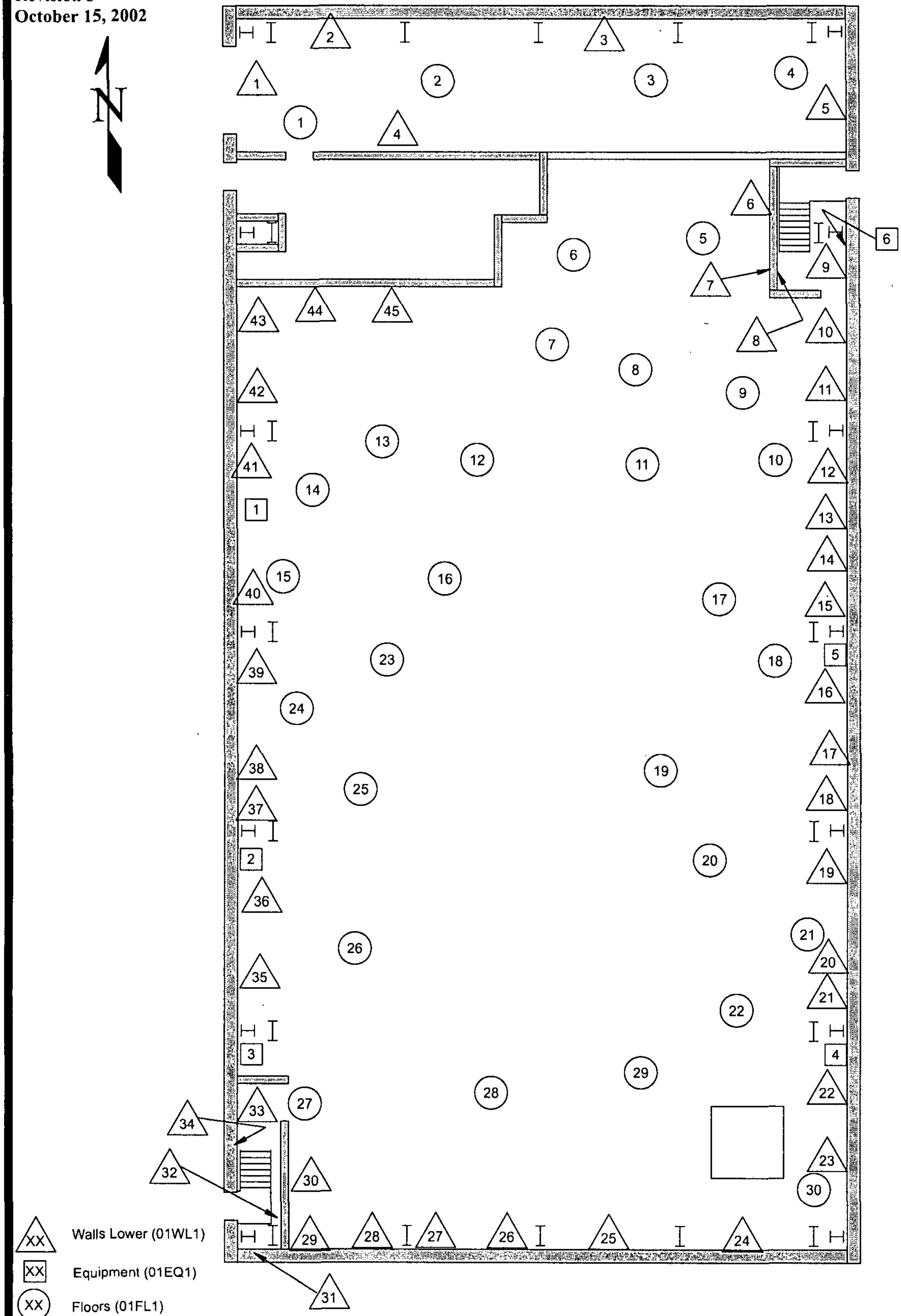
Figure 2-48



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Primary Auxiliary Building 11 ft Survey Package A0600	Figure 2-49



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Service Building Hot Side 21 ft Survey Package A0900	Figure 2-50



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Site Characterization Low Level Waste Storage Building 21 ft Survey Package A1100

Figure 2-51

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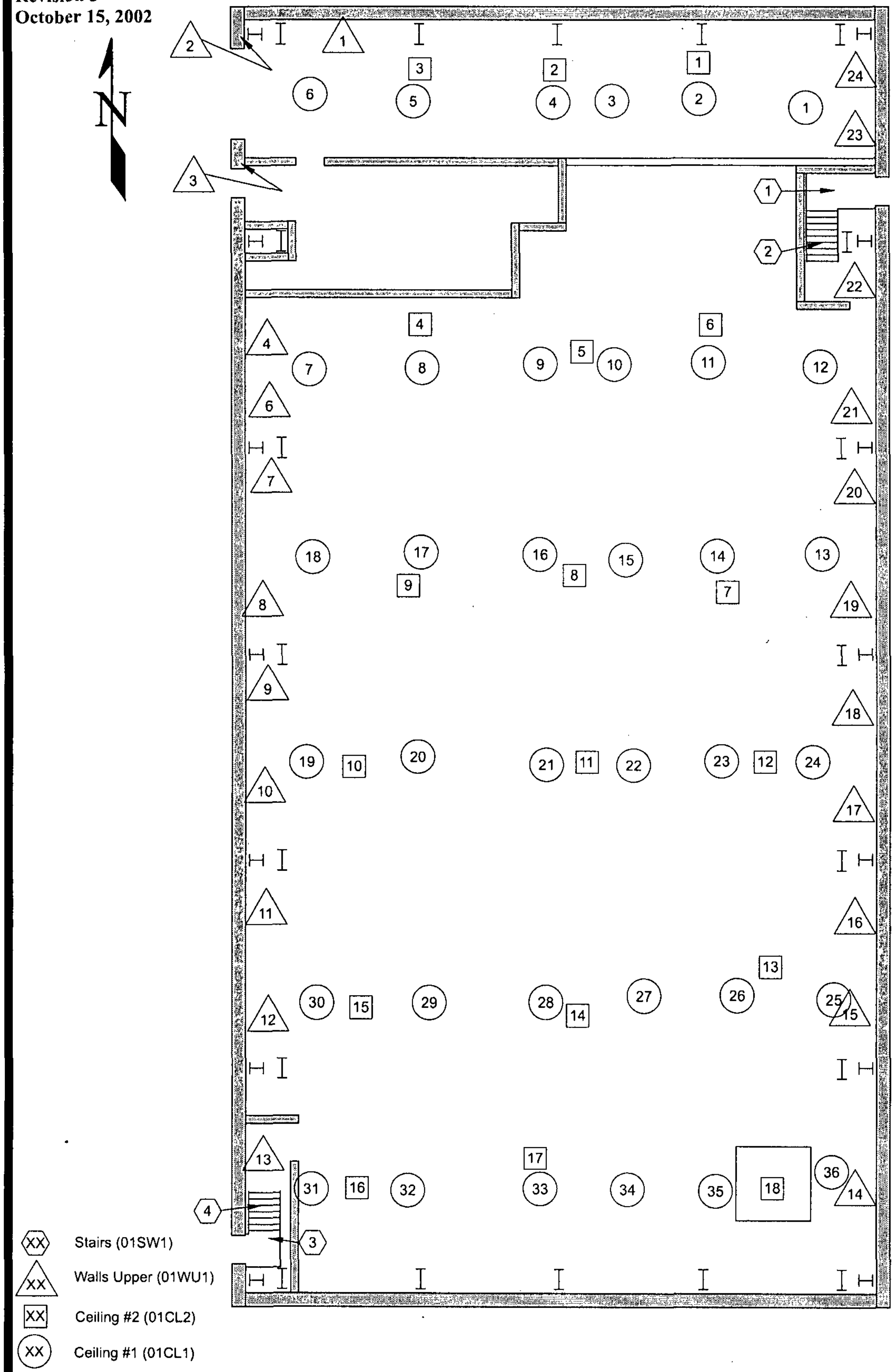
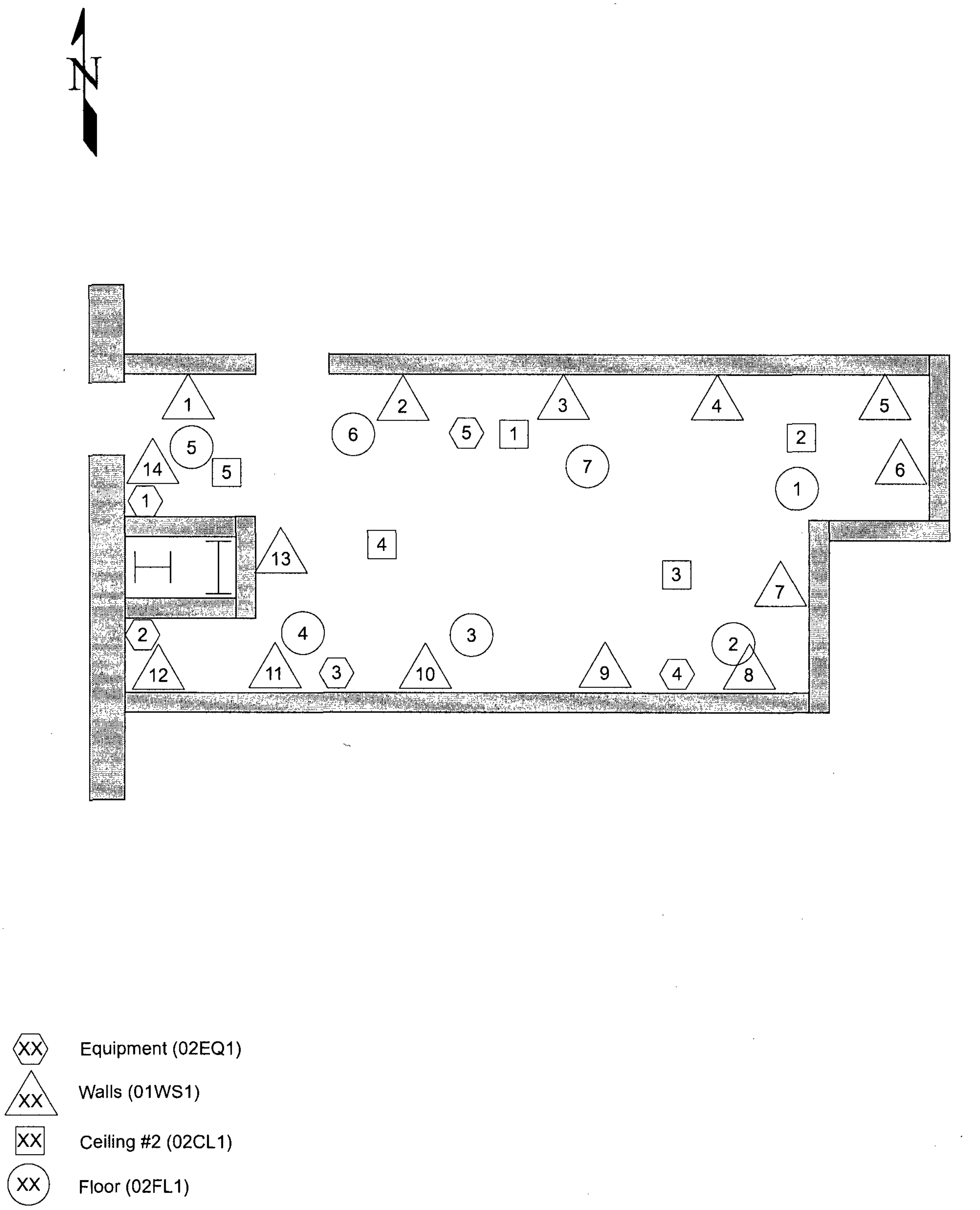
Revision 3
October 15, 2002



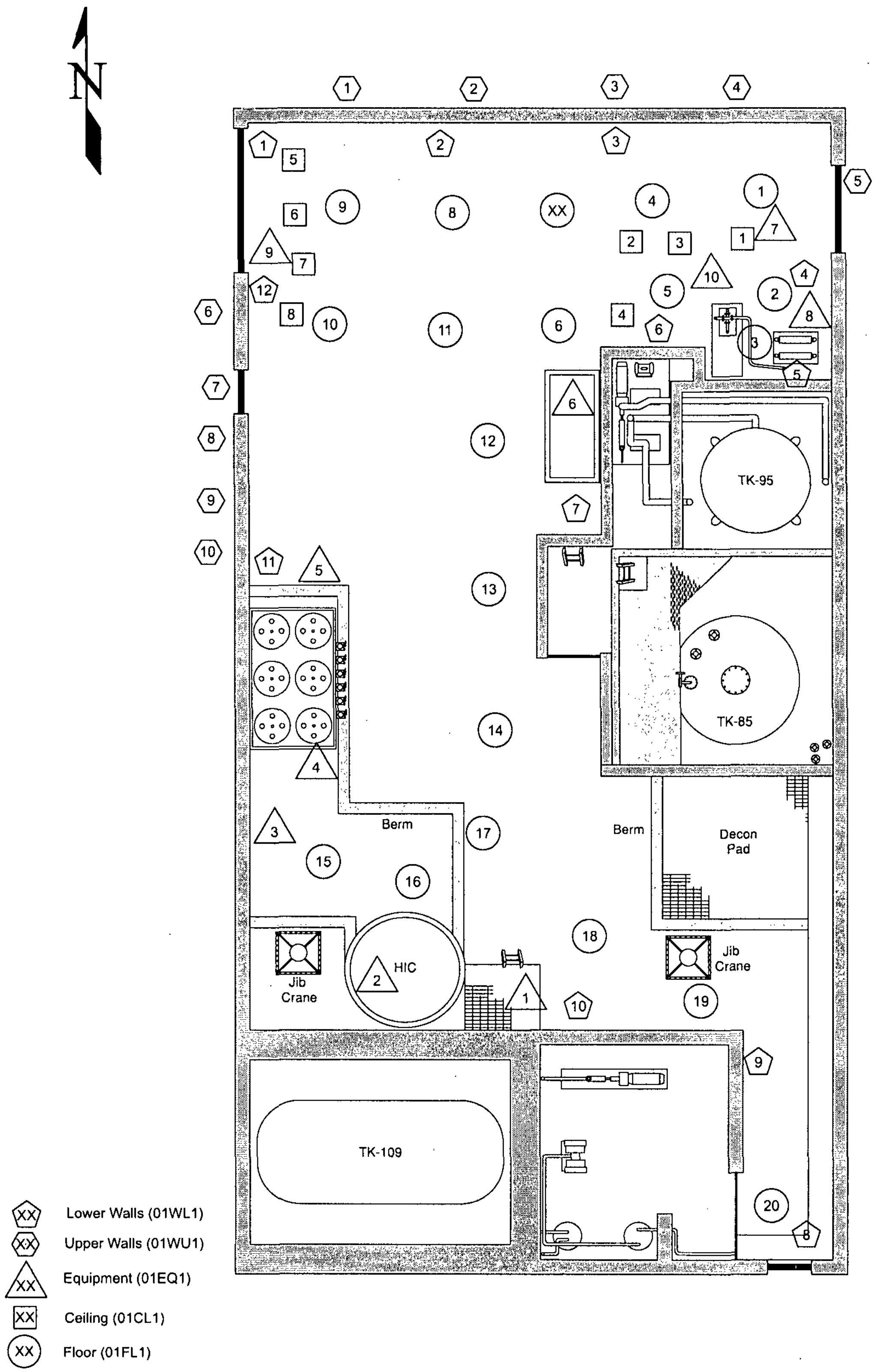
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization Low Level Waste Storage Building 21 ft Survey Package A1100

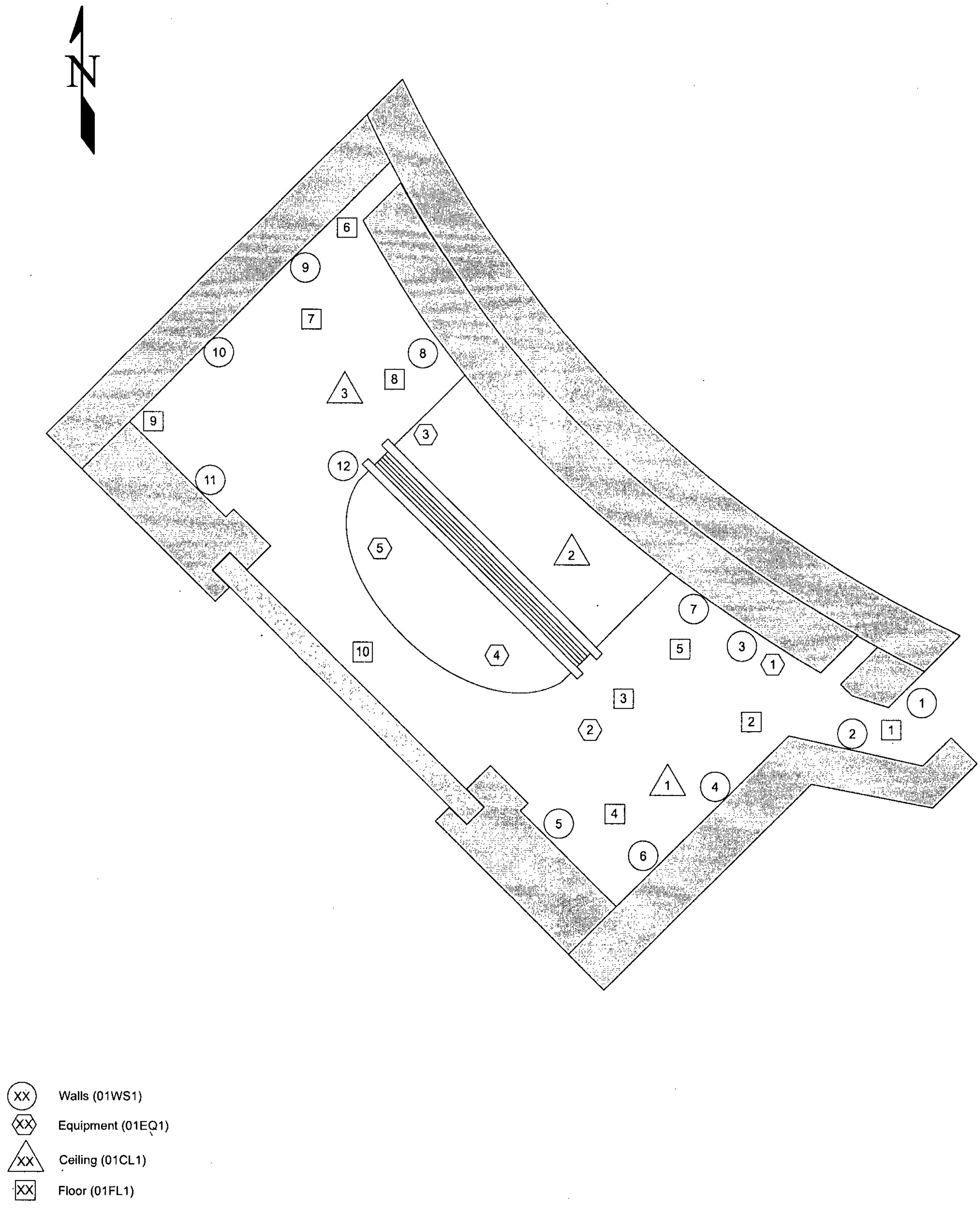
Figure 2-52



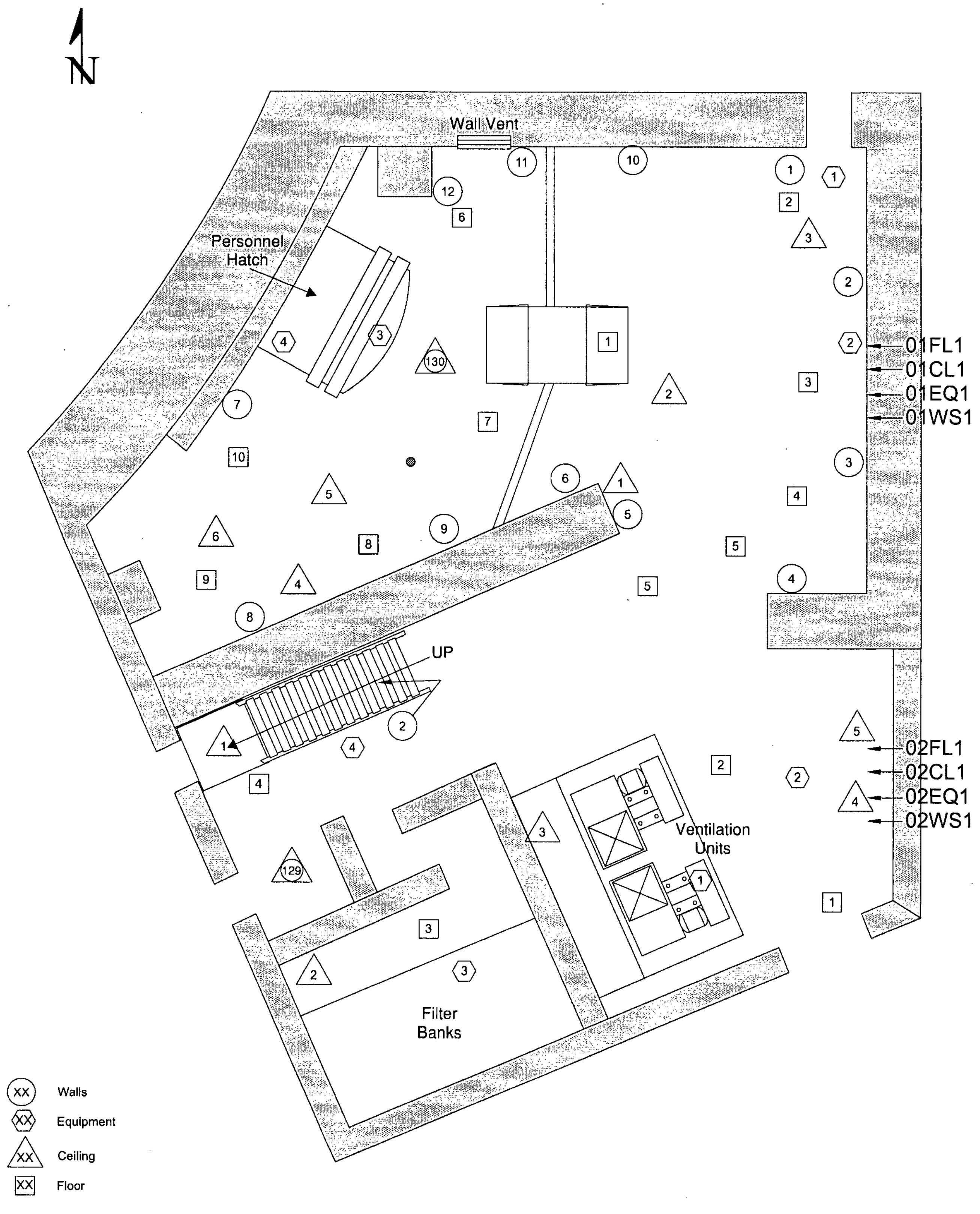
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Low Level Waste Storage Building 21 ft Office Survey Package A1100	Figure 2-53



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization RCA Storage Building Survey Package A1200	Figure 2-54



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Equipment Hatch Area 21ft Survey Package A1300	Figure 2-55



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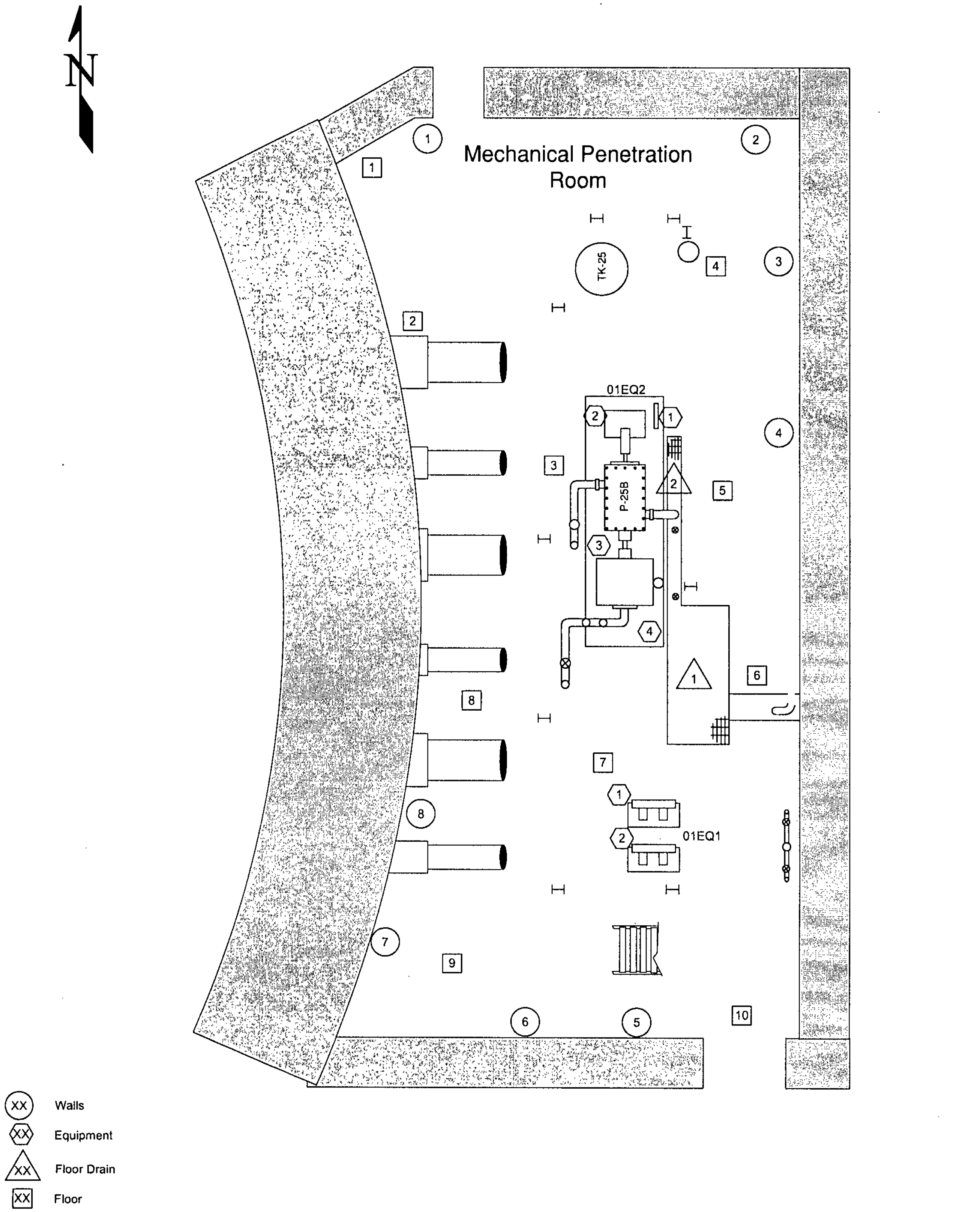
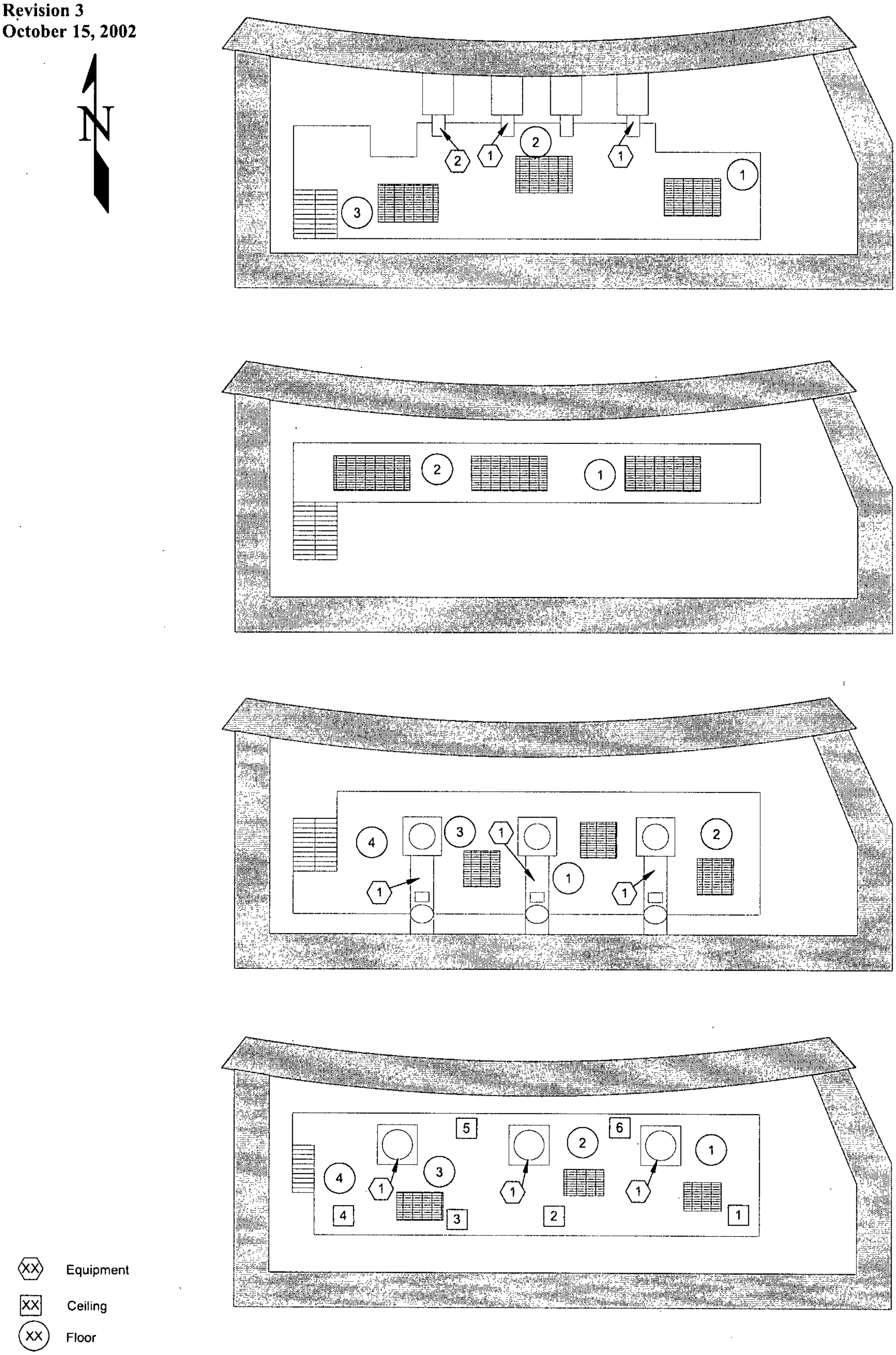
Site Characterization Personnel Hatch Area 21ft
Survey Package A1400

Figure 2-56

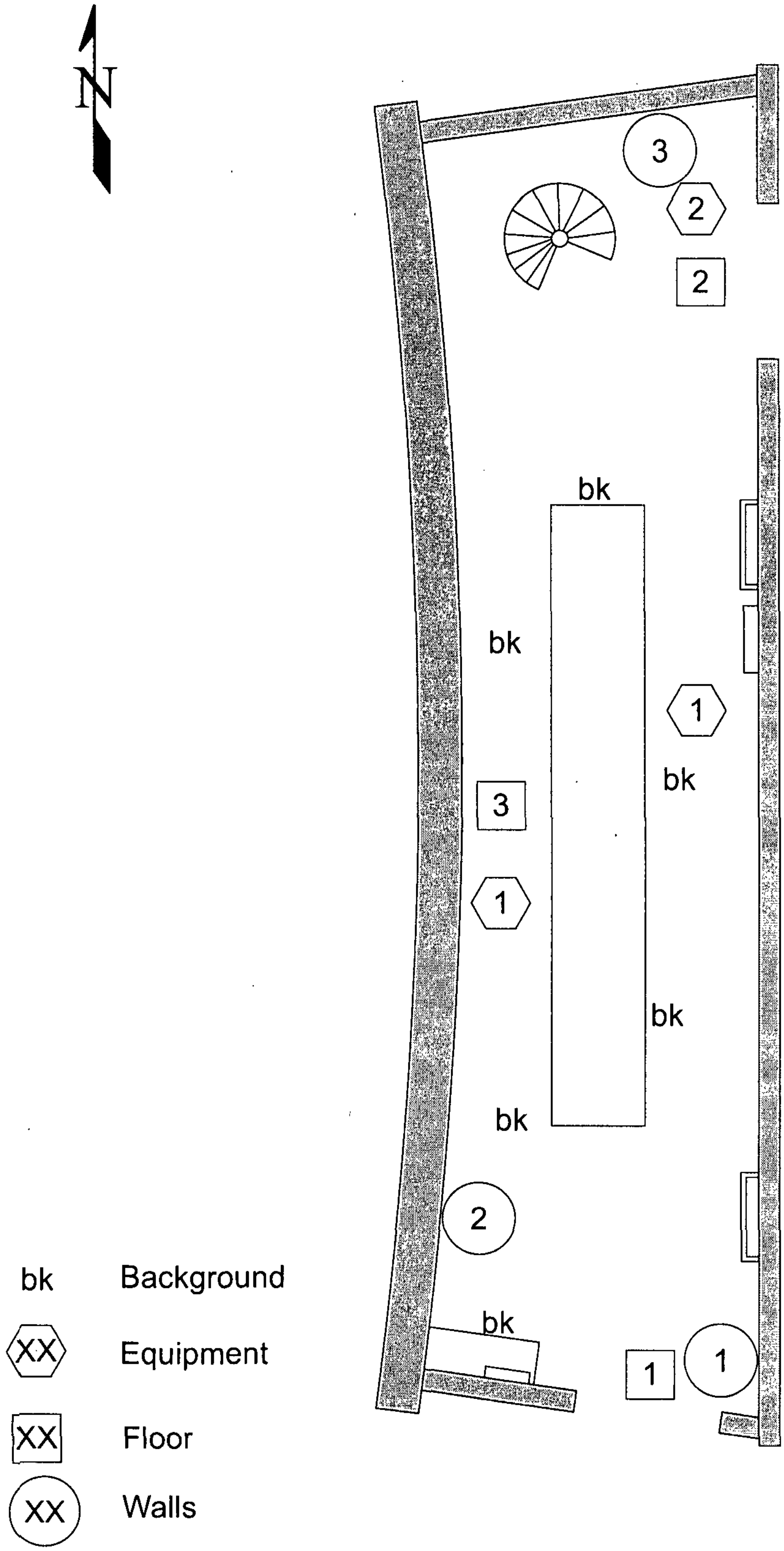


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Mechanical Penetration Room 21 ft Survey Package A1500	Figure 2-57



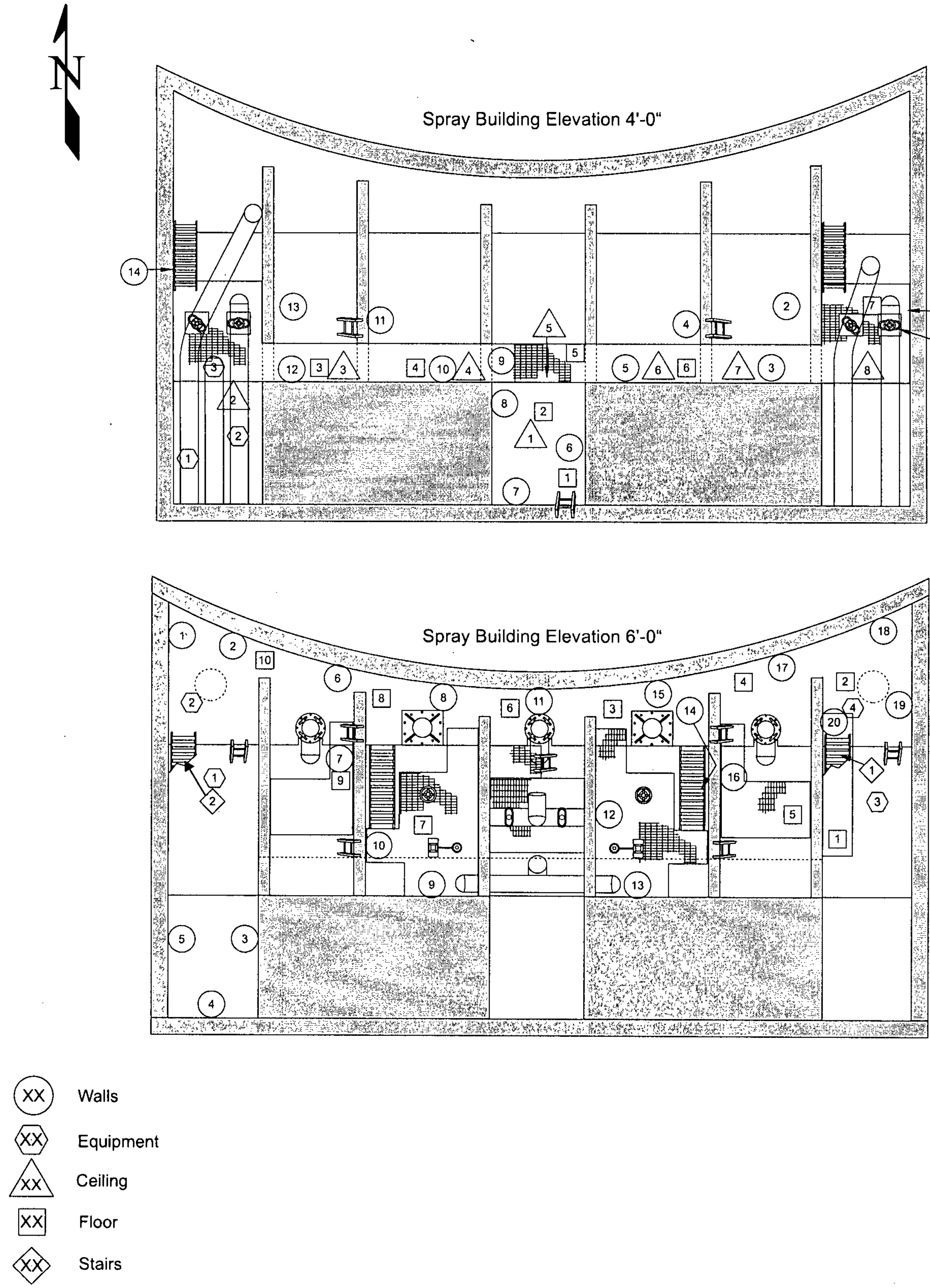
Site Characterization Mechanical Penetration Room Elevations 2, 3, 4, and 5 Survey Package A1500

Figure 2-58

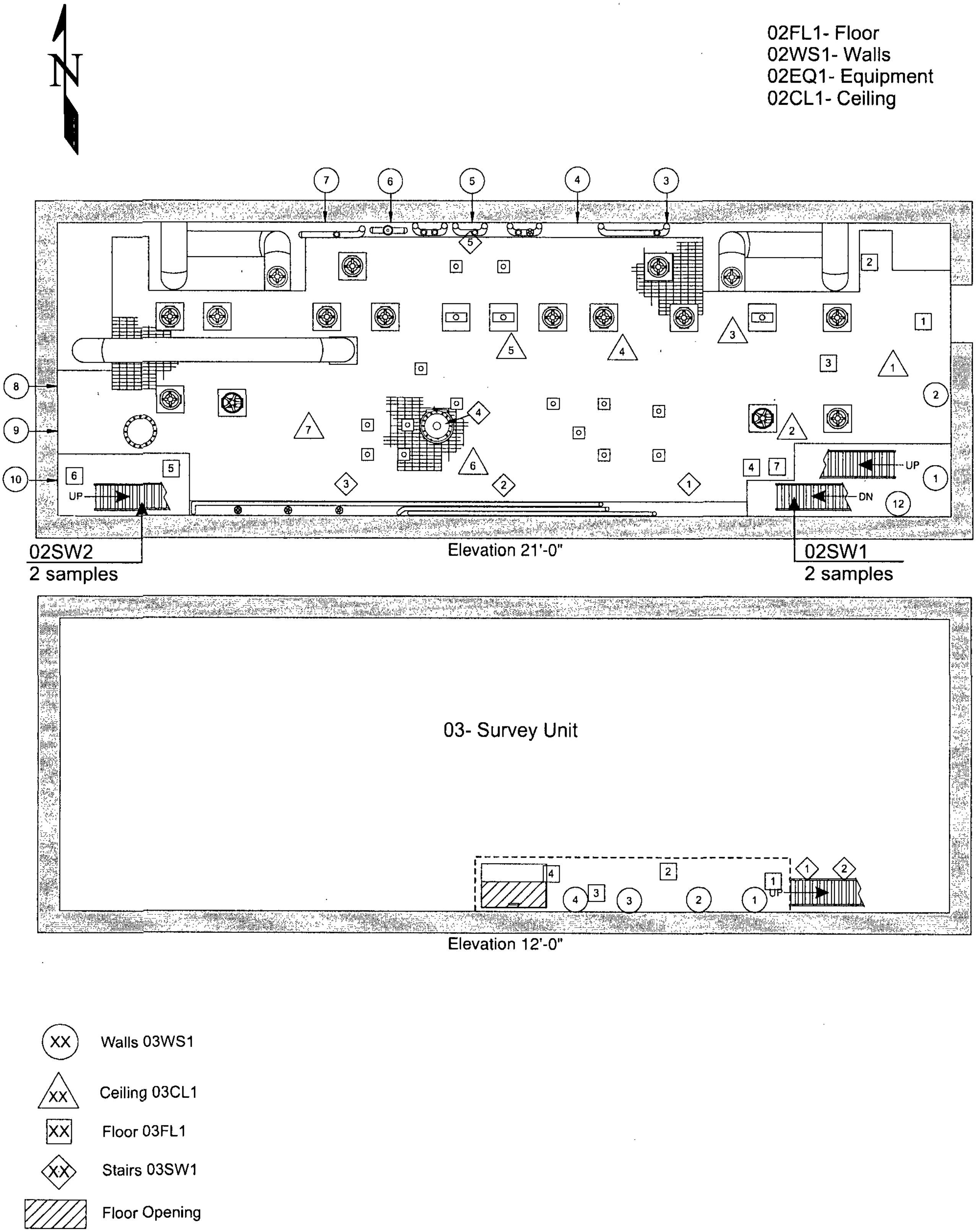


Site Characterization Electrical Penetration Room
Bottom Level Survey Package A1600

Figure 2-59

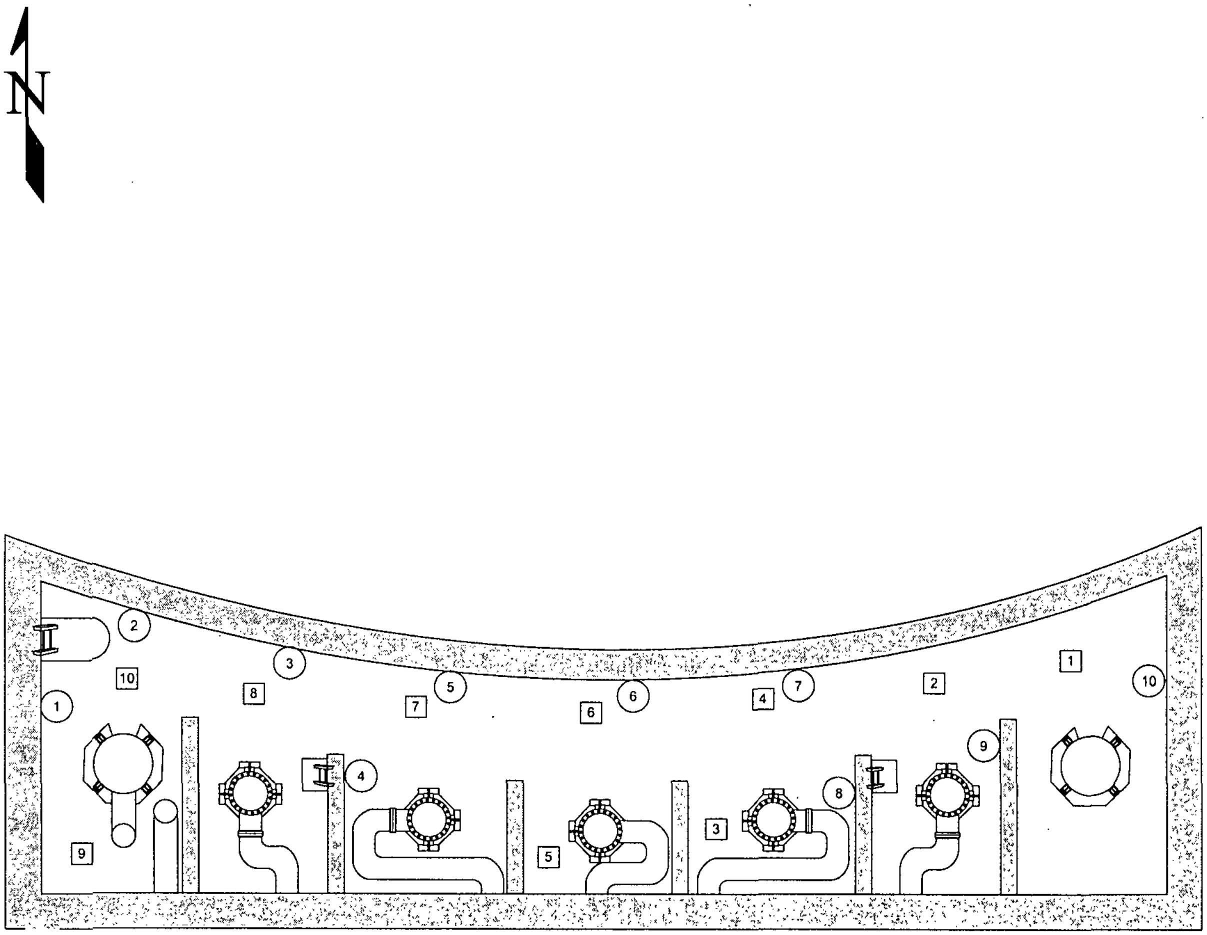


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Containment Spray Building 4 ft and 6 ft Elevations Survey Package A1700	Figure 2-60

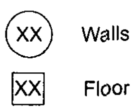


Site Characterization Containment Spray Building 12 ft and 21 ft Elevations Survey Package A1700

Figure 2-61



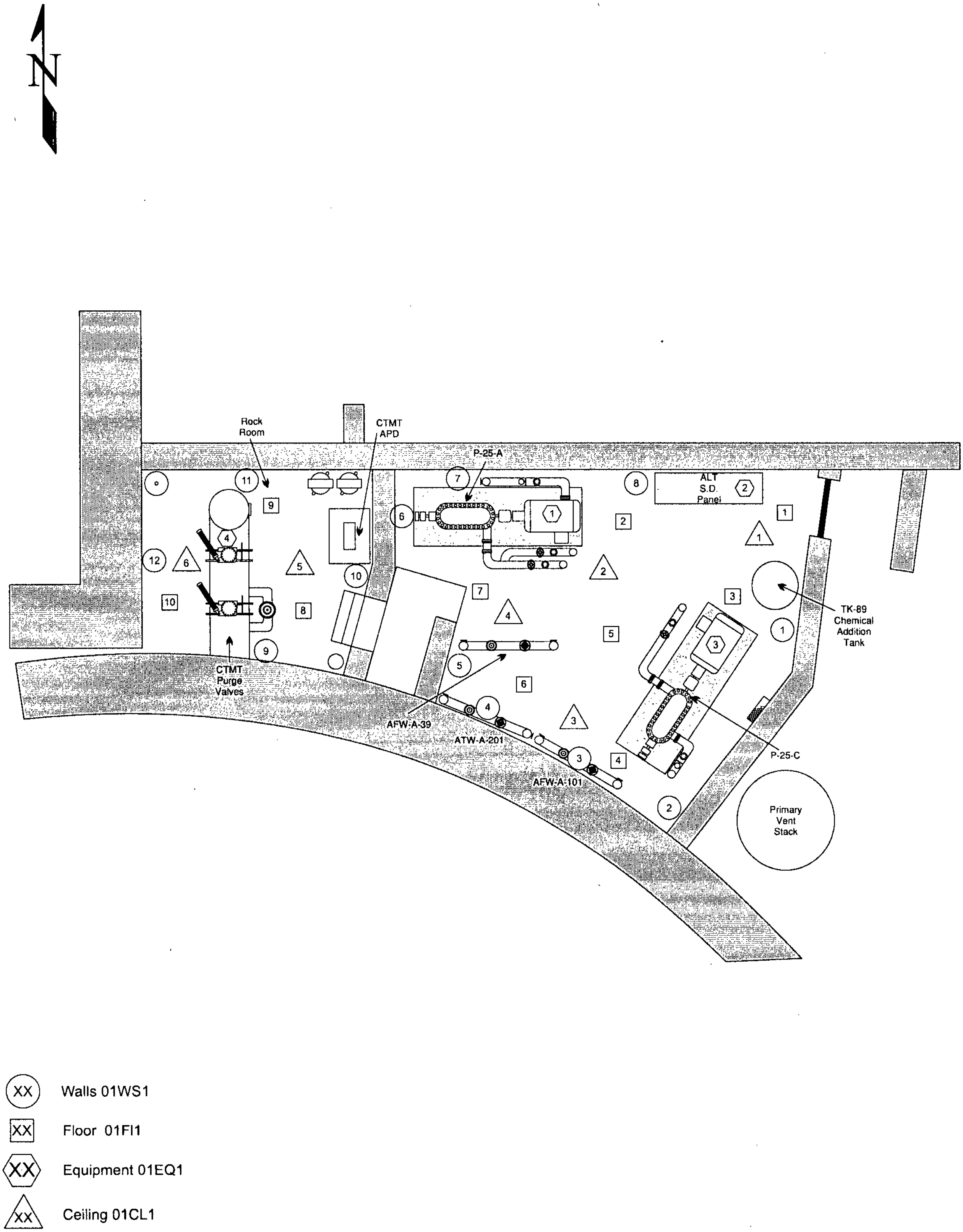
Spray Building 14'-0" (Motors)



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

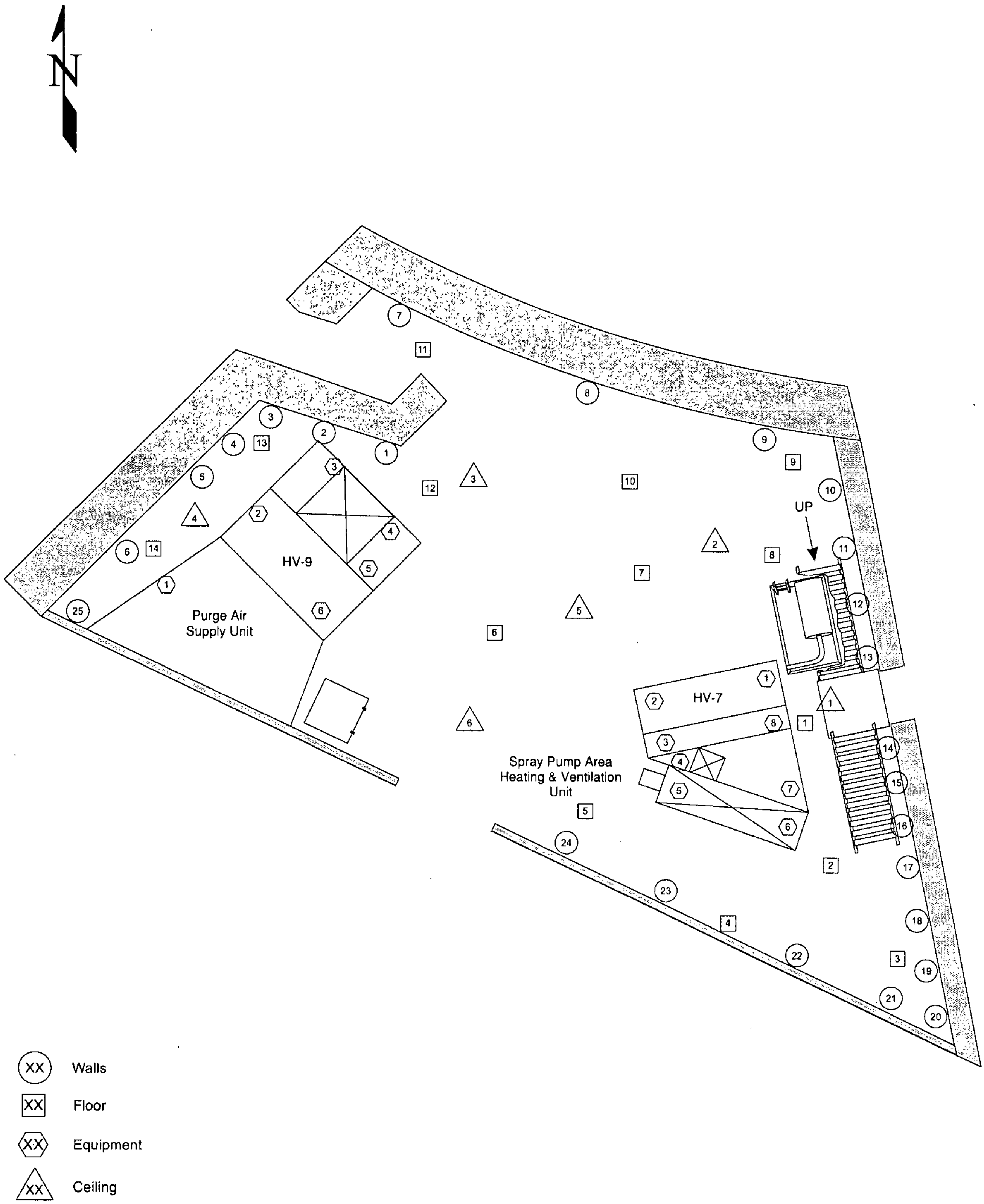
Site Characterization Containment Spray Building 14 ft Elevation Survey Package A1700

Figure 2-62

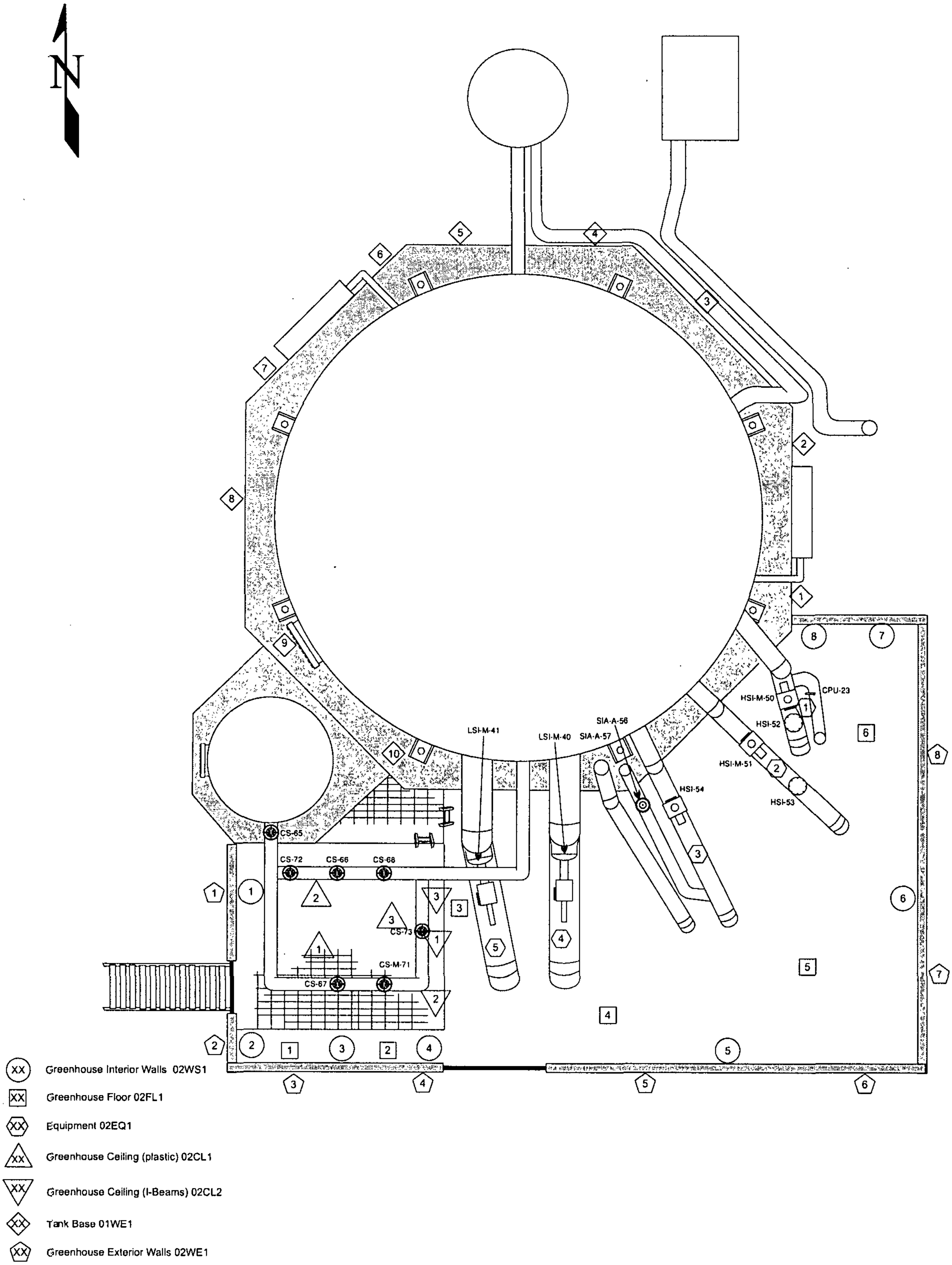


Site Characterization Auxiliary Feed Pump Room 21 ft Survey Package A1800

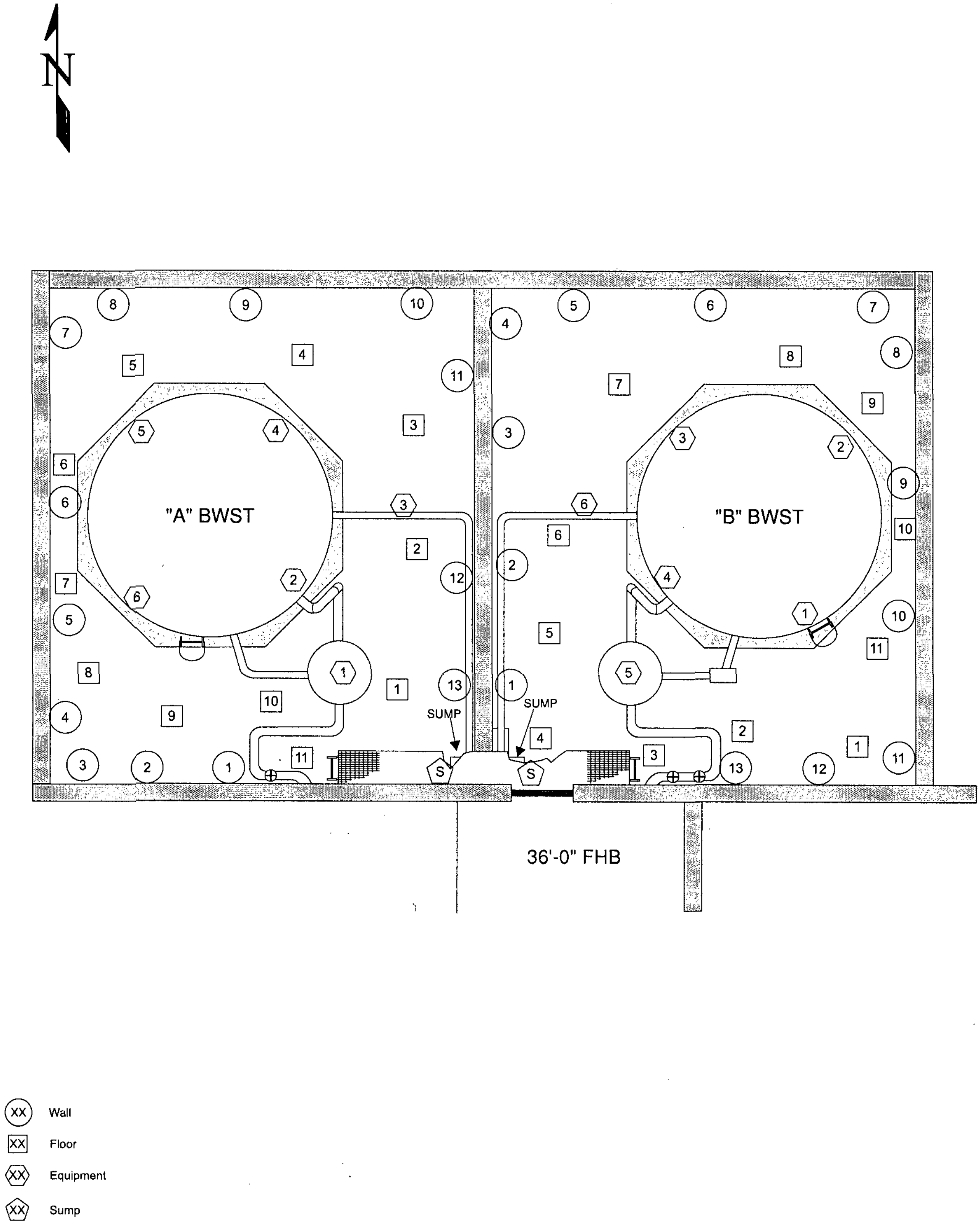
Figure 2-63



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Equipment Access Area 21 ft Survey Package A1900	Figure 2-64



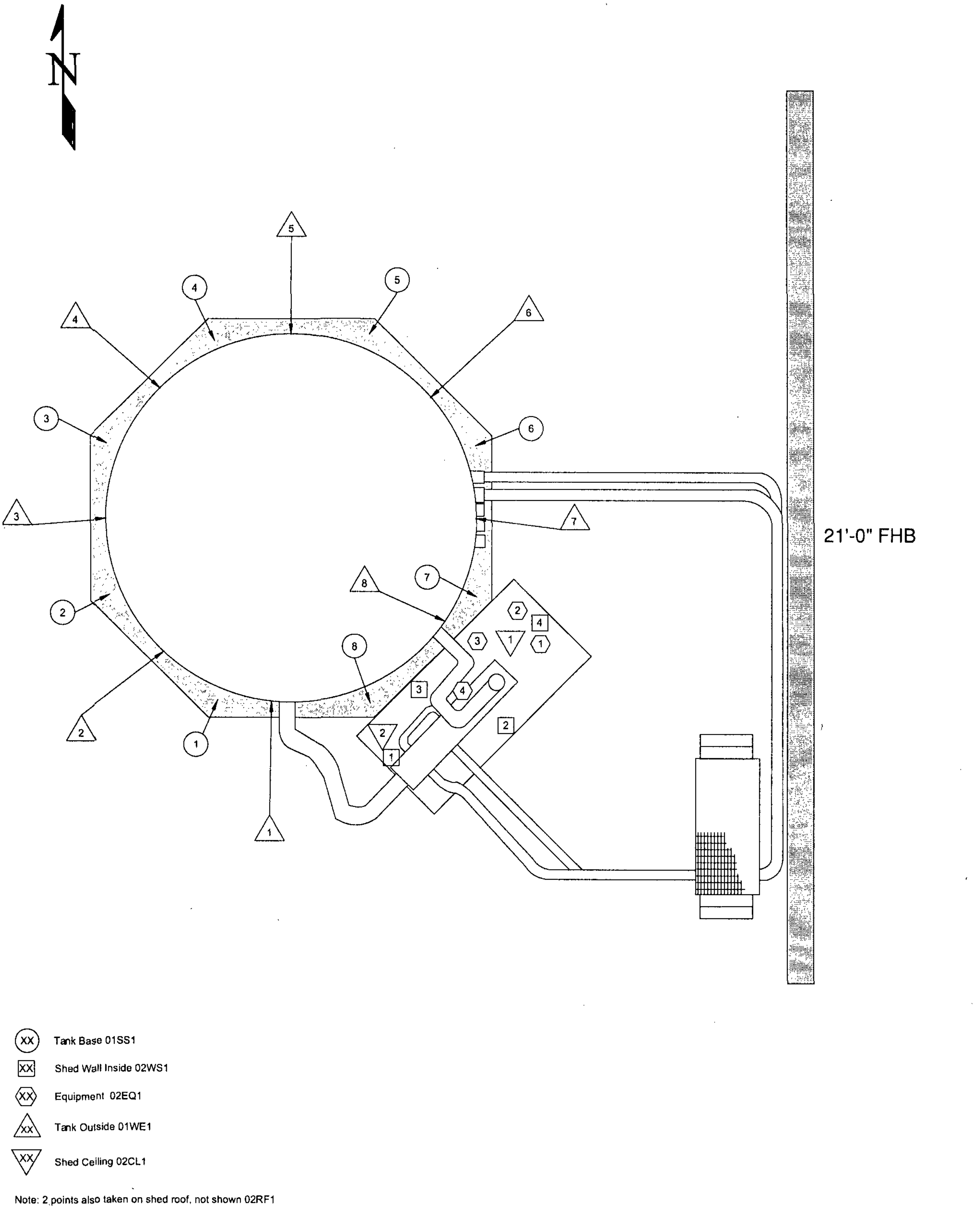
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Reactor Water Storage Tank Survey Package A2100	Figure 2-65



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

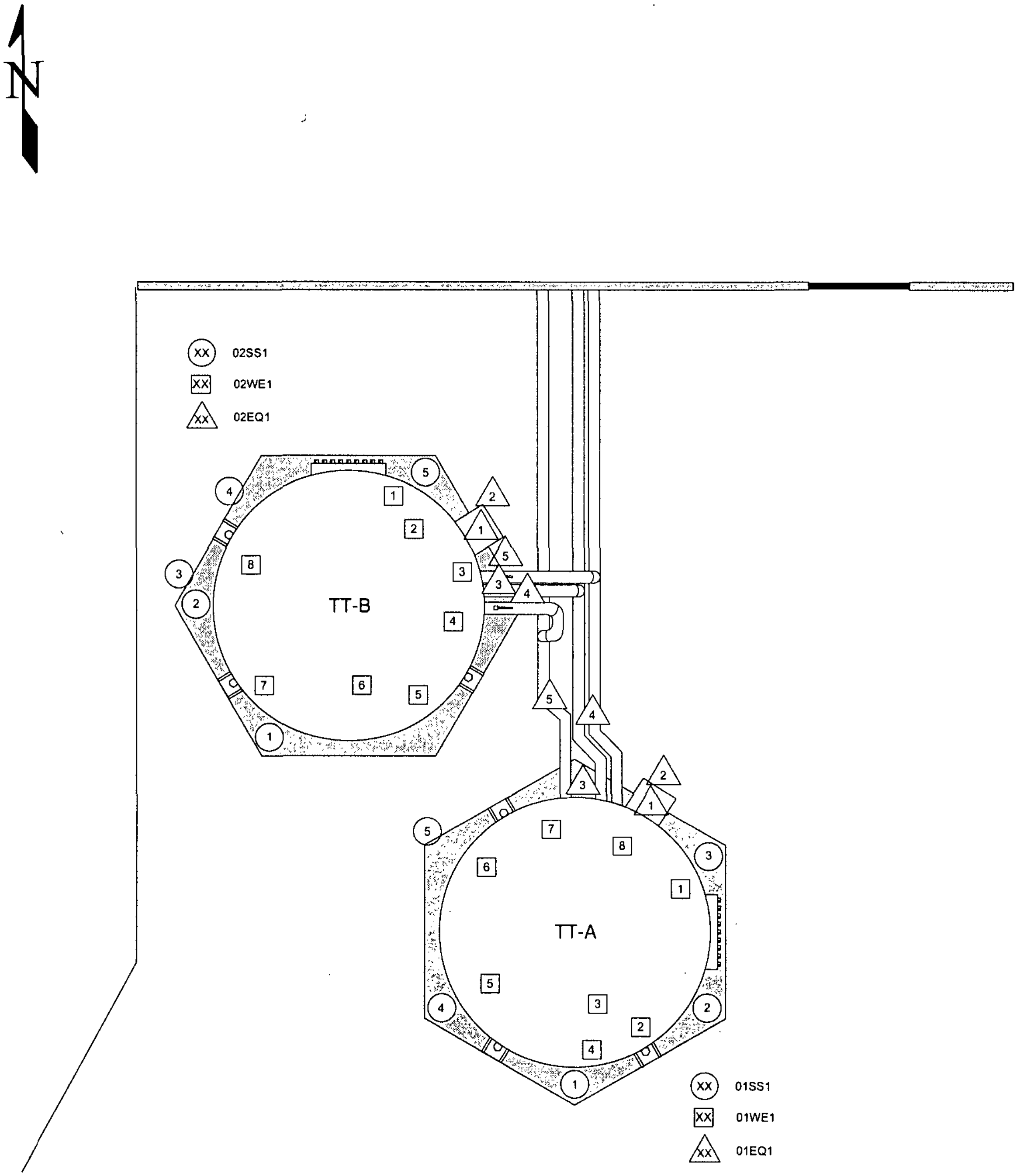
Site Characterization Borated Water Storage Tanks Survey Package A2200

Figure 2-66



Site Characterization Primary Storage Tank Area
Survey Package A2300

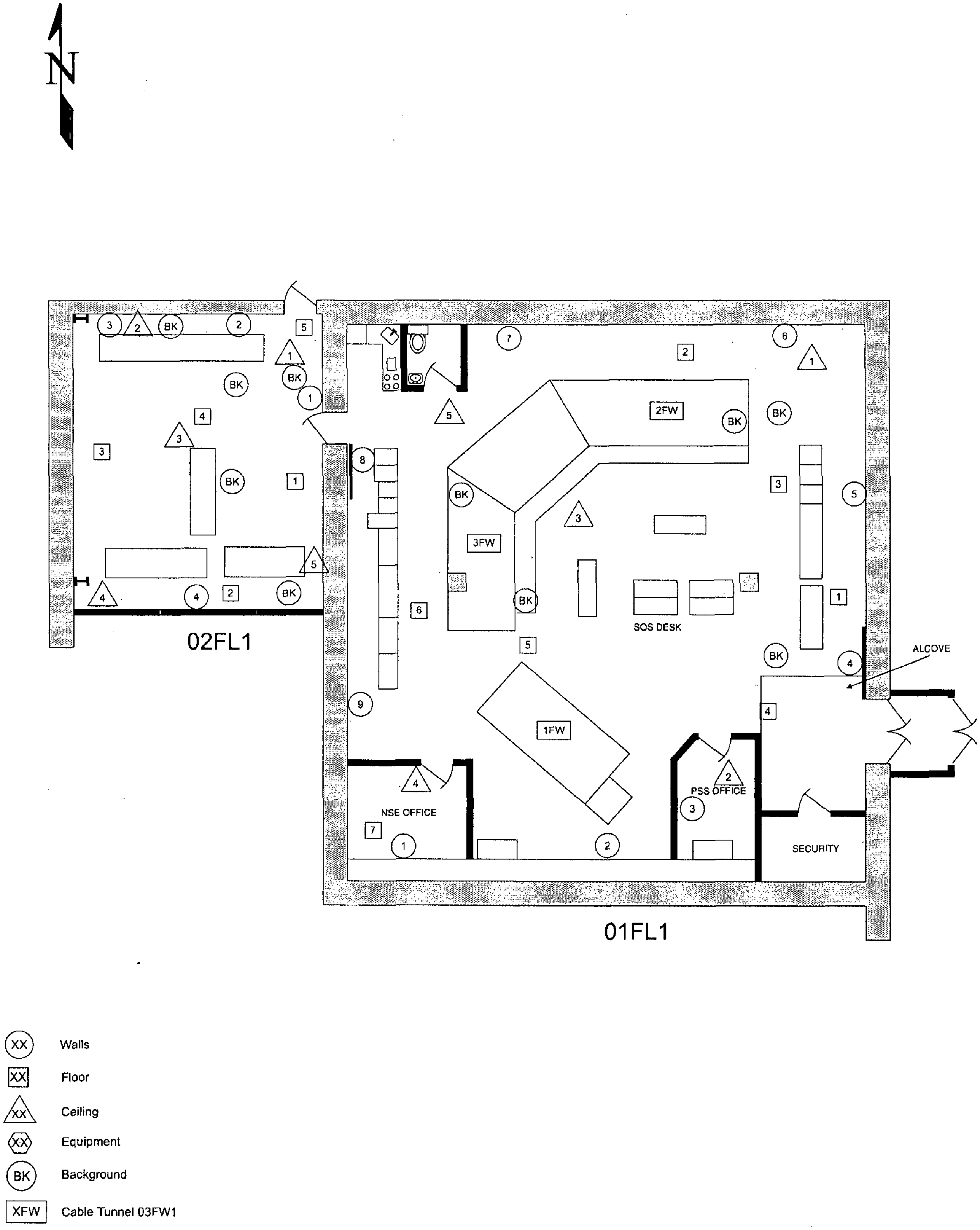
Figure 2-67



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization Test Tanks 21 ft
Survey Package A2400

Figure 2-68



Site Characterization Control Room & Computer Room Survey Package B0200

Figure 2-69

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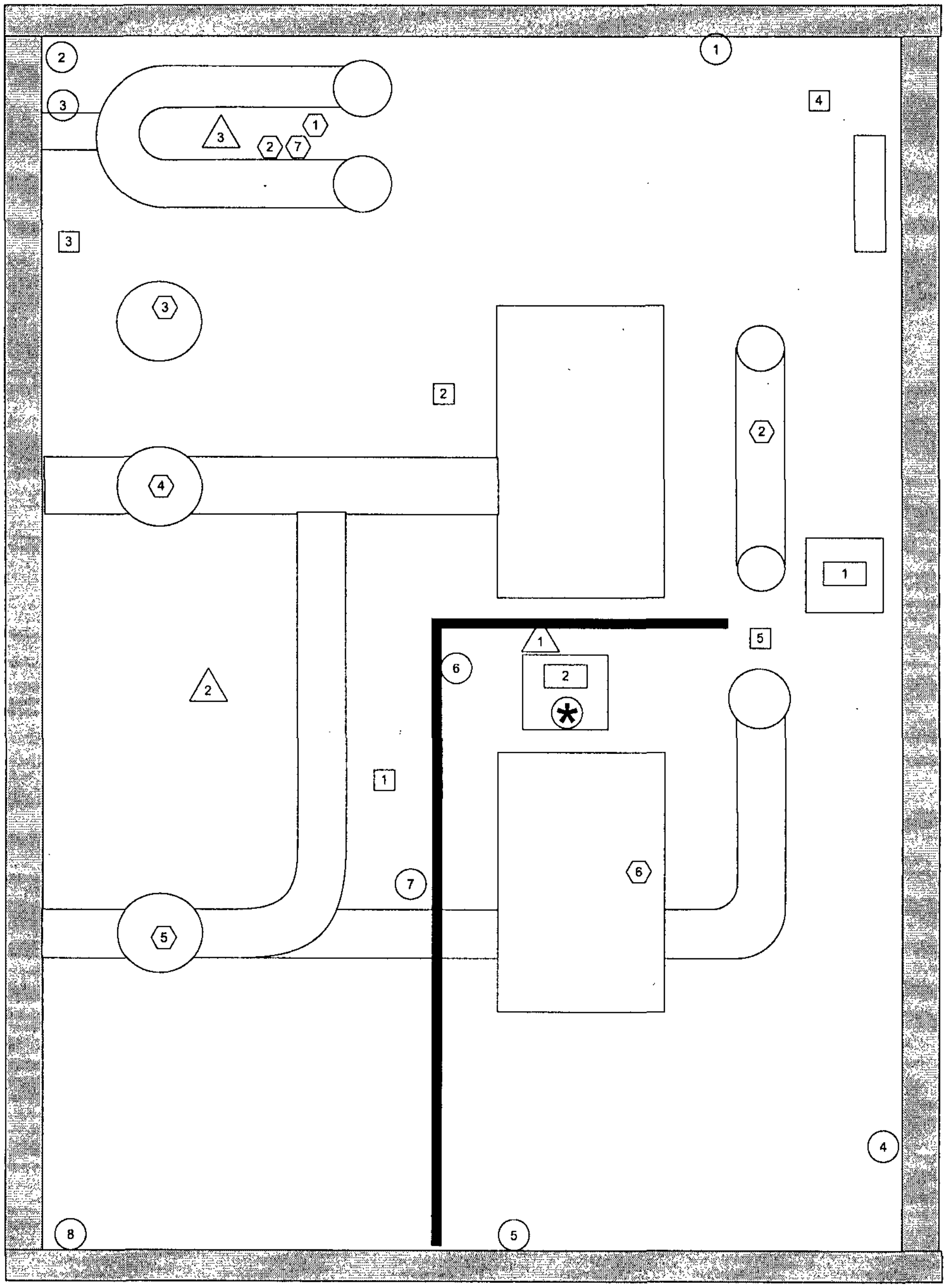
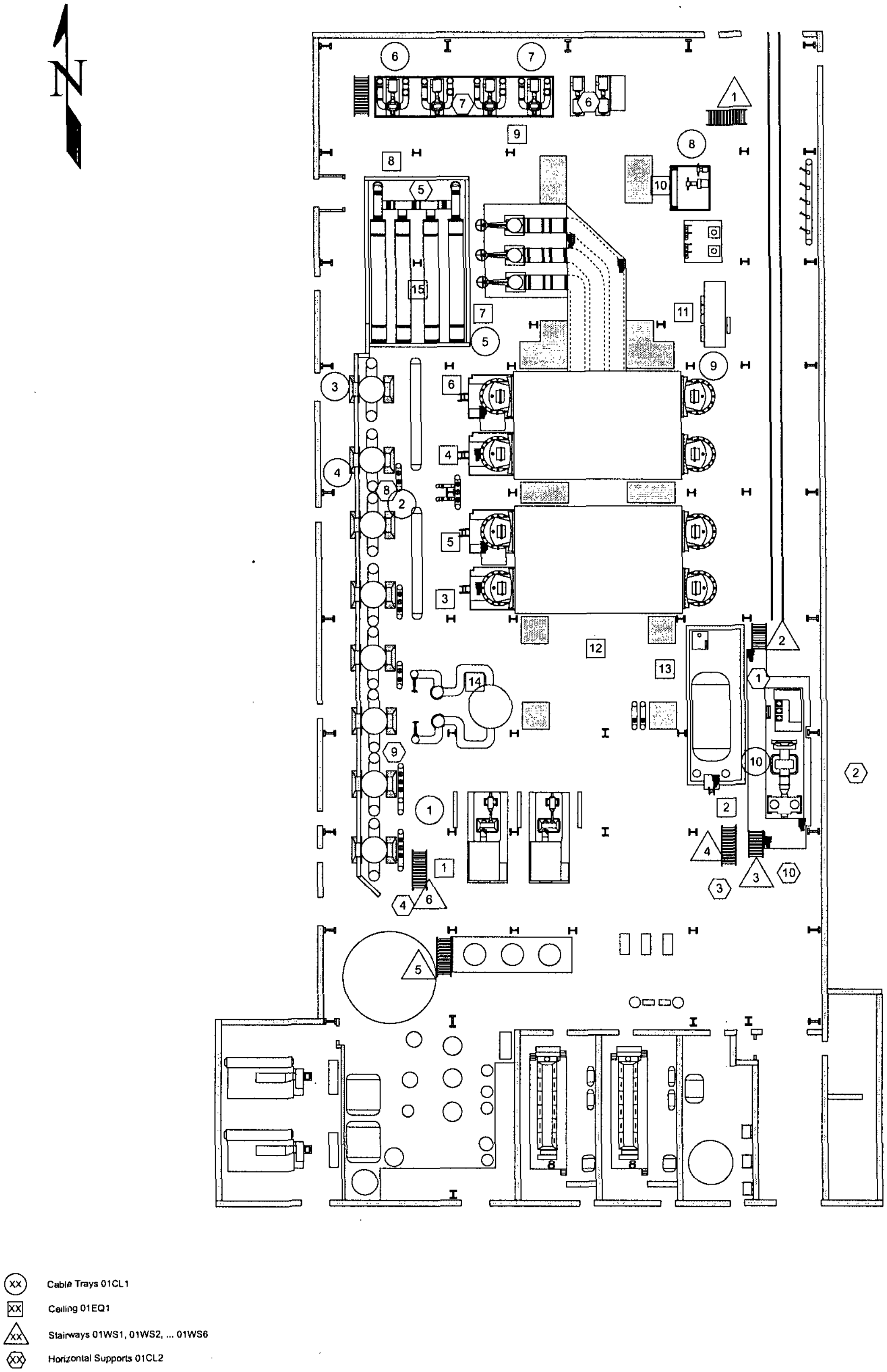


Walls 01WS1
Floor 01FL1
Ceiling 01CL1
Equipment 01EQ1
Sediment Sample
Floor Drain 01FD1

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Fire Pump House Survey Package B0400	Figure 2-70



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Turbine Building 21 ft Survey Package B0500	Figure 2-71

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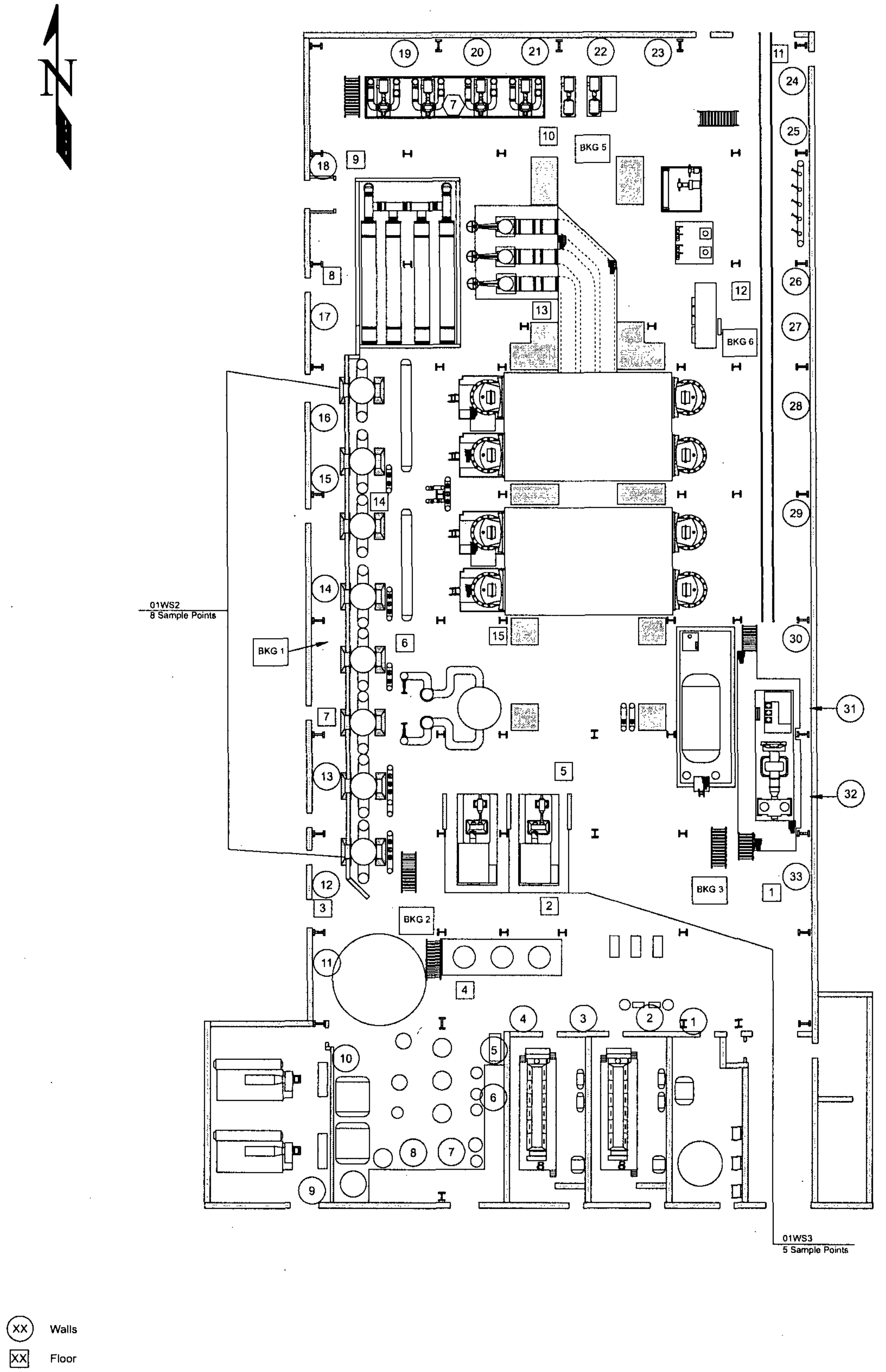
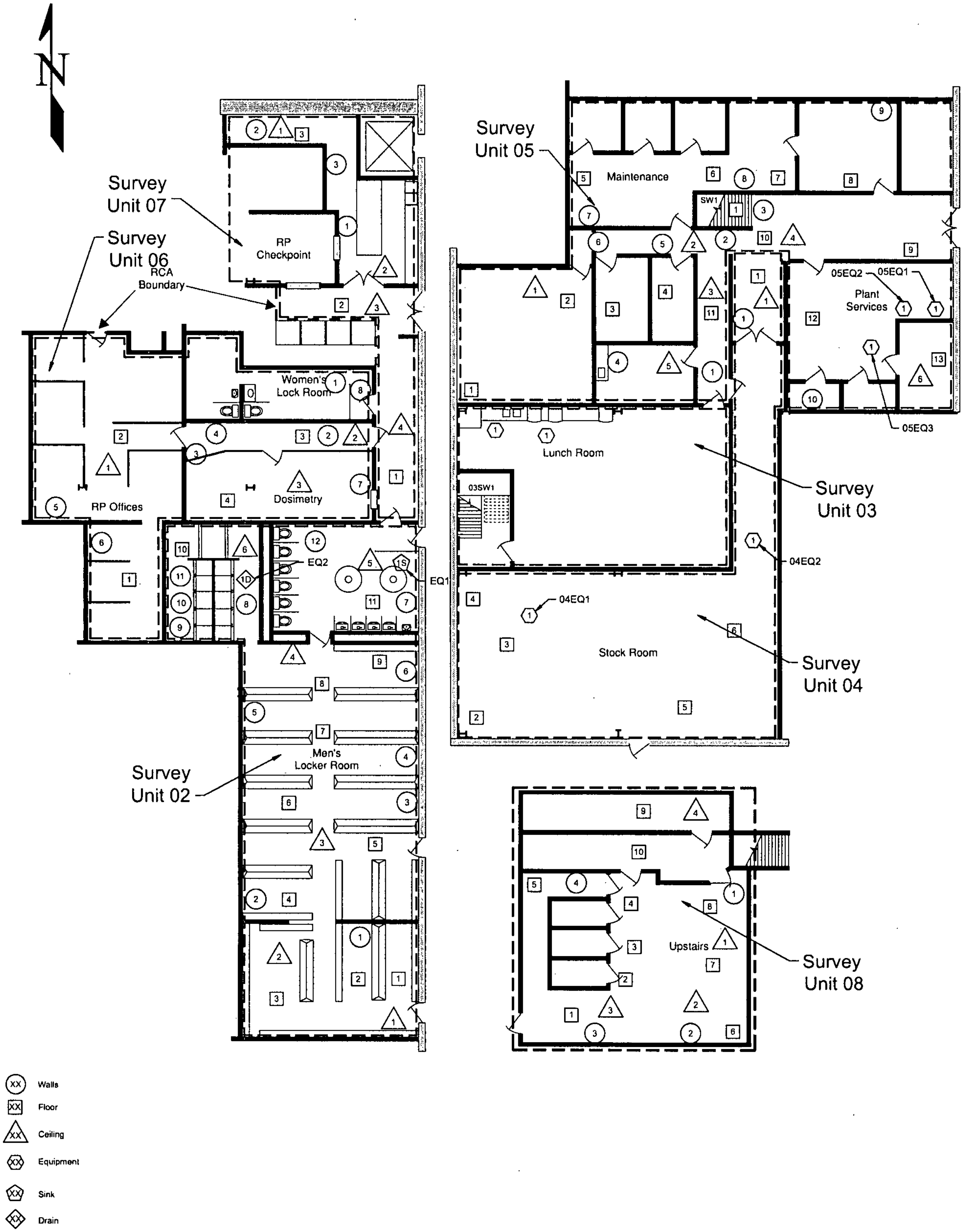
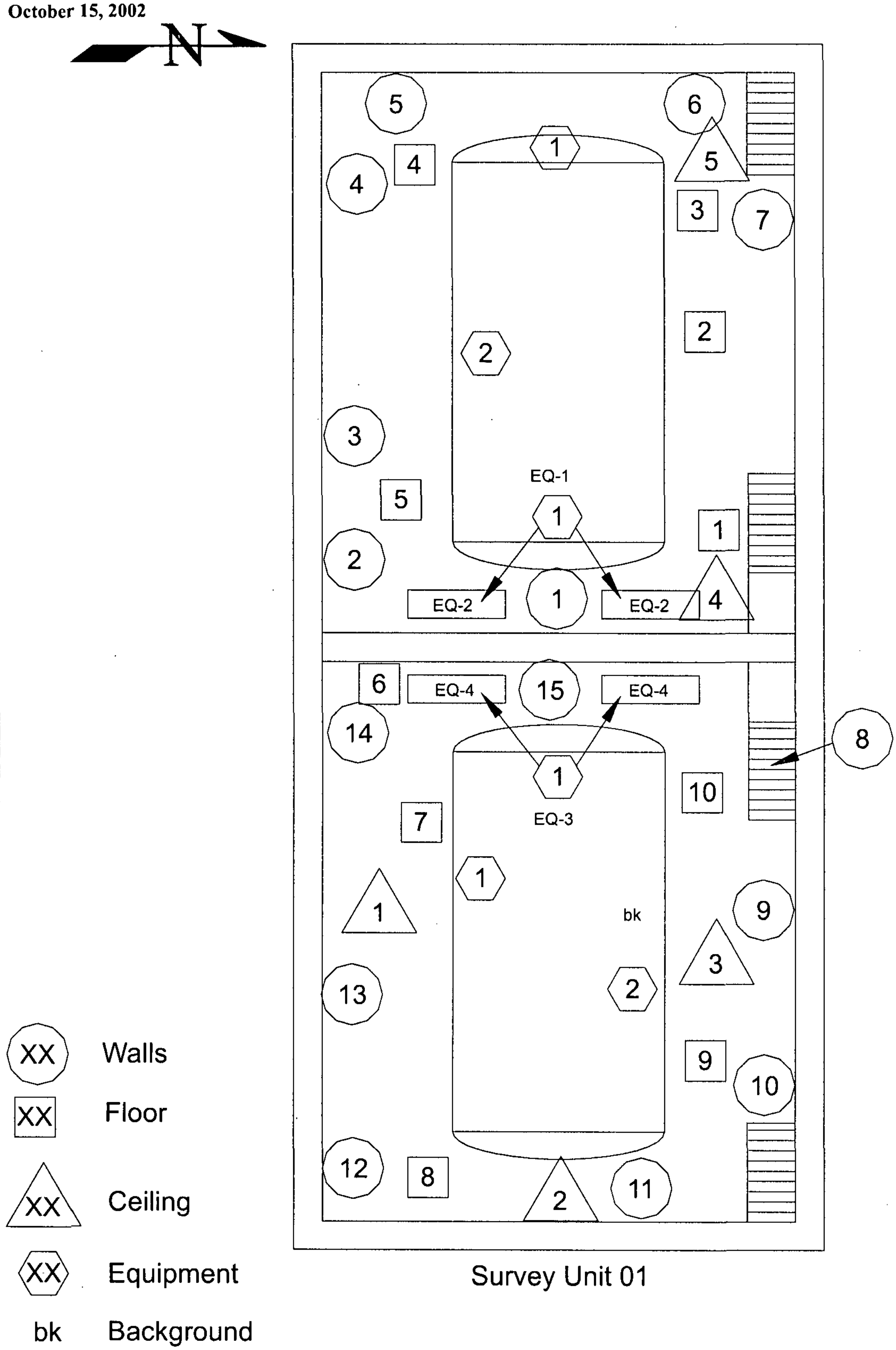


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Turbine Building 21 ft Survey Package B0500	Figure 2-72

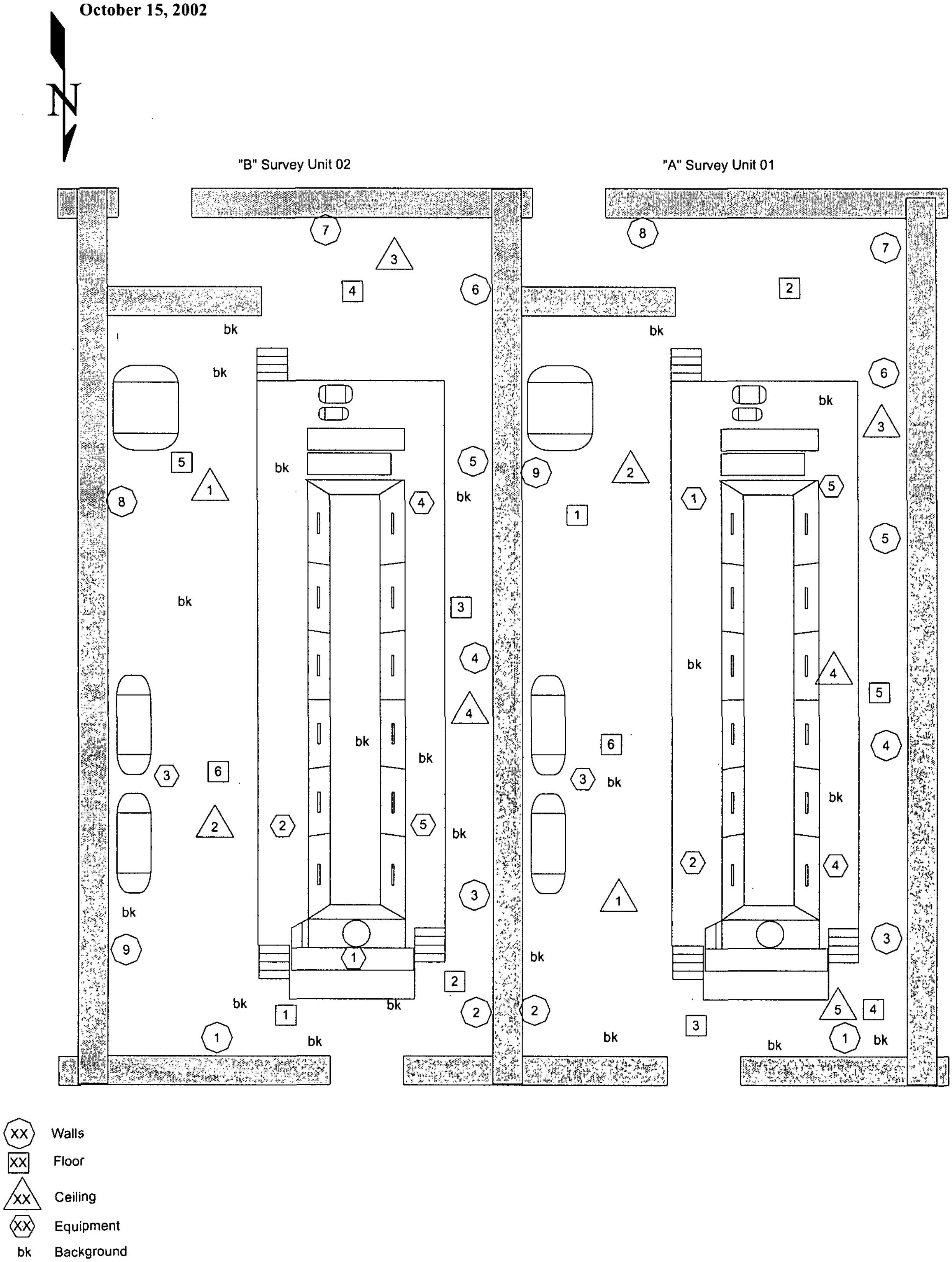


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Service building Cold Side 21 ft Survey Package B0700	Figure 2-73



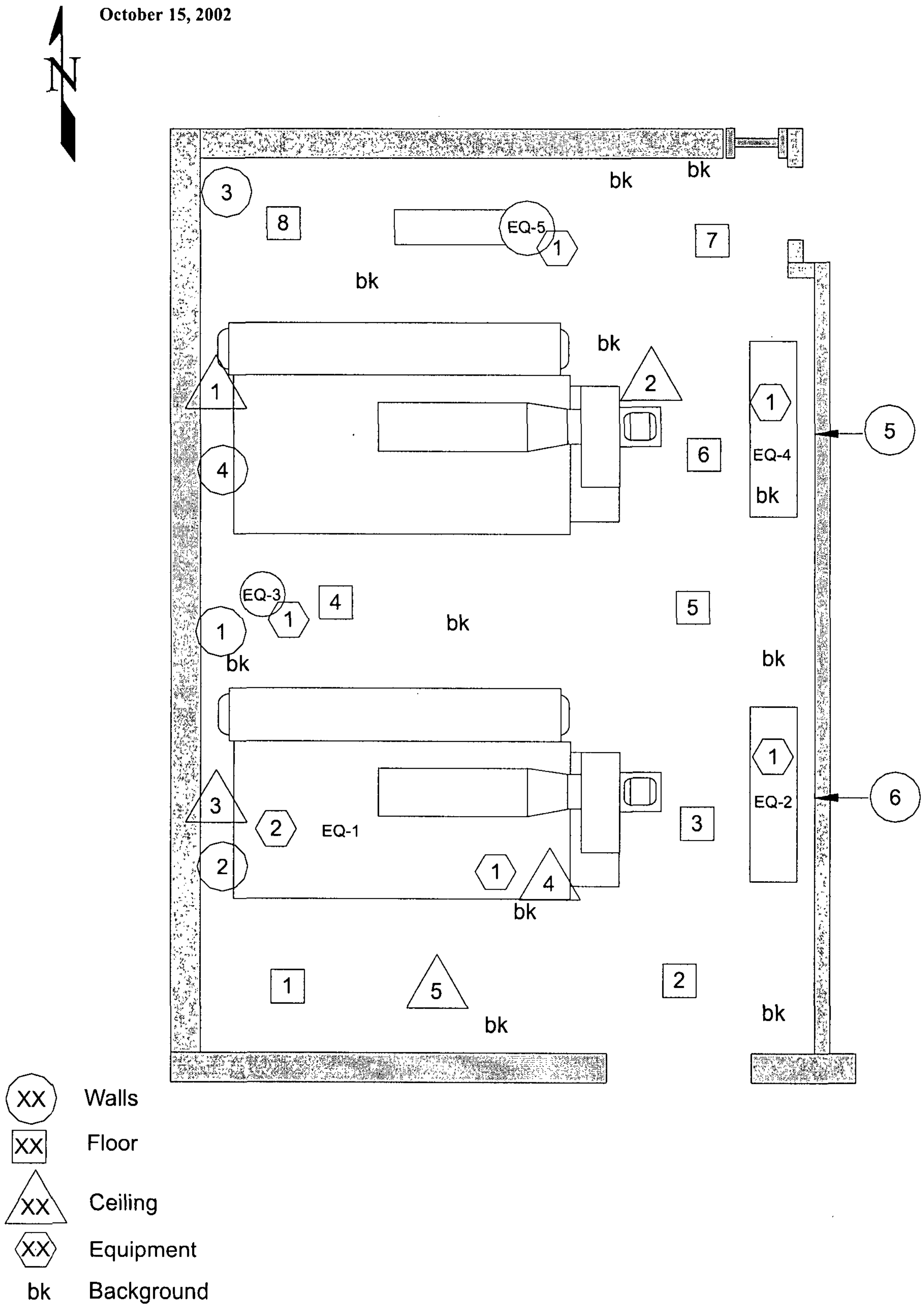
Site Characterization Fuel Oil Storage Building
Survey Package B0800

Figure 2-74

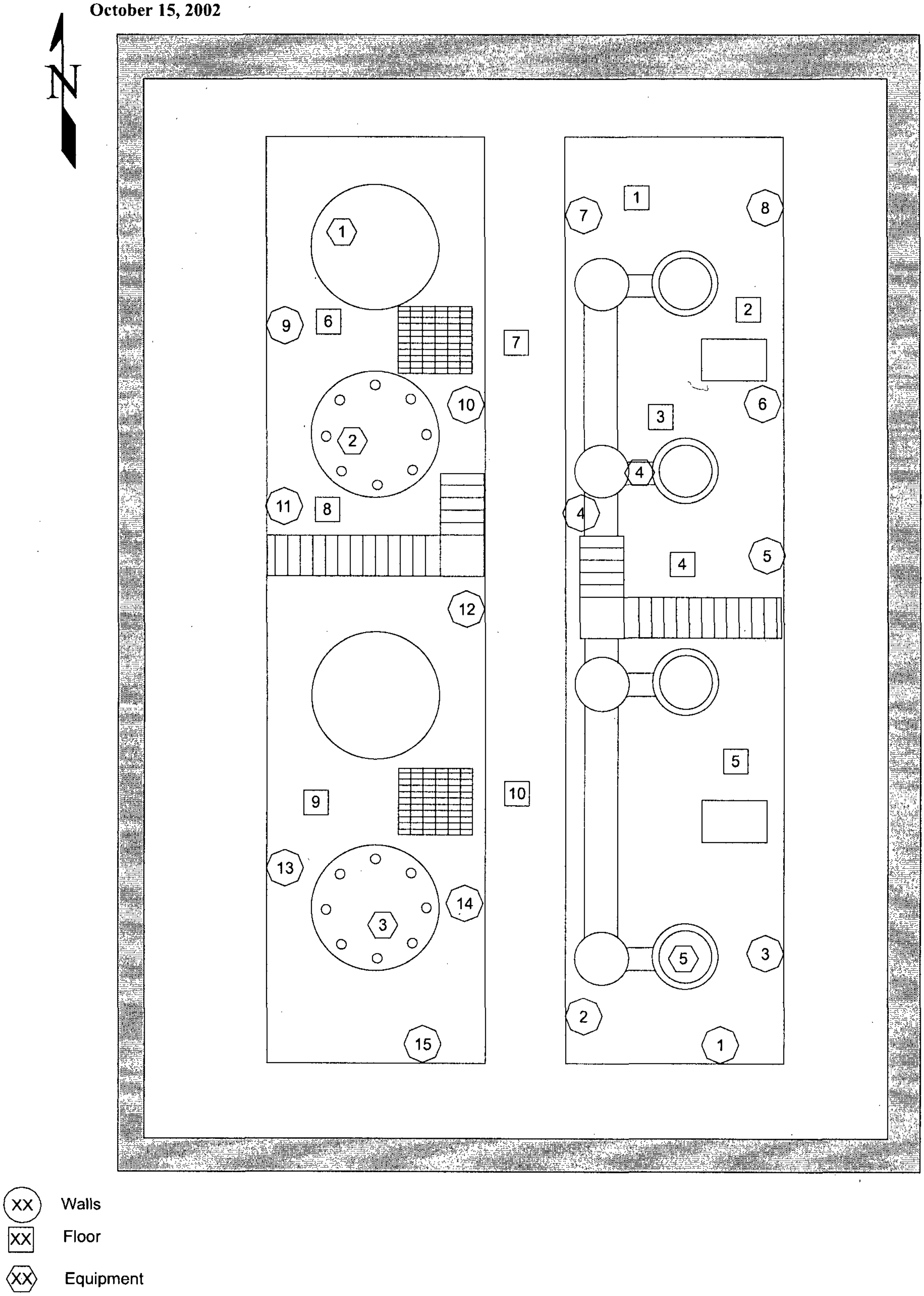


Site Characterization Diesel Generators Survey Package B0900

Figure 2-75



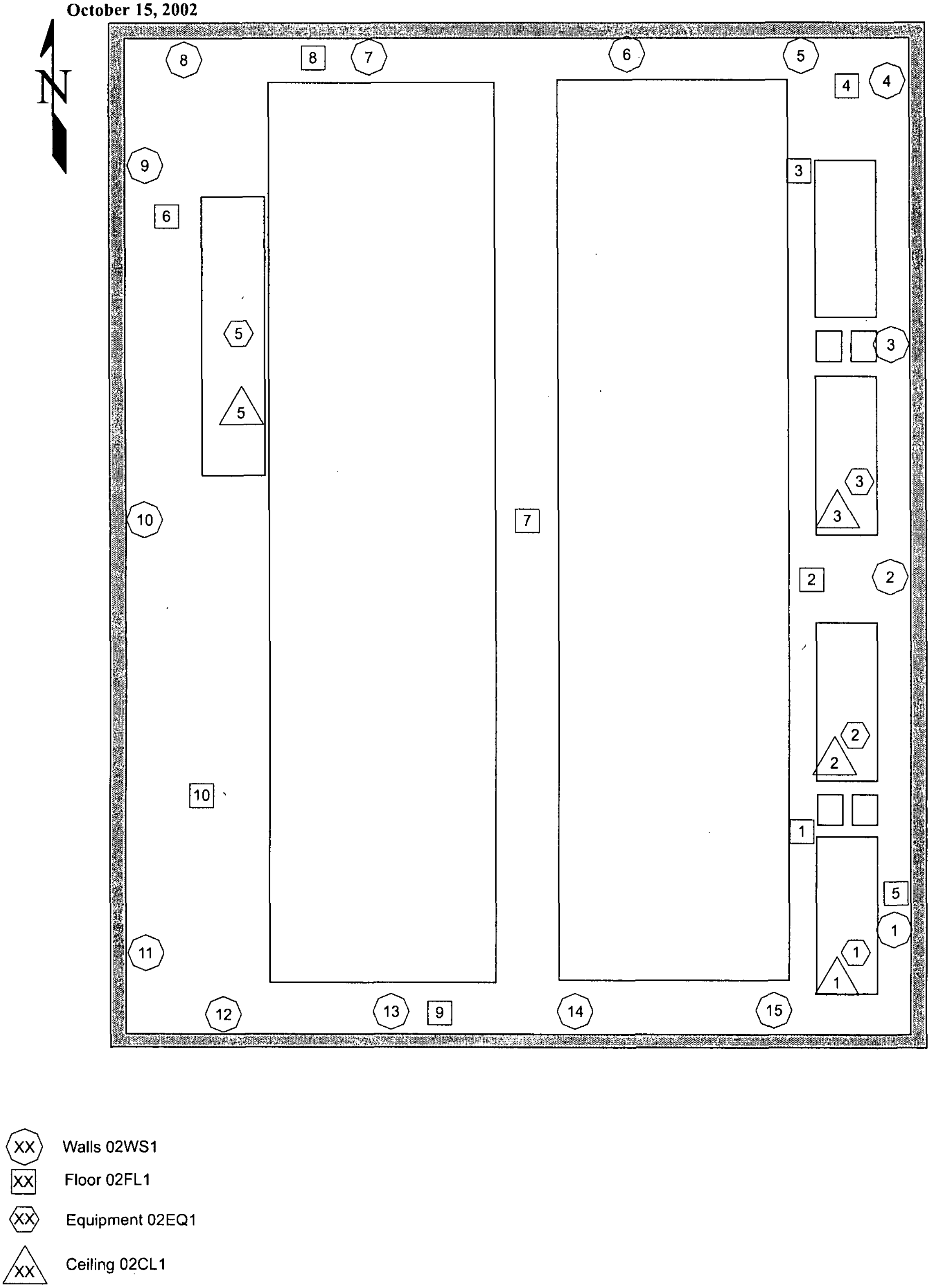
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Auxiliary Boiler Room Survey Package B1000	Figure 2-76



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization Recirc Water Pump House Lower Elevation Survey Package B1100

Figure 2-77



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization Recirc Water Pump House Upper Elevation Survey Package B1100

Figure 2-78

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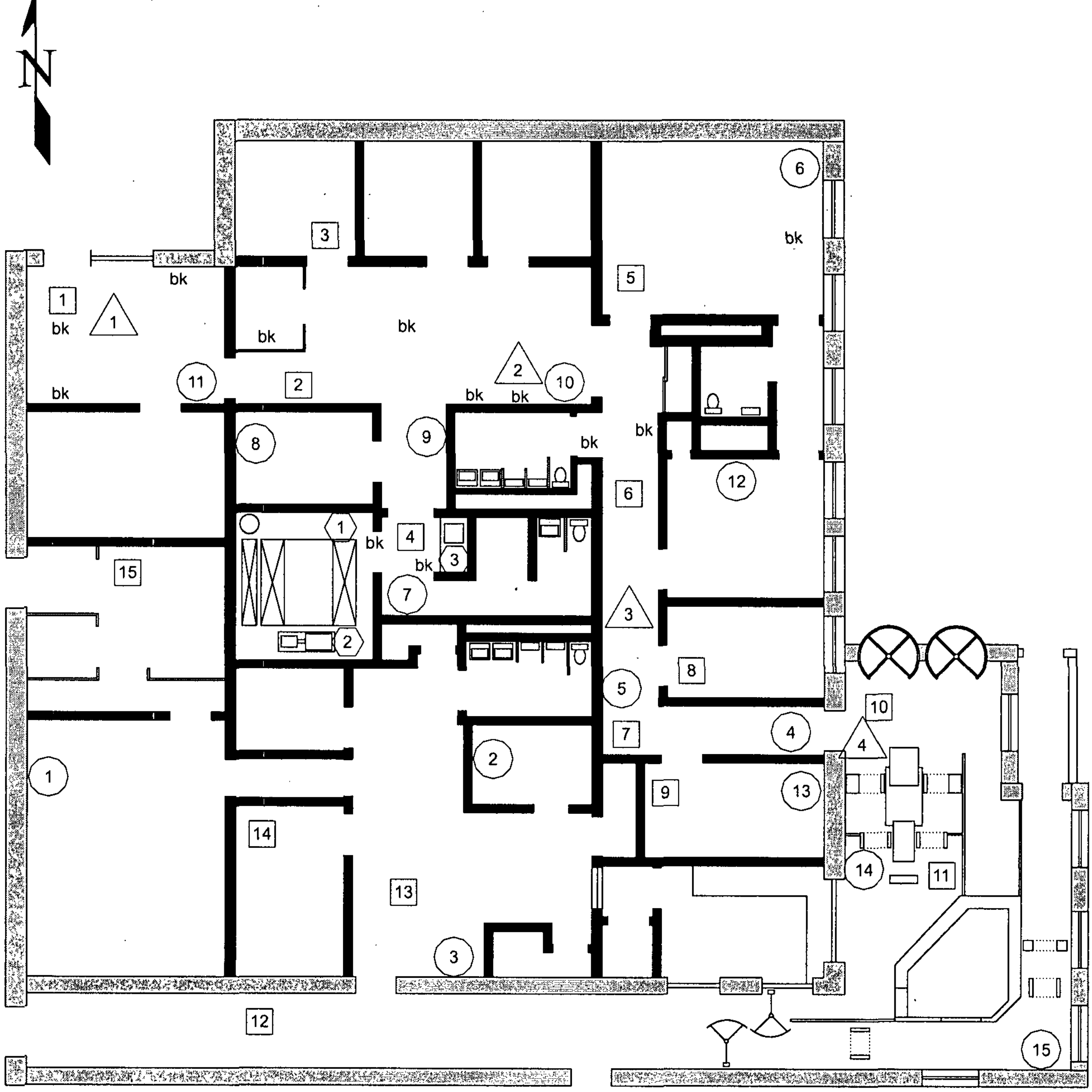


XX Walls 01WS1

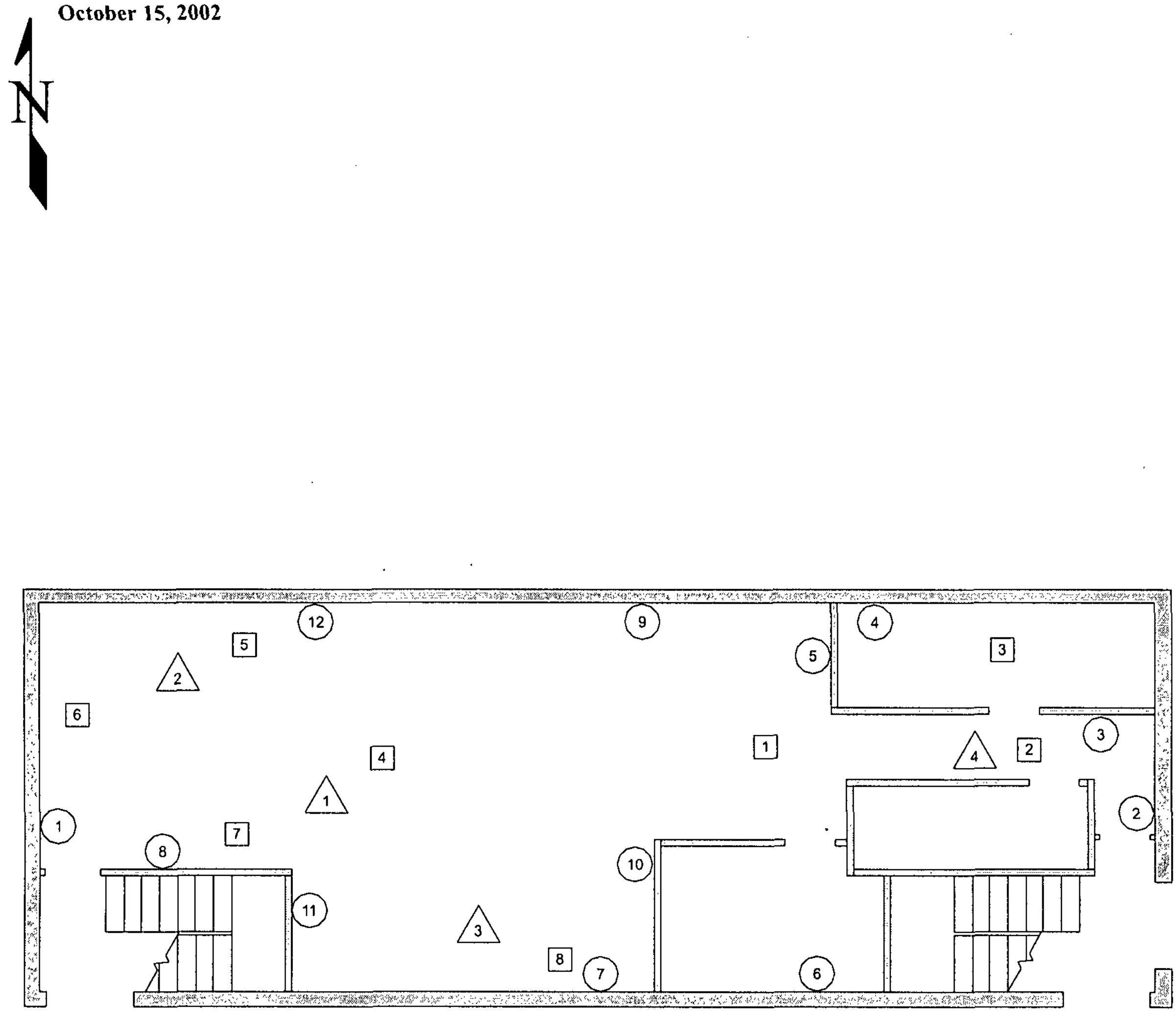
XX Floor 01FL1

XX Equipment 01EQ1, Q2, Q3

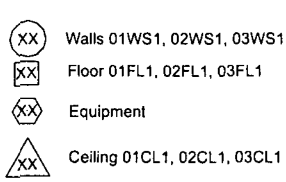
XX Ceiling 01CL1

bk Background

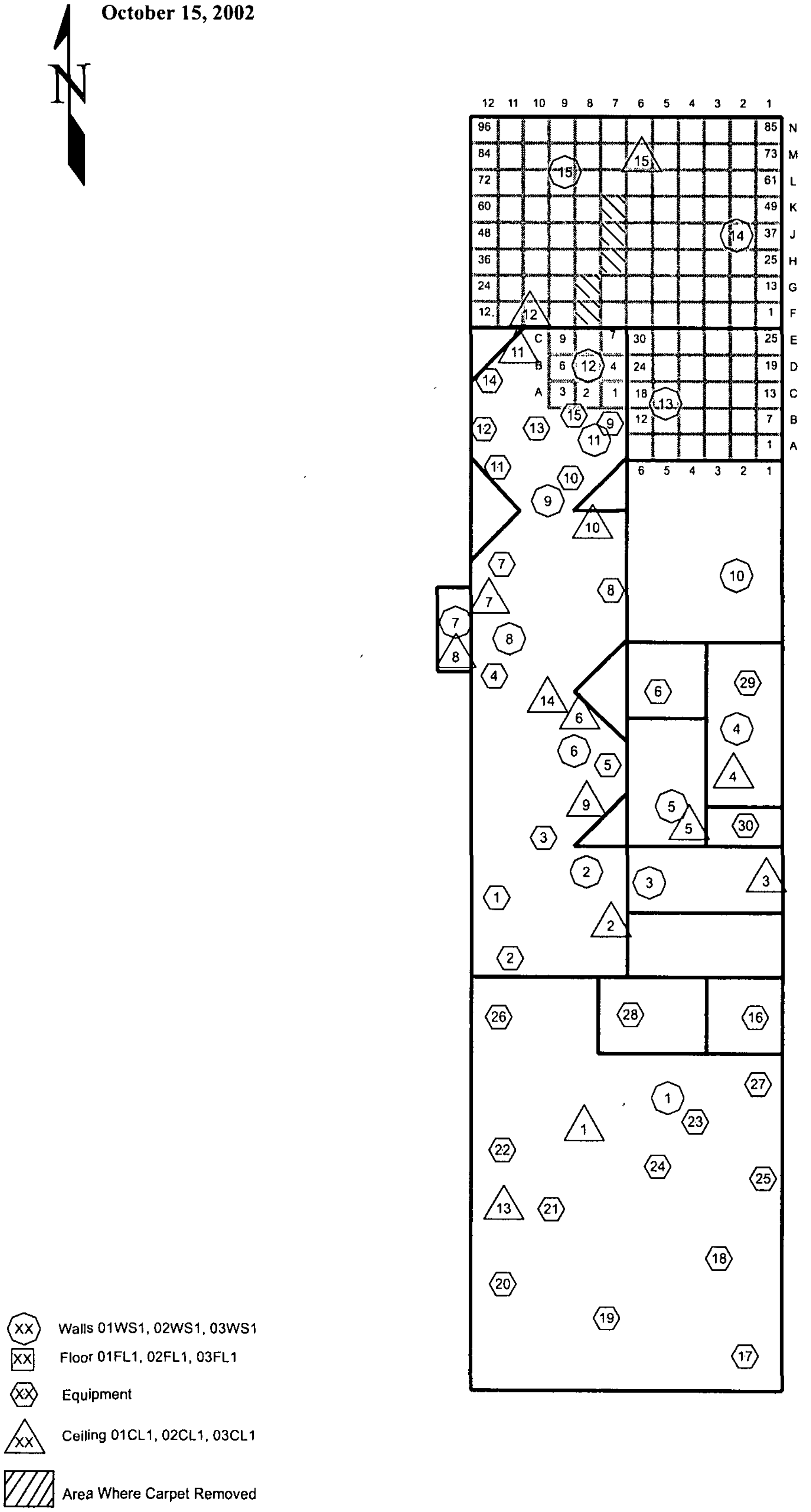
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Administration Building Front office Survey Package B1100	Figure 2-79



First Floor / I & C Shop



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization New Office Building (WART Bldg.) Survey Package B1300	Figure 2-80



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Visitor and Information Center Survey Package B1400	Figure 2-81

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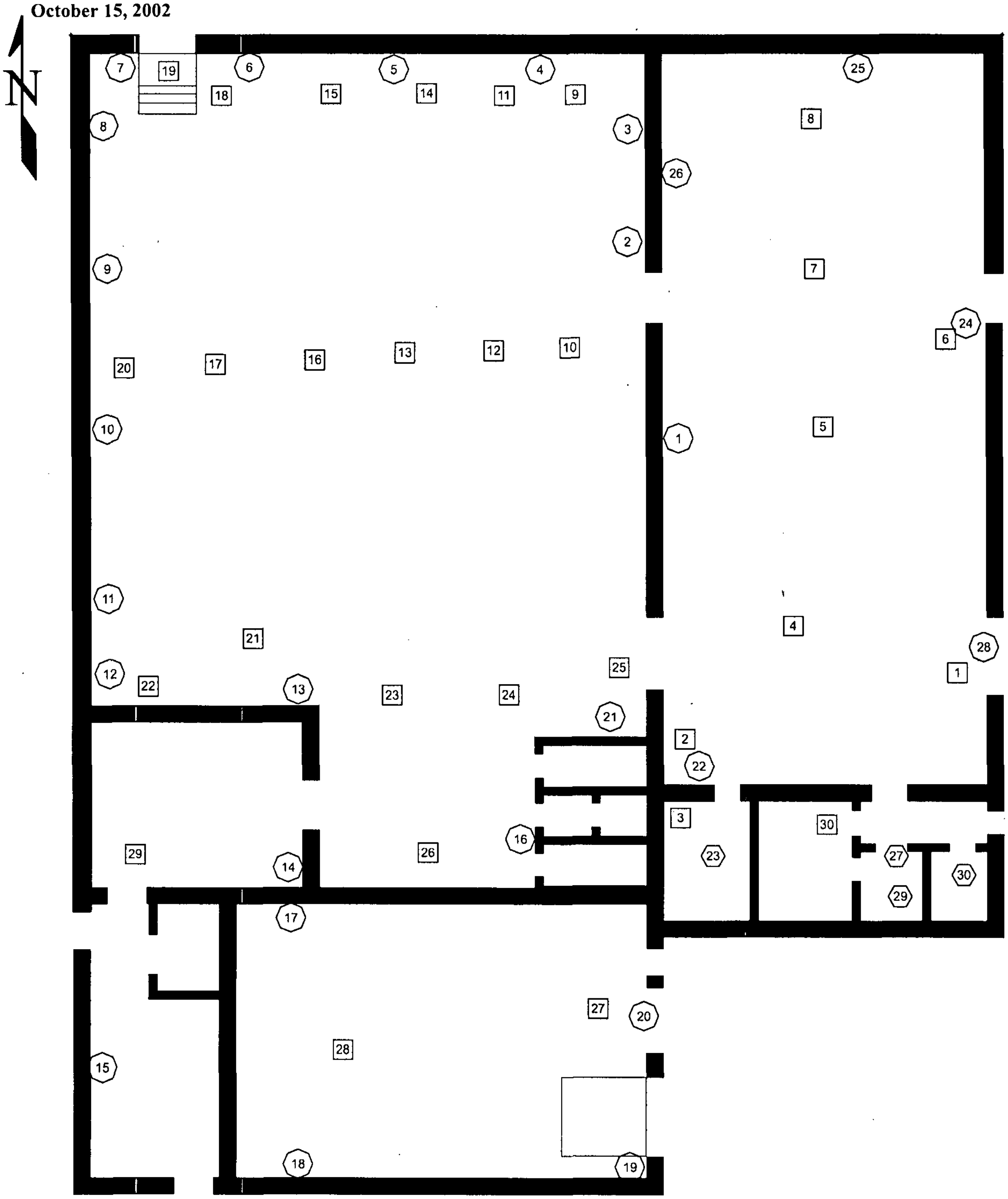
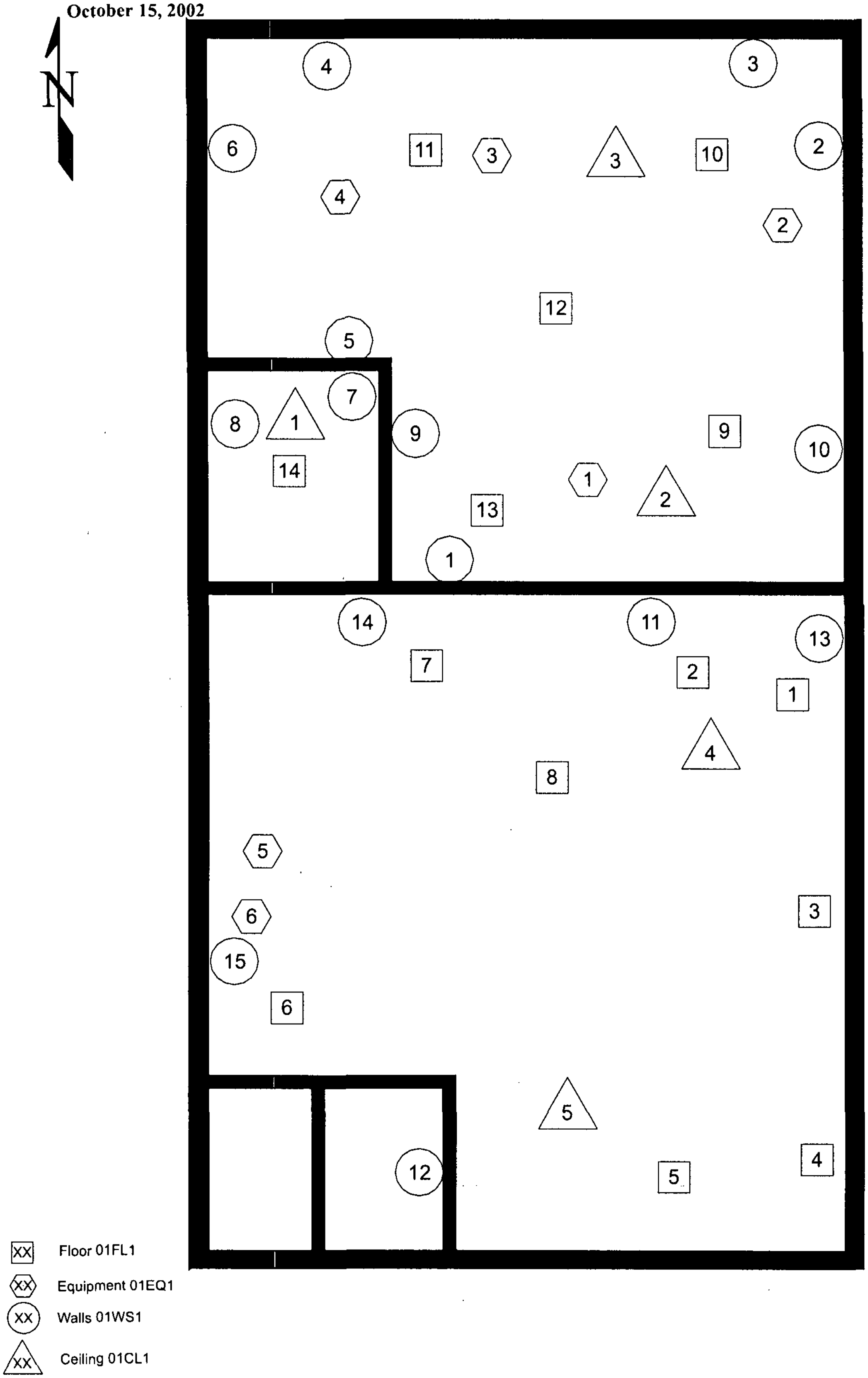


XX Floor 01FL1

XX Ceiling 02WC1

XX Walls 02WC1

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Warehouse 2 Survey Package B1500	Figure 2-82



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Training Annex Survey Package B1600	Figure 2-83

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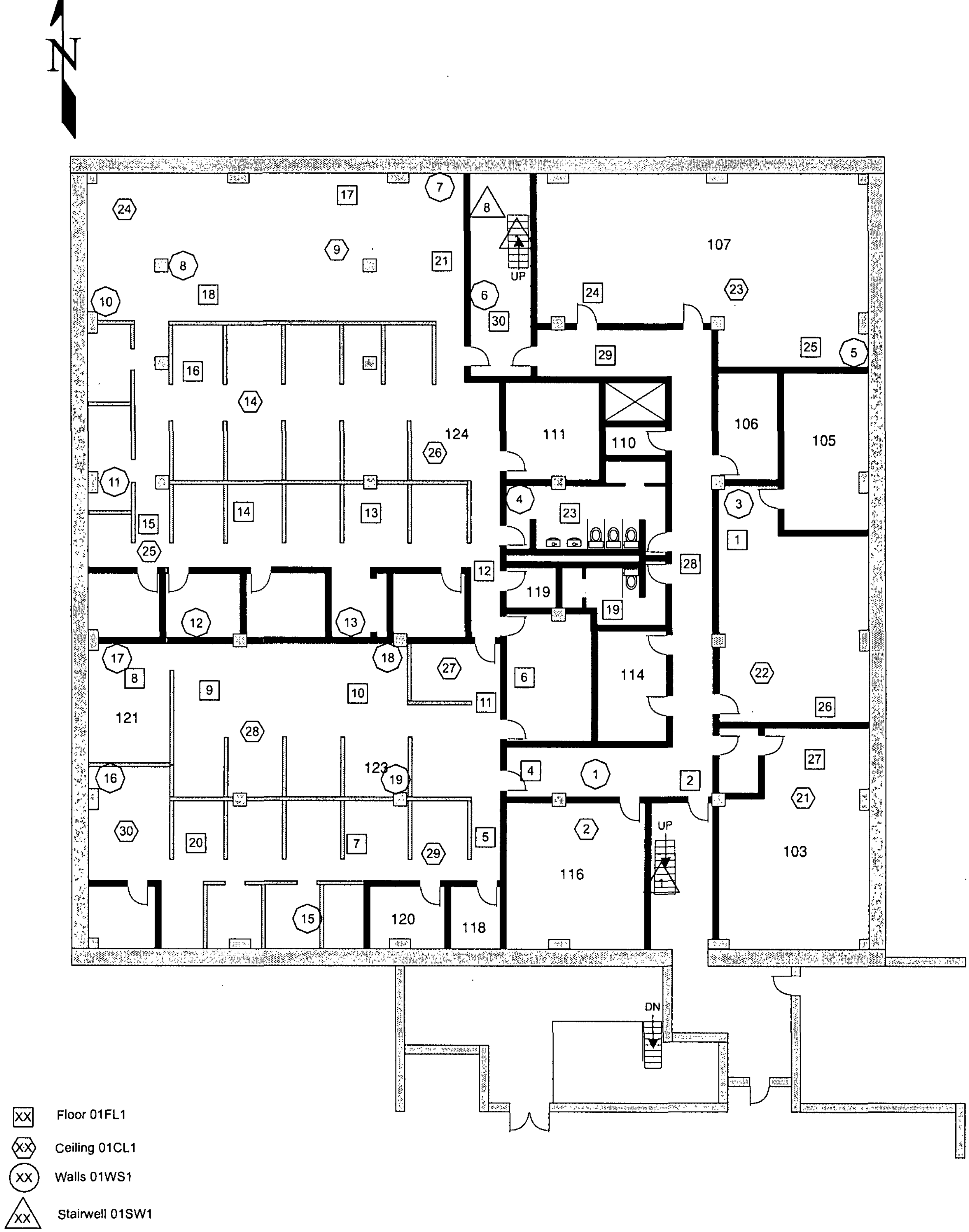
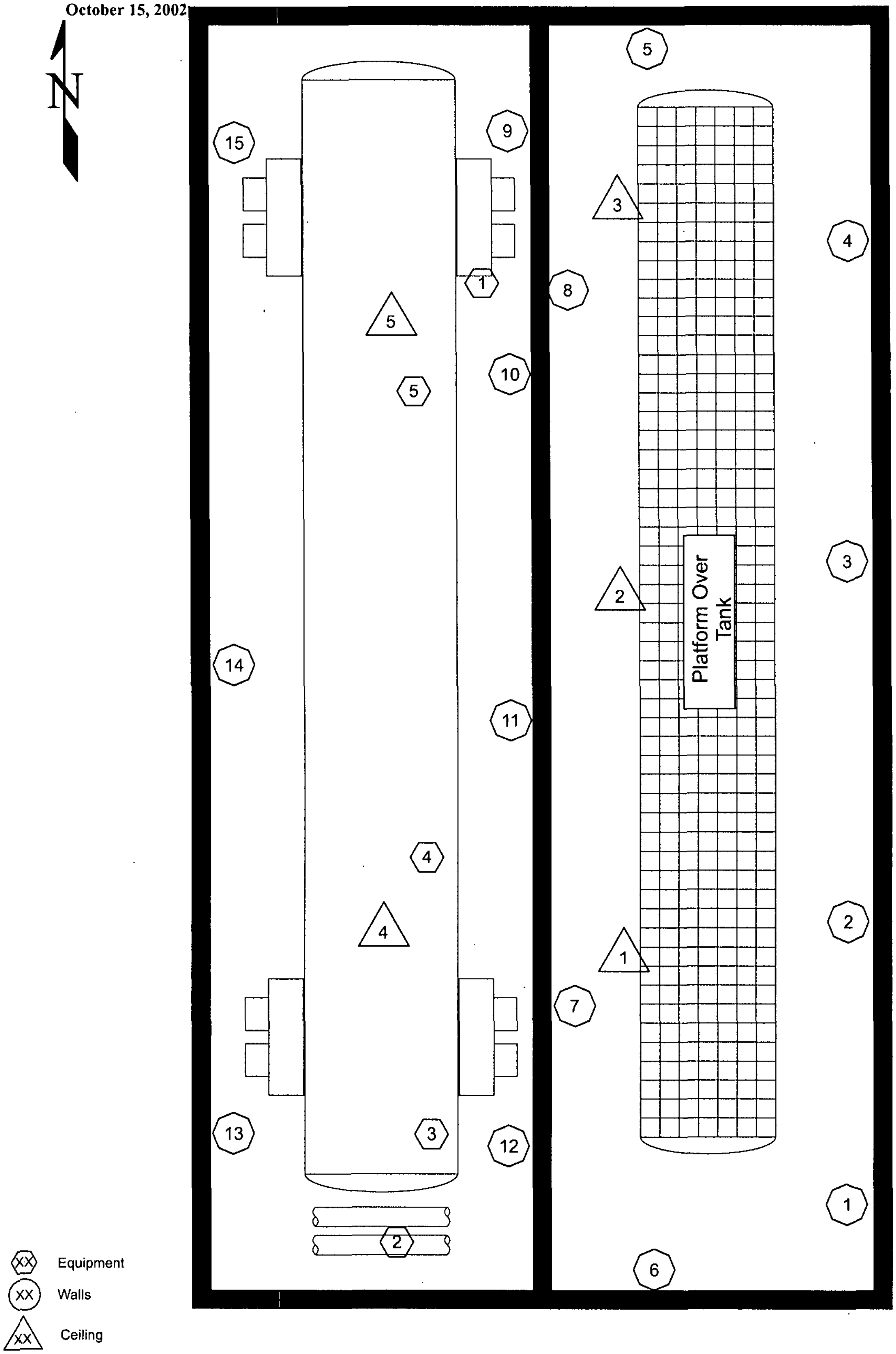
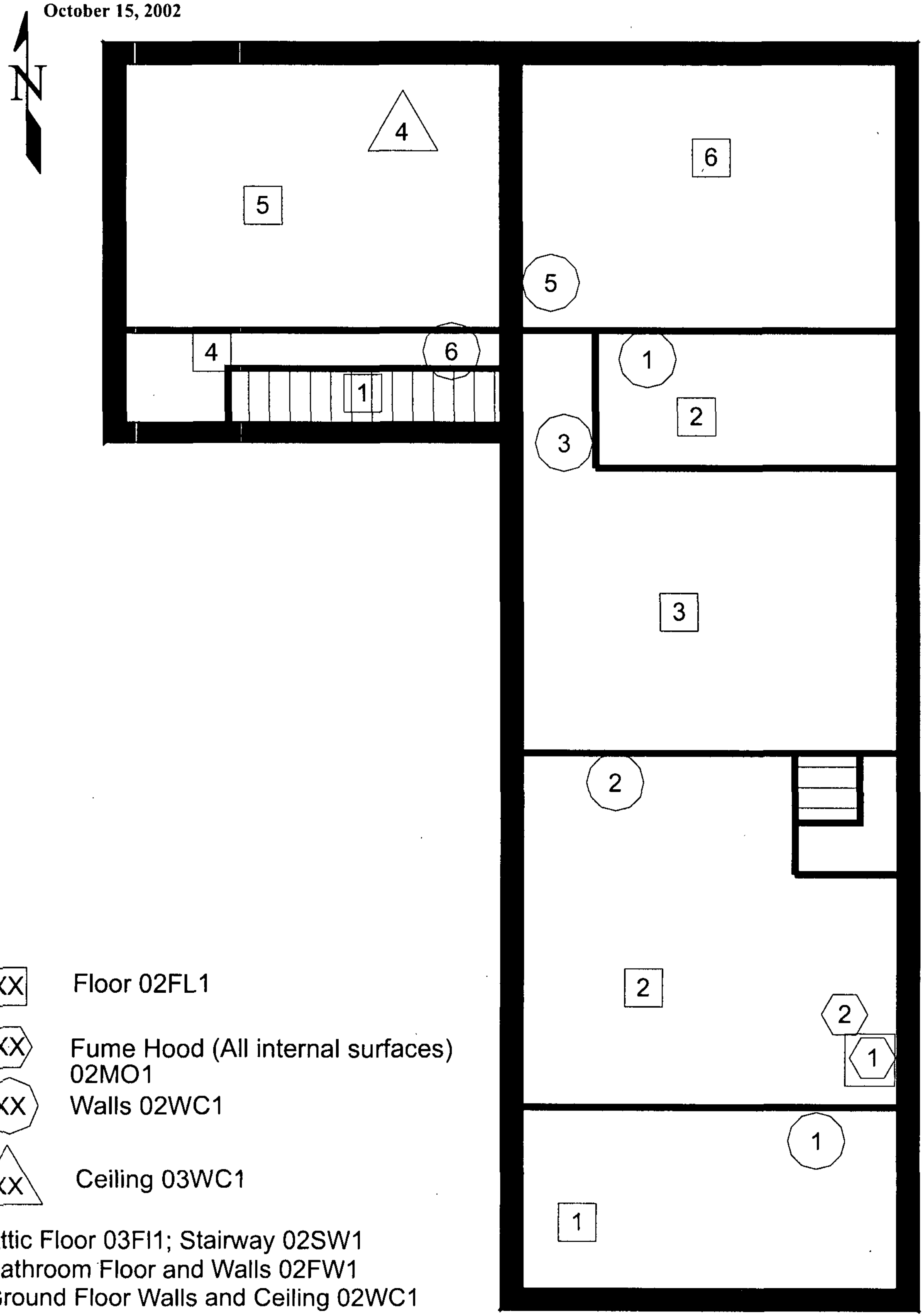


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Staff Building 1st Floor (22ft) Survey Package B1700	Figure 2-84



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Spare Generator Building Survey Package B1800	Figure 2-85



XX Floor 02FL1

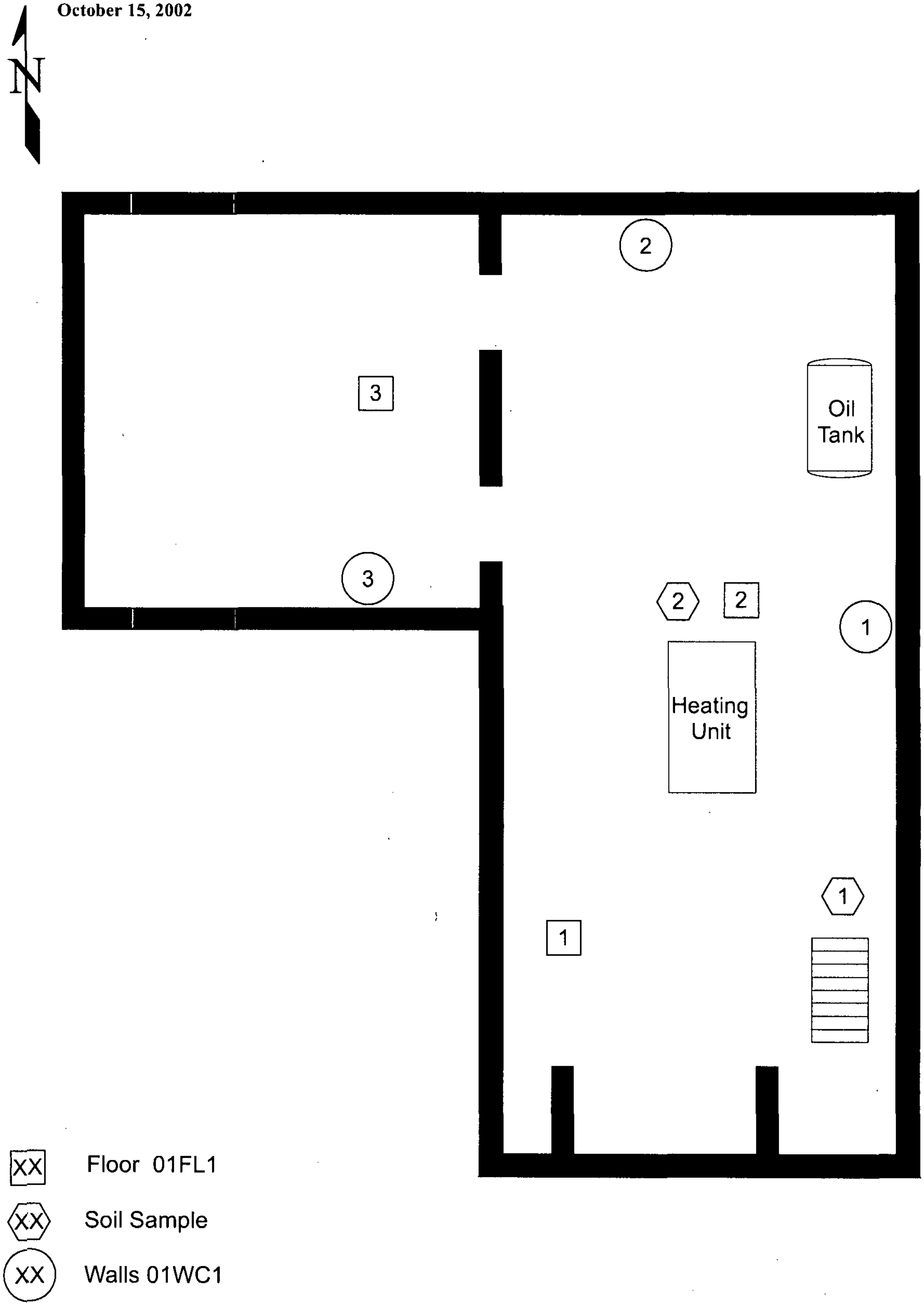
XX Fume Hood (All internal surfaces) 02MO1

XX Walls 02WC1

XX Ceiling 03WC1

Attic Floor 03FI1; Stairway 02SW1
Bathroom Floor and Walls 02FW1
Ground Floor Walls and Ceiling 02WC1

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Environmental Services Building Survey Package B1900	Figure 2-86



Site Characterization Environmental Services Building
Basement Survey Package B1900

Figure
2-87

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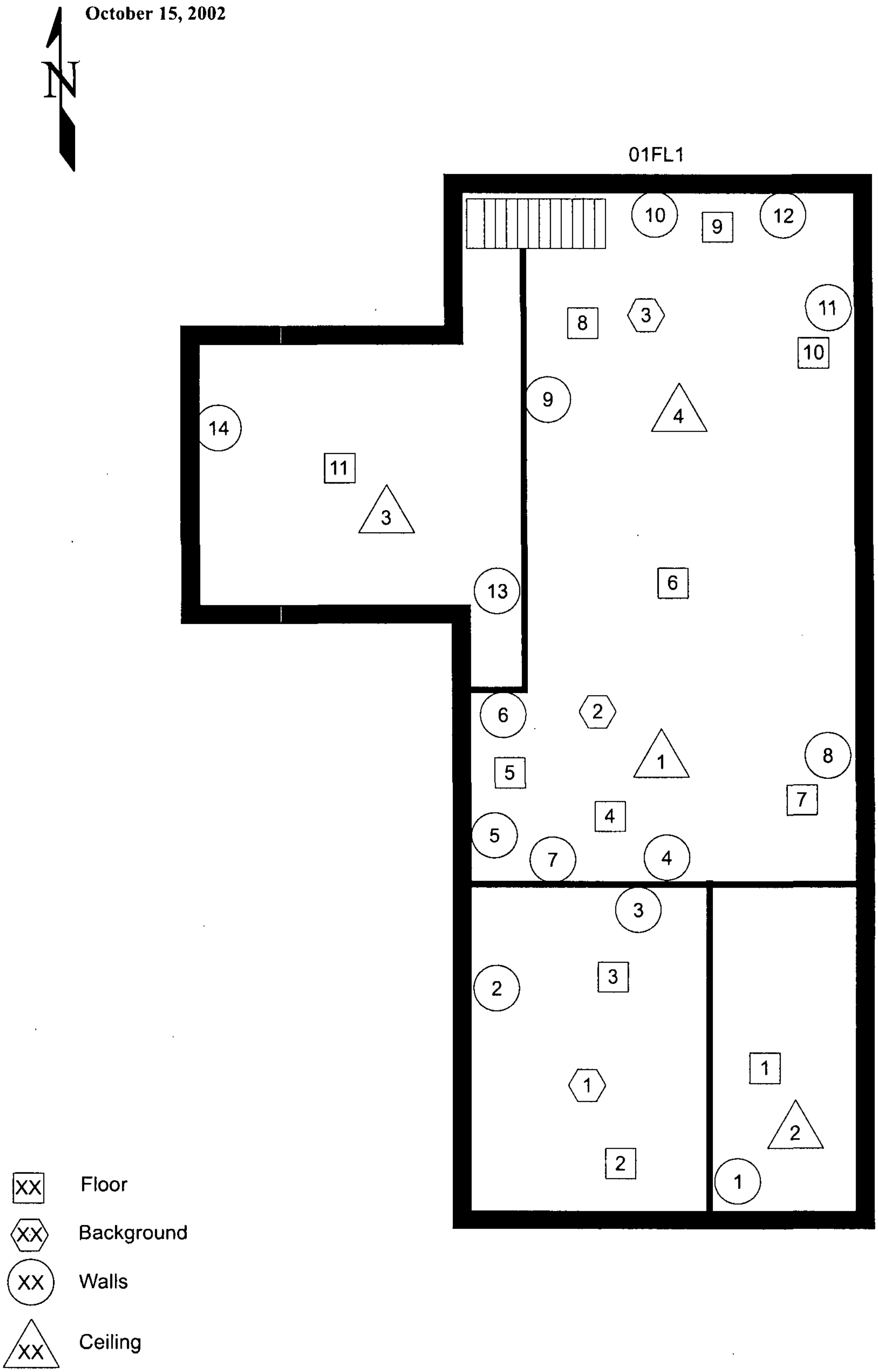
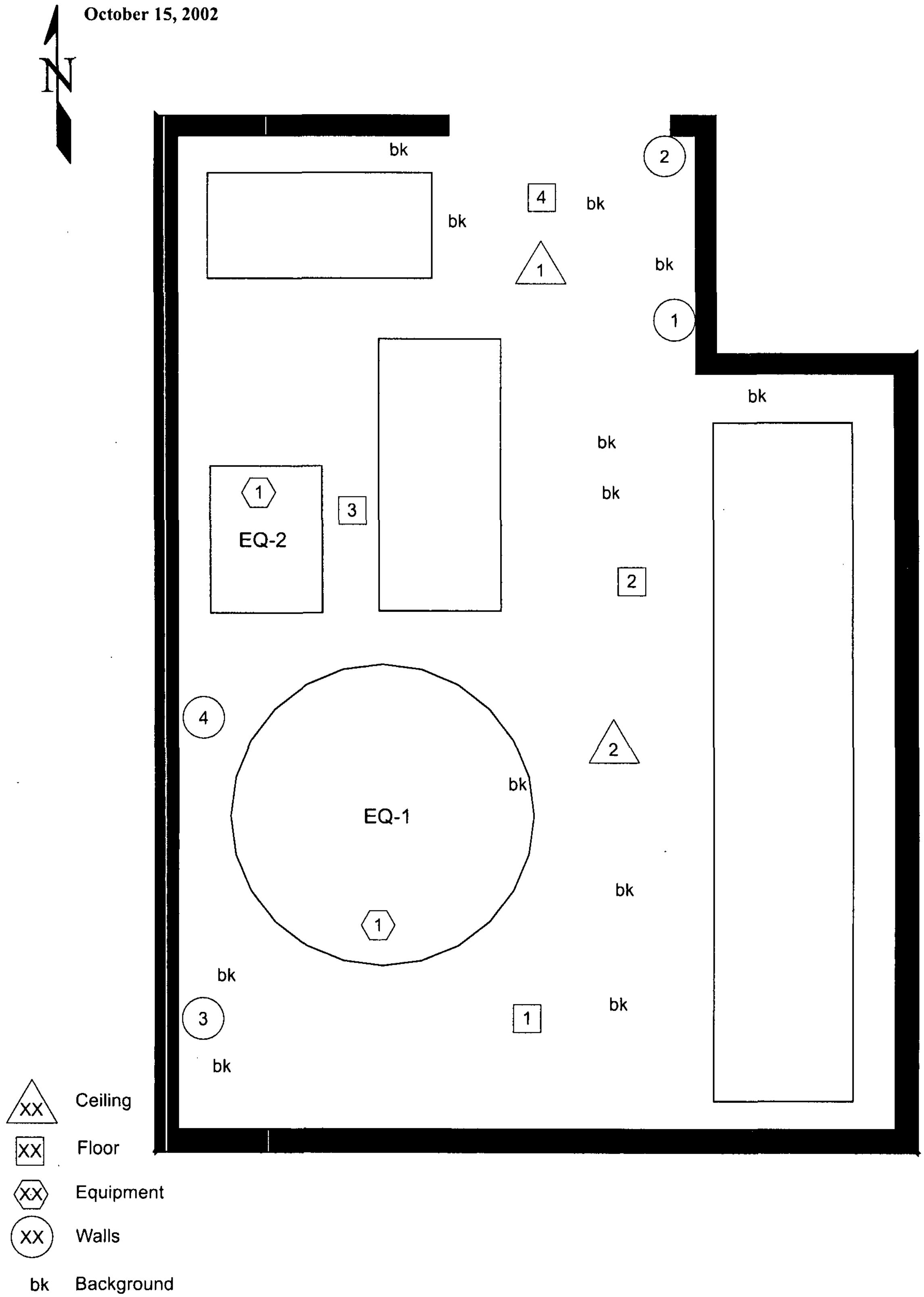
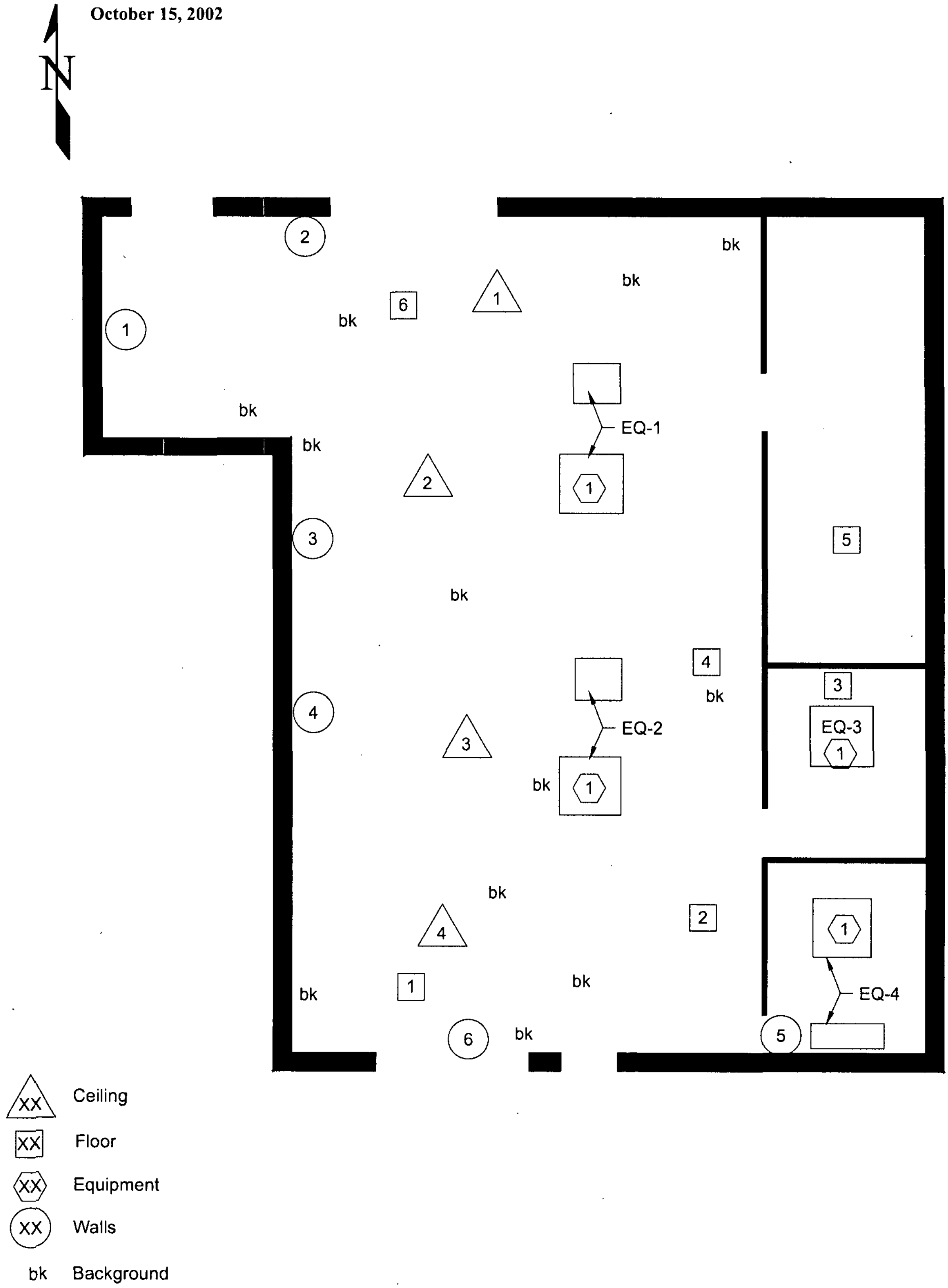


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Bailey Barn Survey Package B2000	Figure 2-88



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Lube Oil Storage Room 21 ft Survey Package B2100	Figure 2-89



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN

Site Characterization Cold Machine Shop Turbine Building 21 ft Survey Package B2200

Figure 2-90

MYAPC License Termination Plan
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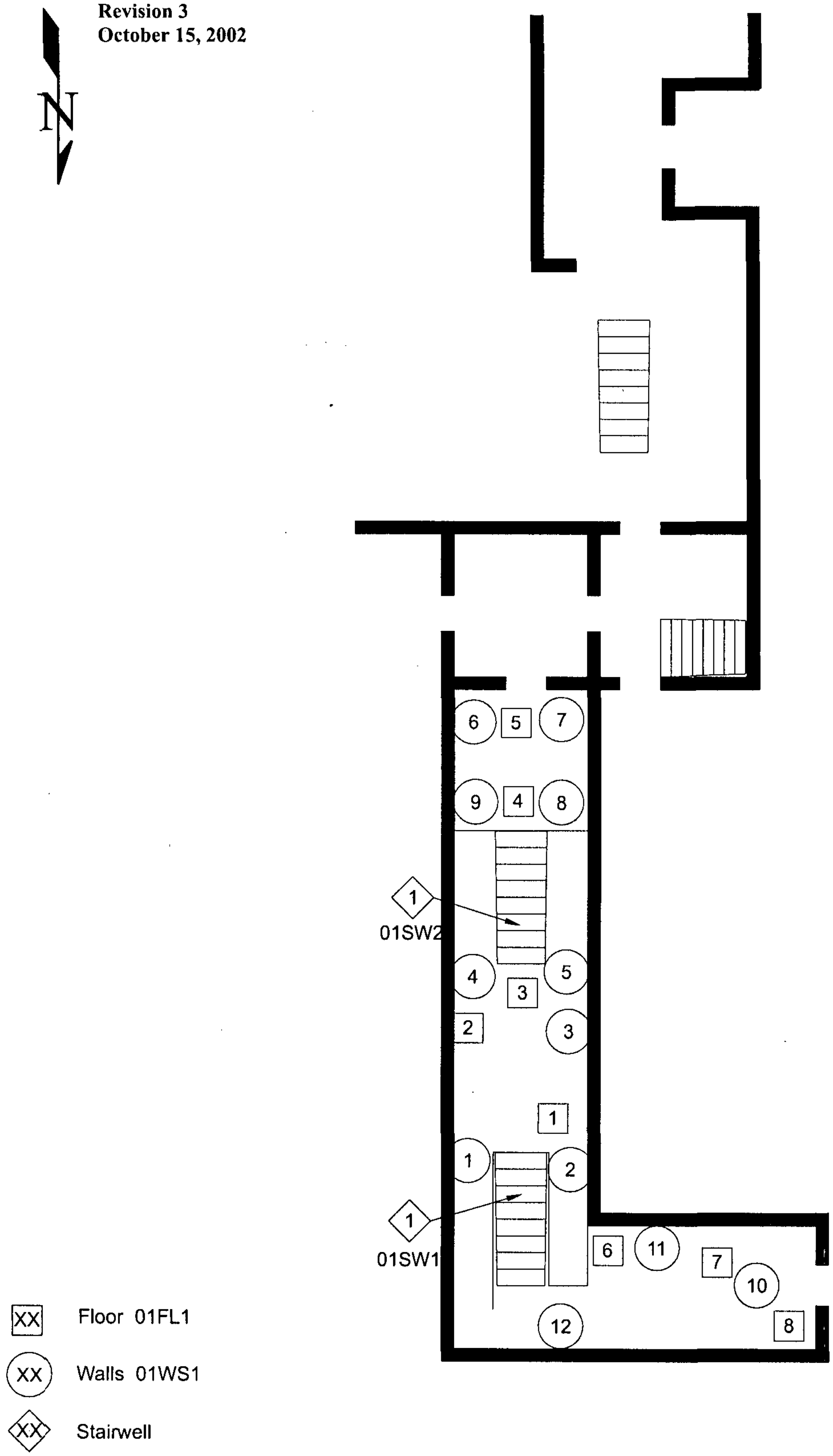
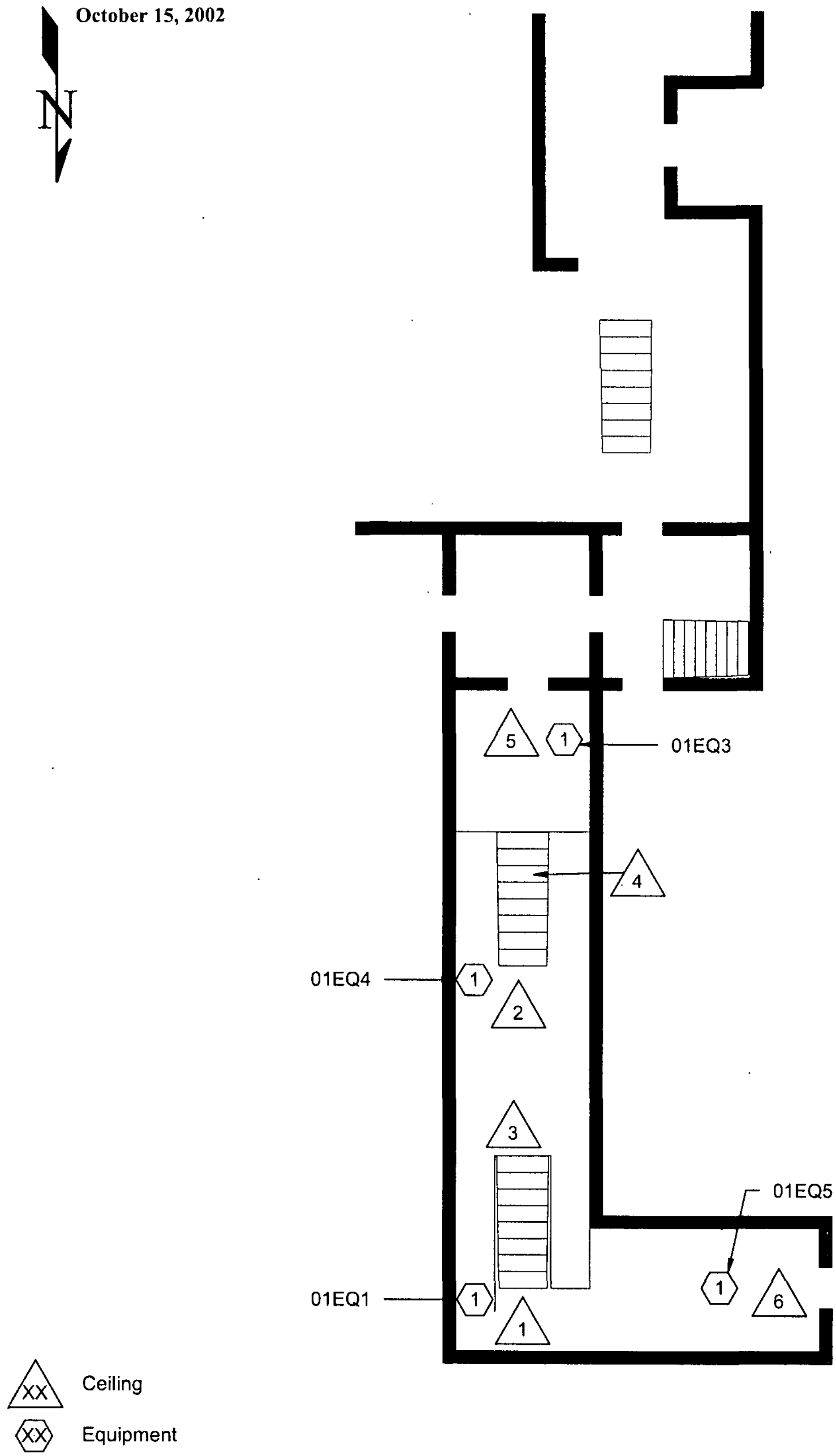


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization Staff building Tunnel Survey Package B2400	Figure 2-91



Site Characterization Staff building Tunnel Survey Package B2400

Figure 2-92

MYAPC License Termination Plan
Revision 3
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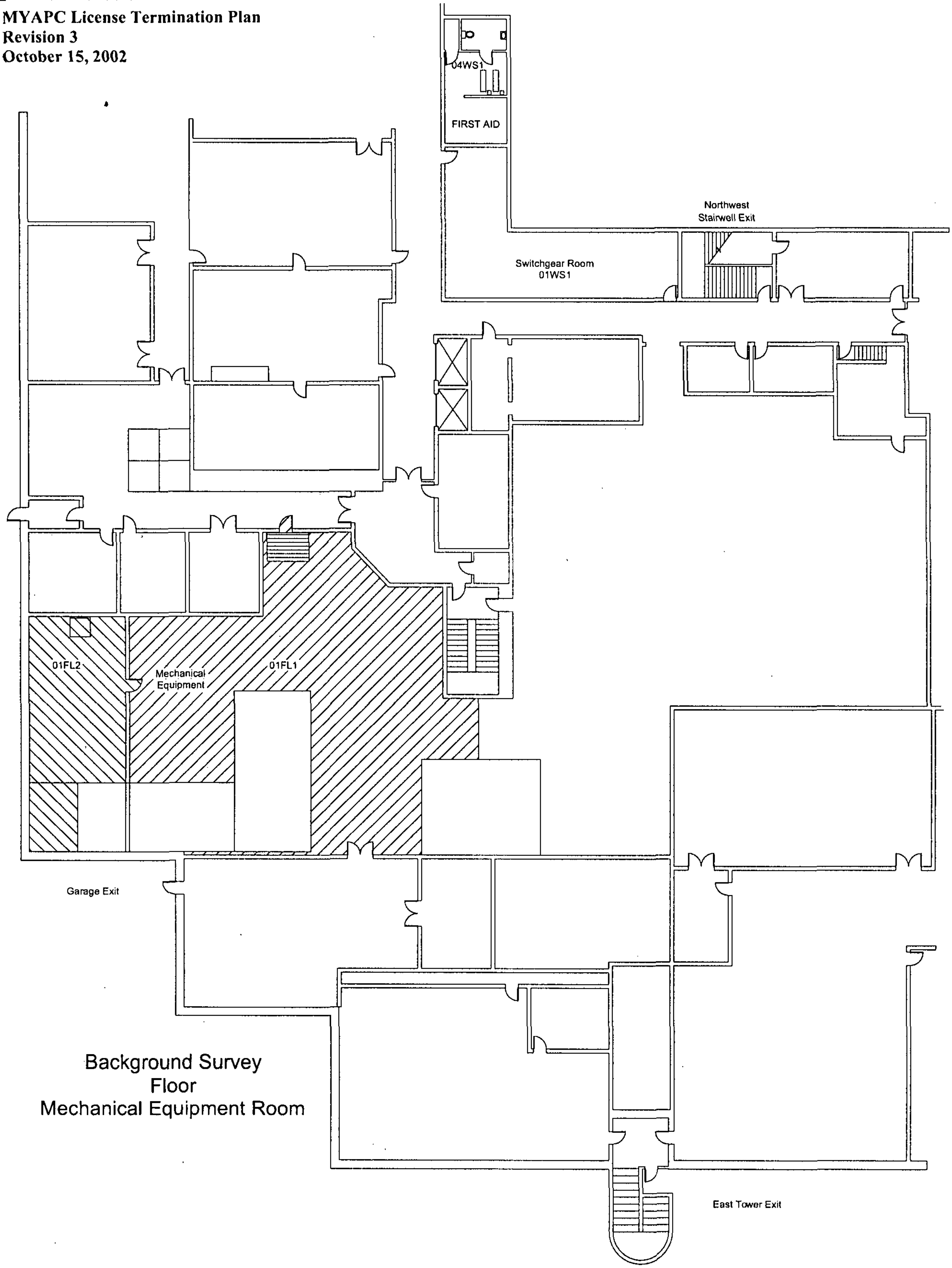
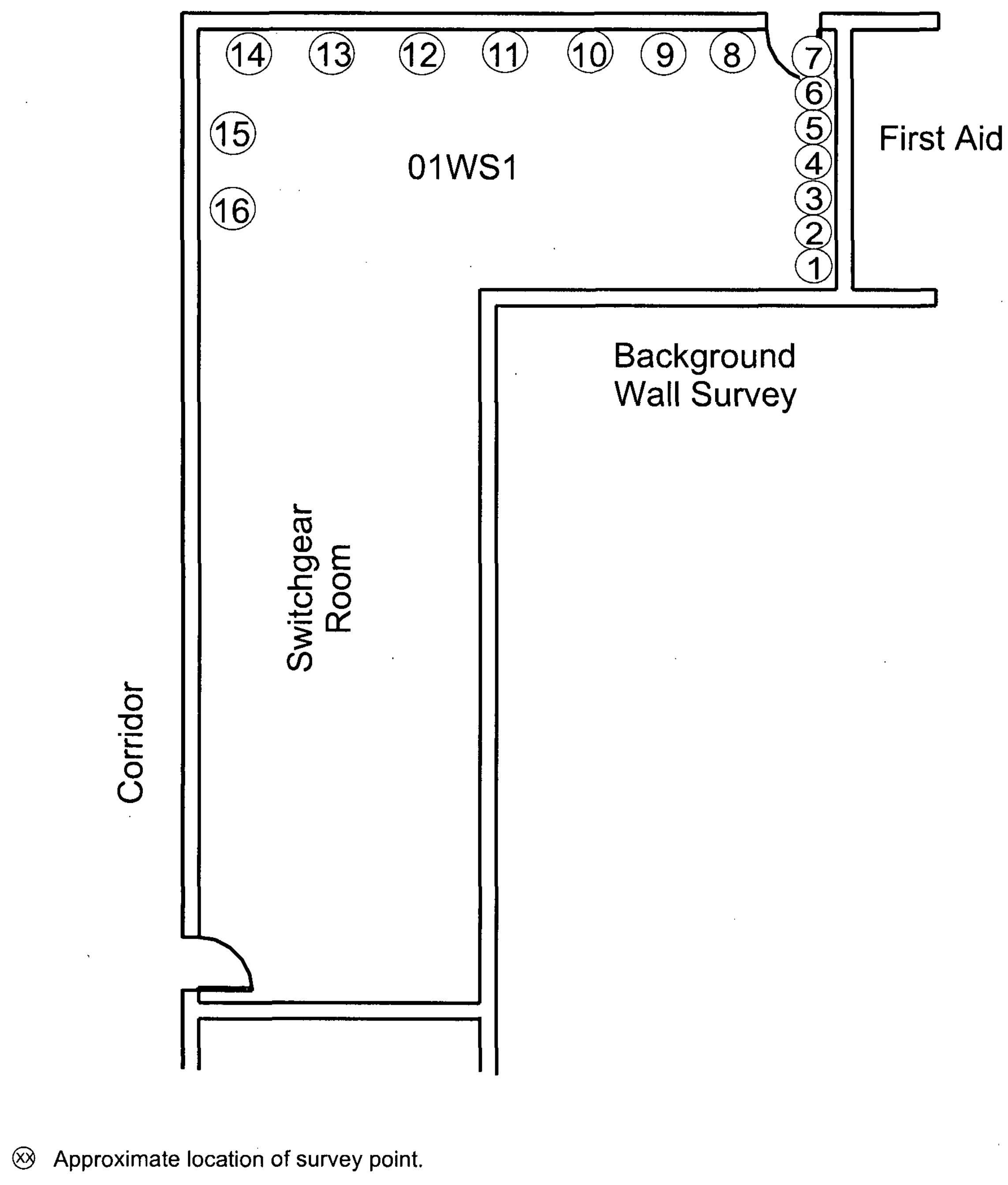
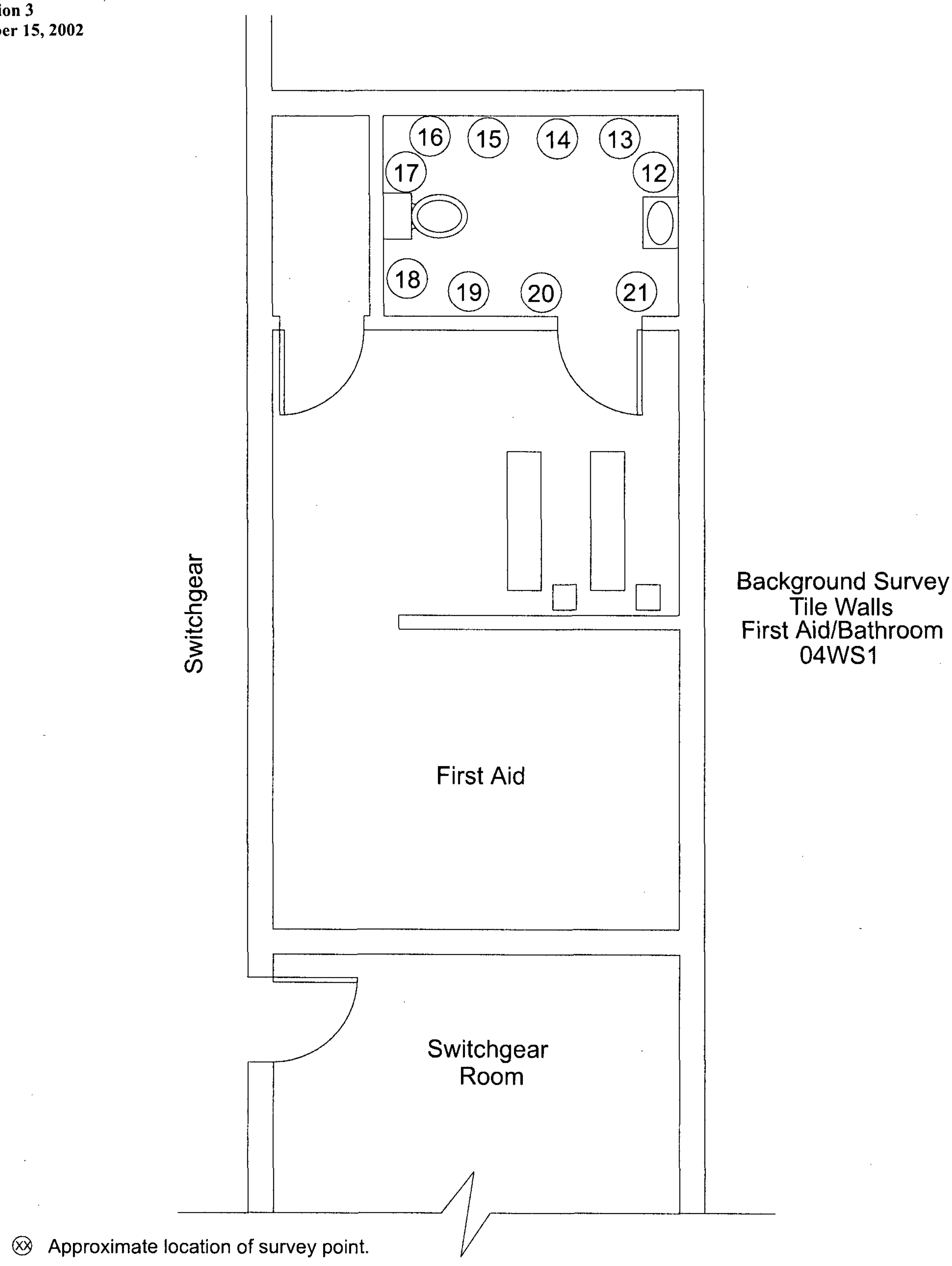


Background Survey
Floor
Mechanical Equipment Room

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP Building Augusta Survey Package B9800	Figure 2-93

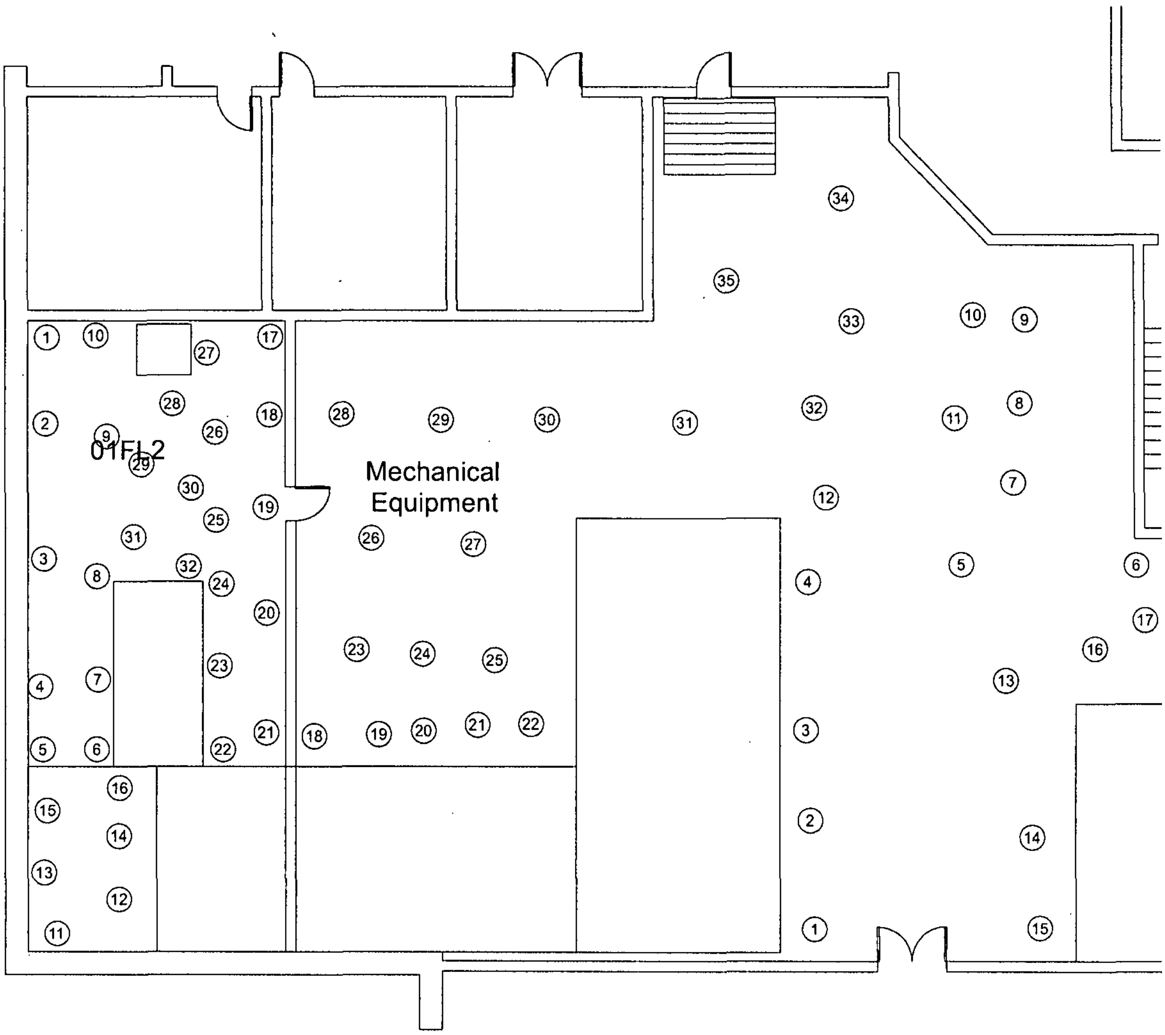


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP building Augusta Survey Package B9800	Figure 2-94



Site Characterization CMP Building Augusta Survey Package B9800

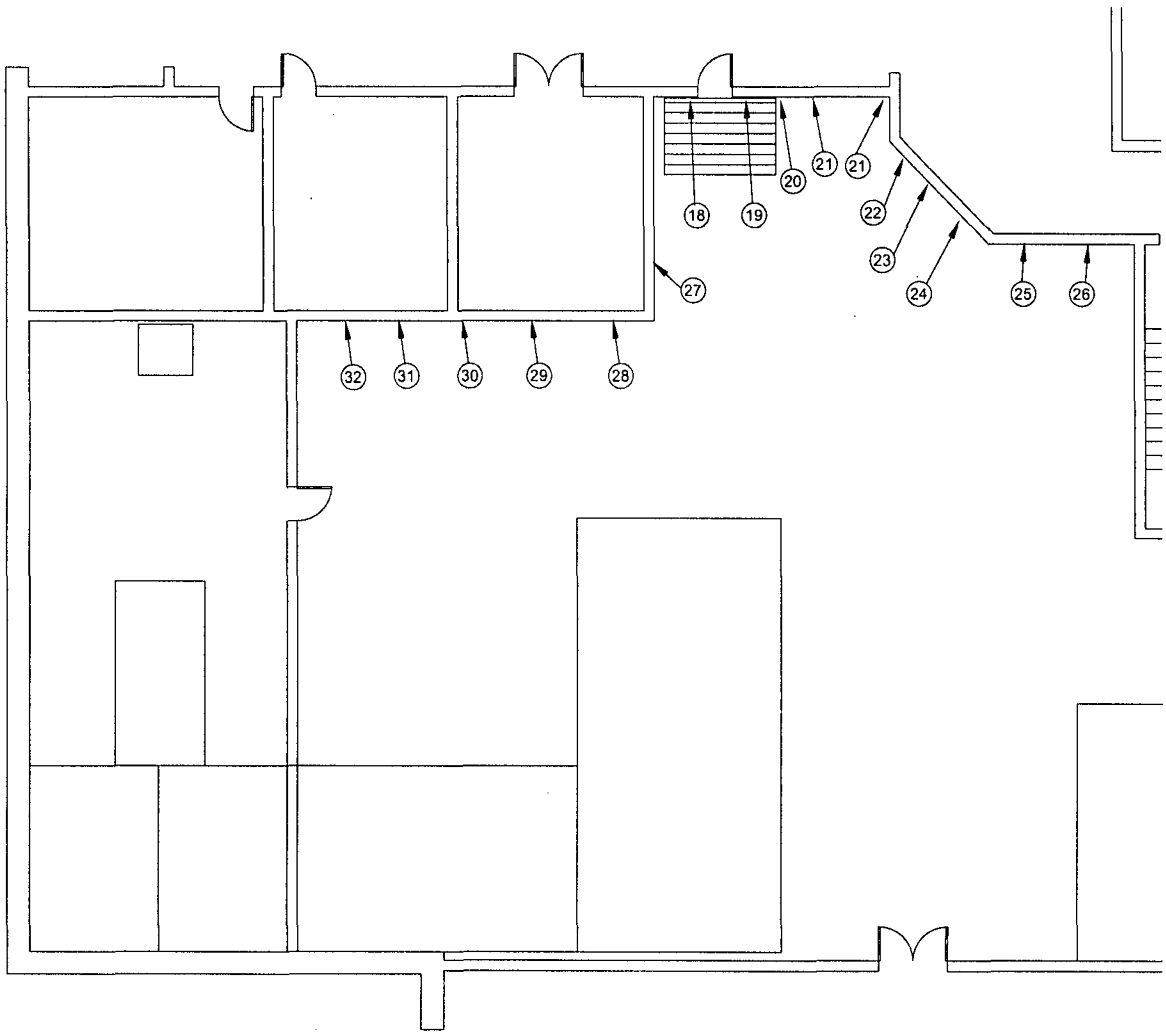
Figure 2-95



Floor- Garage Exit
Background Survey

(XX) Approximate location of survey point.

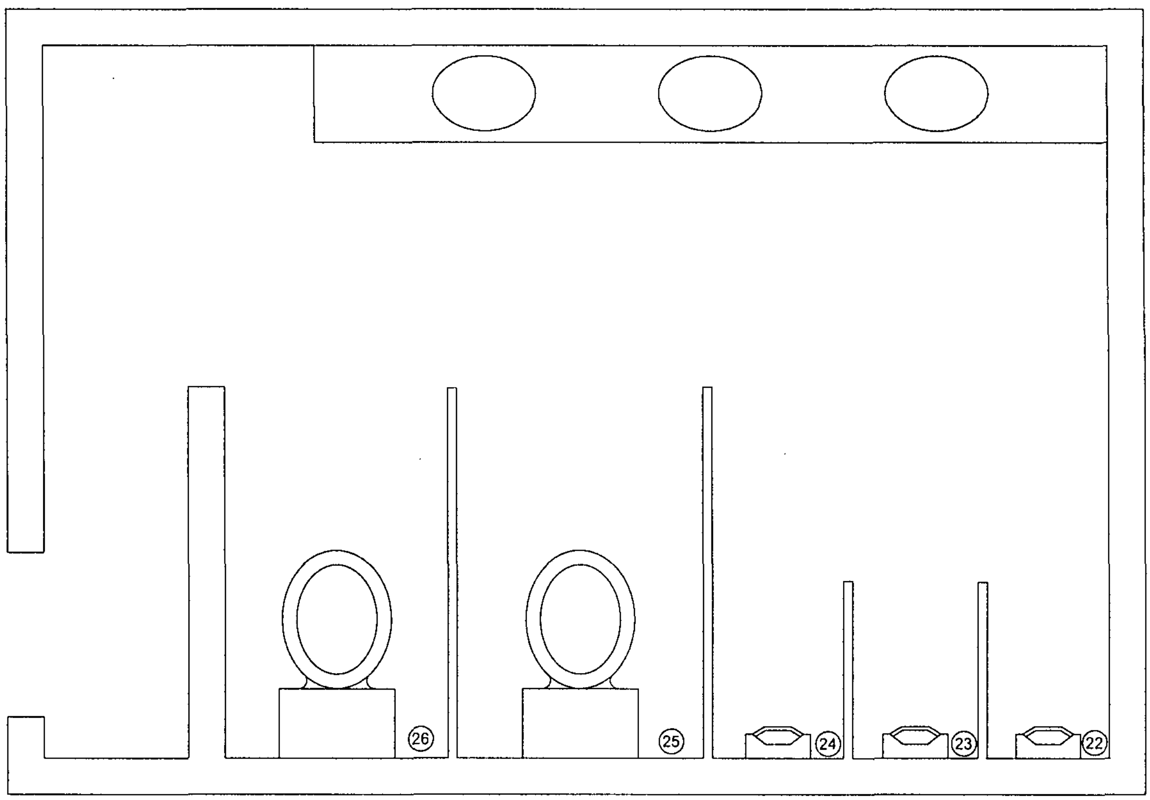
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP Building Augusta Survey Package B9800	Figure 2-96



Mechanical Equipment Room
Background Survey

ⓍⓍ Approximate location of survey point.

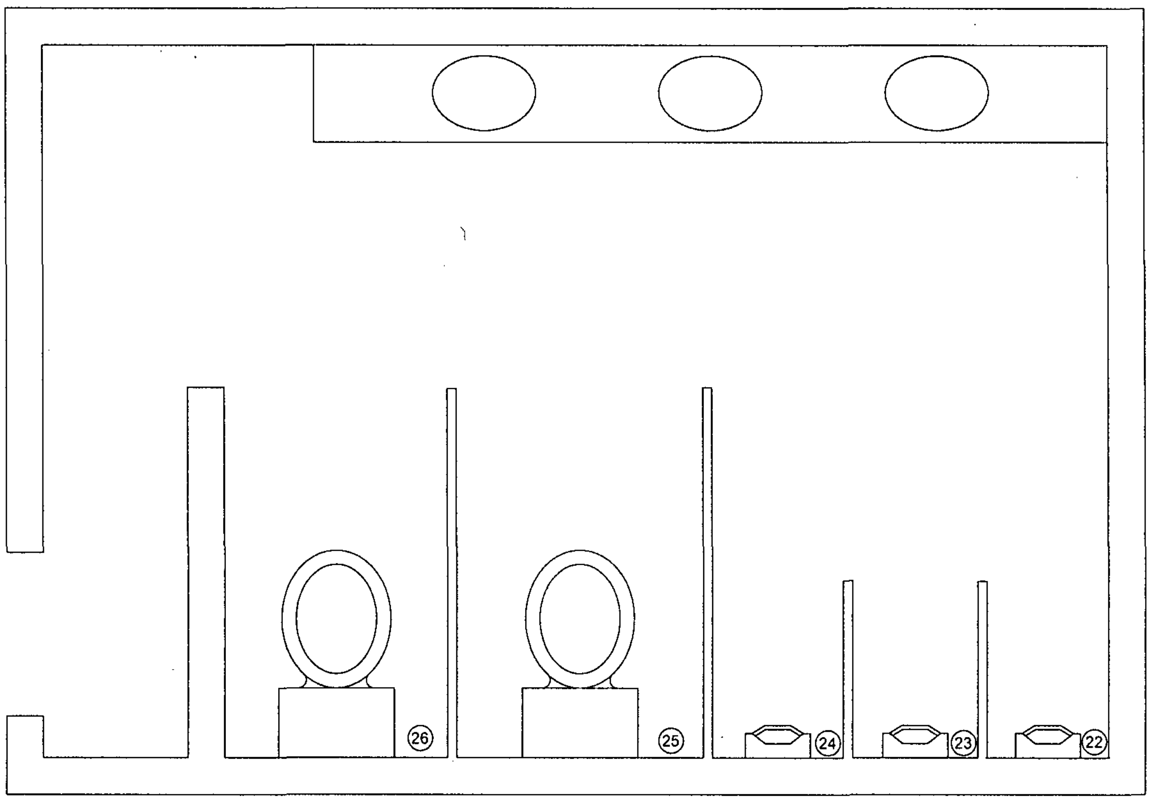
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP Building Augusta Survey Package B9800	Figure 2-97



2nd Floor Men's Bathroom
Tile Wall
Background Survey
04WS1

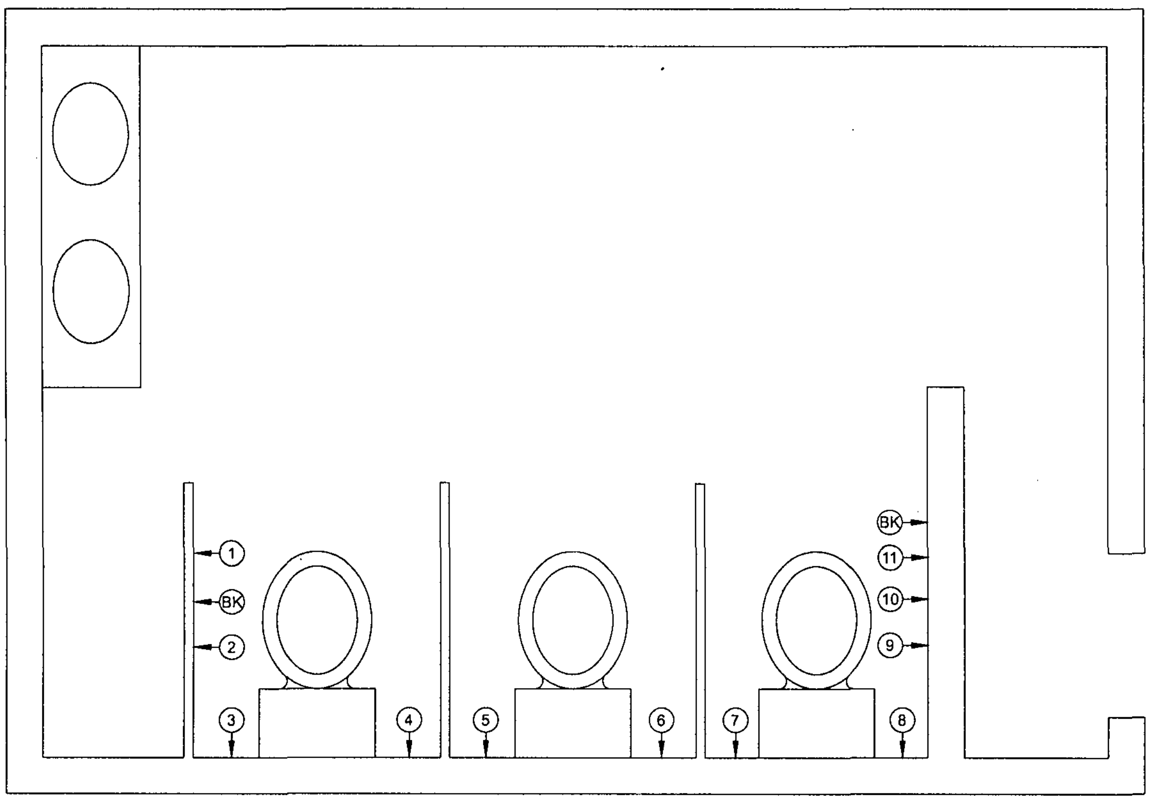
(XX) Approximate location of survey point.

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP Building Augusta Survey Package B9800	Figure 2-98



2nd Floor Men's Bathroom
Tile Wall
Background Survey
04WS1

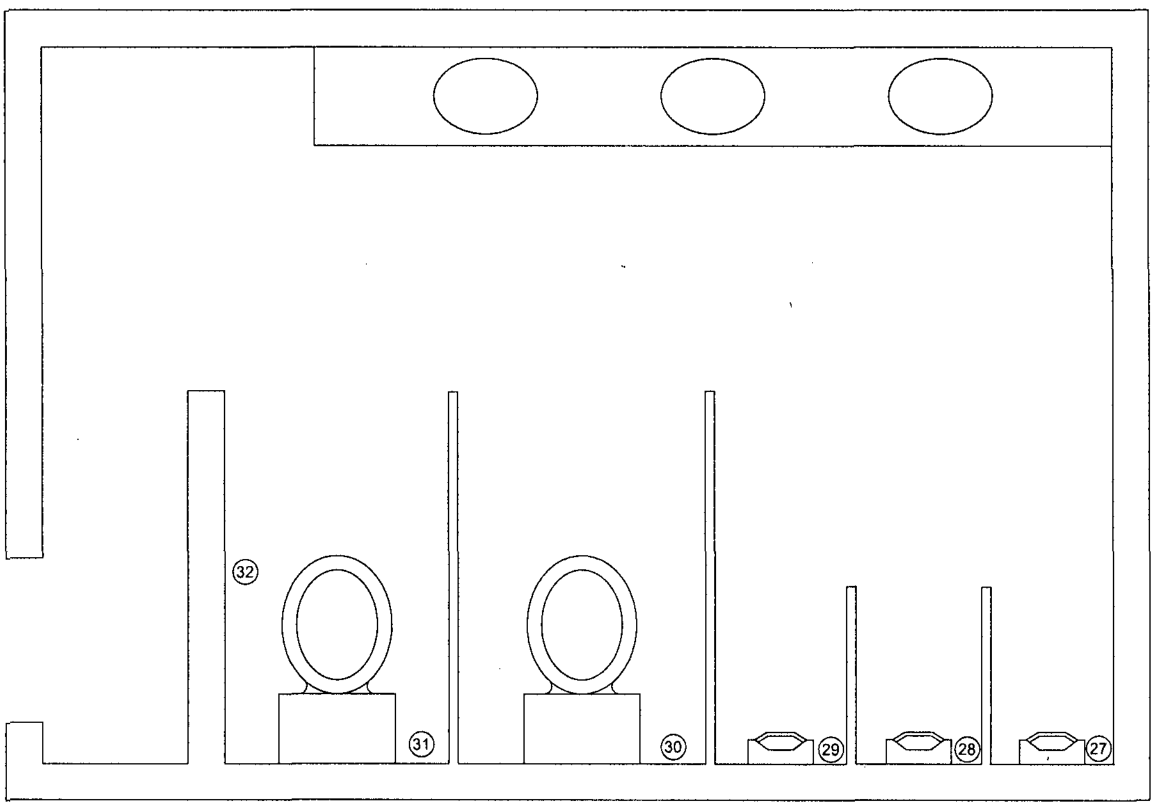
(XX) Approximate location of survey point.



3rd Floor Women's Bathroom
Tile Wall
Background Survey

(XX) Approximate location of survey point.

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP Building Augusta Survey Package B9800	Figure 2-100



4th Floor Men's Bathroom
Tile Wall
Background Survey
04WS1

(XX) Approximate location of survey point.

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Site Characterization CMP Building Augusta Survey Package B9800	Figure 2-101

ATTACHMENT 2F

Analysis of Concrete Sample Variance

Concrete Core Data Variance Analysis

Introduction

A series of concrete core samples were collected and analyzed[1] as described in Engineering Calculation 011-01(MY) to determine the radionuclide mixture to use in the DCGL calculation for contaminated concrete and other contaminated materials. The nuclide mixture determination included an analysis of the data to ensure that the established dose criterion will be satisfied with sufficient confidence when the selected mixture was used. This analysis was performed primarily on the basis of dose.

This attachment describes the process used to evaluate the nuclide mixture for contaminated concrete surfaces and to determine that the mixture is representative and ensures that the established dose criterion will be met with sufficient confidence.

Nuclide Data

The concrete core data used to determine the nuclide mixture was collected during two sampling campaigns. The first data set was comprised of seven cores collected during the site characterization that were representative of concrete contamination in the majority of plant areas. This majority area is called the "balance of plant" (BOP). The first data set was used to determine the nuclide fractions for the BOP, which includes most of the areas in the building basements.

A second data set, consisting of eight samples, was collected to replace cores consumed during analysis processes, to investigate suspect data, and to provide additional information on the nuclide mixture in certain areas that had some potential for containing nuclide mixtures that differed from the BOP. The second data set consisted of two cores from the Containment Outer Annulus (O/A) trench, three cores from within the loops of Containment, and three cores from the Primary Auxiliary Building (PAB). See Tables 1 and 5 for the listing of actual core identification numbers and the associated plant locations. (Location maps are included in Engineering Calculation EC-011-01 (MY).)

Conversion of BOP Concrete Core Analytical Results to Dose

The first step in determining the acceptability of the BOP nuclide mixture was to normalize[2] the nuclide data and convert the normalized data to dose. Dose was used in the evaluation since the unrestricted use criterion is defined in terms of dose and expressing potential uncertainty in terms of dose provides the most direct means of demonstrating acceptability.

There were several steps required to convert the raw radioanalytical core data to dose. First, the nuclide data for each core was decay corrected to 1/1/2004 to correspond to the approximate

[1] Core analyses were performed by Duke Engineering and Services Environmental Laboratory
[2] Normalization, in this case refers to converting the reported nuclide concentration results into nuclide fractions.

time of the last final surveys. The initial and decay corrected data as well as other supporting documents is provided in EC-011-01. Second, the decay corrected nuclide concentration results from each of the cores were converted to fractions. The sum of the nuclide fractions in each core then represent 1.0 dpm/100 cm^2 total activity. Analytical results that were reported as less than the minimum detectable activity (MDA) were assumed to be present at the MDA value in the initial review. The nuclides that were listed as less than MDA in each of the seven cores are indicated in Table 1 by a "<" sign. See Table 1, Column 3 for example of normalized nuclide fractions for the core 1FL1.

The basement fill model (LTP Section 6.6.1) was used to convert the normalized nuclide fractions to dose. Note that there were two other materials, i.e., buried pipe/conduit and embedded pipe, that were assumed to contain the BOP nuclide mixture and each of these materials has a different dose model. However, because the basement concrete contains the overwhelming majority of the contamination inventory and results in the highest dose, the basement fill model was selected for the core dose calculations.

The dose was calculated by multiplying the normalized nuclide fractions by the unitized dose factors determined in Engineering Calculation EC-011-01 (MY). The unitized dose factor is the dose that would result from 1.0 dpm/100 cm^2 activity of a given radionuclide. See Table 1, Column 4 for an example of the dose from the nuclide fractions in the Core 1FL1 mixture.

The sum of the normalized doses from all radionuclides in a given core represents the dose from each core assuming that the core contains a total activity of 1.0 dpm/100 cm^2. The last conversion required to perform the analysis of uncertainty in the radionuclide mixture is to convert the 1.0 dpm/100 cm^2 normalized doses to a dose that represents 18,000 dpm/100 cm^2 detectable beta activity. This is accomplished by dividing each of the nuclides in a given core by the detectable beta fraction of the core and multiplying by 18,000 dpm/100 cm^2. This conversion allows direct comparison with the dose that would result if residual contamination were present in each core at the DCGL concentrations of 18,000 dpm/100 cm^2 observable data. See Table 1, Column 5 for an example of the nuclide dose from core 1FL1 after converting to 18,000 dpm/100 cm^2.

The use of the various dose-converted core data sets in the evaluation of core variability is described in the sections below.

Evaluation of Less than MDA Nuclides

Before the nuclide data variability could be evaluated, the results reported as less than MDA were considered. It was expected that several of the 31 nuclides would be reported as less than MDA since these nuclides have a low probability of being present and were included in the analyses only as a conservative measure. Two approaches were considered for evaluating MDA results; 1) include the MDA values as representing actual concentrations, and 2) remove the non-detected nuclides from the mixture. Removing the nuclides was considered more appropriate and representative of actual site conditions because the non-detected nuclides are believed either to not be present or to be present at concentrations well below the reported MDA value.

However, it cannot be ruled out with 100% certainty that the non-detected nuclides are not present at activities approaching the MDA values. Therefore, an analysis was performed, based on relative dose, to review the affect of leaving the MDA's in the mixture versus removing the MDA's.

To perform this analysis, the dose from the MDA nuclides was compared to the total dose including all nuclides. A nuclide was included in the "MDA" category if it was not detected in any of the cores. If a nuclide was detected in one or more cores, the nuclide was retained and included in the mixture calculation, including MDA values in some instances. For example, Sr-90 was detected above MDA in three of the seven primary cores. For the remaining 4 cores, the MDA value was conservatively assumed to represent detectable activity. As shown in Table 2, the MDA nuclides contributed 1.8% of the total dose (5.1E-03 mrem/y/2.8E-01mrem/y). Since the MDA contribution was low the MDA nuclides were removed from the mixture.

Table 3 contains the nuclide mixture after the MDA nuclides were removed. Note that the nuclide fractions listed in Table 3 was renormalized to 1.0 after removal of the MDA radionuclides that were not detected in any of the cores. This is a conservative yet appropriate approach since the MDA radionuclides were not believed to be present in appreciable quantities.

Evaluation of Variability of Dose from Primary Seven Core Data Set

The variability of the dose from the cores in the primary seven core data set, after removal of MDA's, was evaluated to demonstrate that the variability is low relative to the unrestricted use dose criteria of 10 mrem/yr all pathways and that the seven core data set is sufficiently representative of BOP areas. The variability was evaluated by reviewing the dose from individual cores and the dose from the average of the nuclide fractions. The mean and standard deviation of the dose from both the individual cores and the nuclide fractions were evaluated to determine: 1) if there were a significant difference in the means calculated using the two methods, 2) whether any individual core dose appeared to be significantly different from the mean dose, and 3) whether the variability of the mean dose using the average of the fractions method was sufficiently low relative to the 10 mrem/yr all pathways unrestricted use criteria to provide confidence that the dose criterion would be satisfied using the average of the fractions method.

Calculation of Mean and Standard Deviation

The mean dose and standard deviation of the mean from the individual seven cores were calculated using the data set generated after removal of MDAs and converting to 18,000 dpm measurable gross beta (See Table 4). The calculation of the standard deviation of the mean from the individual cores used the following standard equations:

$$\sqrt{\frac{n\Sigma x^2 - (\Sigma x)^2}{n(n-1)}}$$

then,

$$\frac{\text{Standard Deviation}}{\sqrt{n}}$$

The mean dose and standard deviation of the mean from the nuclide fractions in the seven cores were calculated using the data set generated after removal of MDA's but before converting to 18,000 dpm/100 cm^2 measurable gross beta. Use of this data set was required because the relative nuclide fractions found in the original core analyses need to be retained to correctly calculate the average of the nuclide fractions over the seven cores. After the dose from average nuclide fractions was calculated the result was converted to represent the dose from 18,000 dpm/100 cm^2 measurable gross beta prior to comparison of the two data sets.

The mean dose from the nuclide fractions was calculated by summing the dose from average of each nuclide fraction over the seven cores. The standard deviation of the mean dose from the nuclide fractions required the use of a standard propagation of errors equation to account for the variability within each average nuclide fraction. This was accomplished by squaring the standard deviation of each average nuclide fraction and summing over all nuclides. The propagated error was calculated as:

$$\sqrt{\sum\left(\frac{std.\ dev.}{\sqrt{n}}\right)^2}$$

The first and second data sets were evaluated and compared to ensure that there was not a significant variation between the average dose from the individual cores, which is assumed to represent a given area of the plant, and the average dose as represented by the nuclide fractions in the BOP mixture listed in the LTP.

Data Evaluation

The first evaluation of the individual core and nuclide fractions data sets was performed to demonstrate that there was no significant difference between the means of the two methods for calculating mean dose. It is obvious by a simple comparison of the means and standard deviations provided in Table 4 that there is not a significant difference between the means. The mean dose and standard deviation for the individual cores are 0.29 mrem/yr and 0.030 mrem/yr, respectively. The mean dose and standard deviation using the average of the fractions method are 0.30 mrem/yr and 0.070 mrem/yr, respectively. In fact, the means are essentially identical, differing by less than 0.01 mrem/yr.

The second evaluation entailed a review of the individual core data set to determine if any individual core dose was significantly different from the mean. The standard deviation of the individual core dose, 0.083 mrem/yr, was used for this evaluation. Multiplying the standard deviation by 1.96 and then adding and subtracting the result to the mean results in the upper and lower 95% confidence level bounds. The upper confidence level was 0.46 mrem/yr and the lower confidence level was 0.13 mrem/yr. No individual core dose was outside the 95% confidence levels indicating that no area represented by the core dose was significantly different from the mean. Note that core 02FL51 was at the upper confidence level. This is attributable to an unusually high MDA value for Sr-90 in this core relative to the Sr-90 MDA values reported for the 4 other cores where MDA values were applied and is therefore not significant.

The third evaluation was to review the distribution of the mean from the average of the fractions method. As seen in Table 4, upper 95% confidence level is 0.14 mrem/yr (0.07 mrem/yr times 1.96), which is a very small fraction of the of the 10 mrem/yr dose criterion.

The results of the three evaluations performed above demonstrate: 1) that there is no significant difference between the individual core and average of the fractions methods for calculating mean dose, 2) that no individual core varied significantly from the mean indicating that all of the cores were a part of the same population, and 3) that the variability of the dose using the average of the fractions method is a small fraction of the unrestricted use limit and ensures that the dose criterion will be met with sufficient confidence.

Methods for Evaluating Additional Eight Cores

The discussions and analyses presented above demonstrate that the seven core data set is sufficient to determine the BOP nuclide mixture. The next task was to develop the methods to evaluate the nuclide mixtures in the eight additional cores that were collected during continuing characterization and determine whether they were consistent with the BOP mixture.

If the nuclide mixture of a given core is significantly different from the BOP mixture, then a separate mixture and DCGL may be necessary for the areas represented by the cores. These evaluation criteria would also apply to additional concrete cores collected, if any. Based on evaluation of the 15 cores and a review of the potential for additional plant areas to have a significantly different nuclide mixture than the BOP, no additional cores are deemed necessary to support the LTP.

Three factors were considered in the evaluation of additional cores: 1) whether the core contained detectable transuranics, 2) whether one or more radionuclide fractions are significantly different from the BOP mixture, and 3) whether the dose from an additional core was significantly different from the BOP mixture dose and exceeded 1.0 mrem/yr. The three evaluation factors were developed during the Technical Issue Resolution Process (TIRP) conducted by the State of Maine and Maine Yankee as a part of the Settlement Agreement related to the States motion to terminate their petition to intervene in the matter of MY's proposed LTP. During the TIRP, MY and State technical experts developed and used these three criteria to evaluate additional concrete core samples. Maine Yankee believes the criteria are reasonable and protective and agreed to include the criteria in the LTP. The three criteria for evaluating individual cores are listed below.

1. No detectable TRU.

2. Individual fractions of nuclides:

Nuclide	Maximum Nuclide Fraction
Sr	0.013
Co	0.170
Cs	1.000
Ni	1.000

3. Individual core total dose from all nuclide fractions less than 1.0 mrem/yr.

The first individual core criterion (#1 above) pertained to transuranic (TRU) radionuclides. The TRU's were singled out because their radiological and chemical characteristics differ from the BOP radionuclide mixture, as well as the fact that there is a significant level of stakeholder interest in TRU's. Therefore, the first individual core decision statement was whether or not the core contained TRU's at levels exceeding the minimum detectable activity (MDA). If so, the area represented by the core would either 1) be subject to a unique radionuclide mixture and DCGL or 2) be combined with other TRU cores to generate a single radionuclide mixture and DCGL representing several TRU affected areas.

The second individual core criterion (#2 above) compares the radionuclide fractions in a given core to an upper bound expected given the data provided in the seven-core BOP set. The upper 95% confidence level (UCL) was calculated for each nuclide fraction in the seven core set. The UCL's for four nuclides, Cs-137, Co-60, Sr-90, and Ni-63 are listed above as individual core Criterion #2. Criterion #2 was limited to these four nuclides since they together comprise the overwhelming majority of the dose from concrete basement surfaces. If the nuclide fractions in a given individual core are all less than the values listed in Criterion#2, the BOP radionuclide fraction set is assumed to sufficiently represent the core. However, if an individual core contains a nuclide fraction equal to or exceeding one of the values listed in Criterion #2, then the dose from the core must be calculated and compared to the dose listed in individual core Criterion #3 (#3 above). Criterion # 2 only applies to nuclide fractions that are based upon nuclide activities greater than MDA. If a core's radionuclide fraction, which is based upon a radionuclide activity less than MDA, fails to meet Criterion #2 and the MDA is comparable to the MDA's achieved for the other cores, then it will be considered as having satisfied Criterion #2 for that radionuclide.

The third individual core criterion (#3 above) ensures that the dose potentially represented by an individual core is not significantly different from the seven-core data set. Criterion #3 is required only if Criterion #2 is not satisfied. The dose criterion of 1.0 mrem/yr was selected because at the time of the TIRP original consensus 1.0 mrem/yr was 0.44 mrem/yr above the mean dose (0.556 mrem/yr) calculated using the BOP nuclide mixture. Using the most current dose assessment results, 1 mrem/yr is 0.70 mrem/y above the mean dose of 0.30 mrem/y. The current 0.70 mrem/yr value is more conservative than the 0.44 mrem/yr value found to be acceptable by the TIRP since the actual core variability is a smaller percentage of the acceptable 0.70 mrem/yr variability. A variability of either 0.44 or 0.70 mrem/y above the mean is well below the value that would be acceptable by NRC guidance in NUREG-1727, Page E16, which states that "...the presence of nuclides that likely contribute less than 10% of the total effective dose equivalent may be ignored." The Maine Yankee dose limit is 10 mrem/yr and 10% equals 1.0 mrem/yr. Use of such a 10% criterion is also supported by NRC regulations in 10 CFR 20.1204(g) and 10 CFR 20.1502. Note that the value of 0.70 mrem/yr represents the variability attributable to uncertainty in the nuclide mixture for the individual concrete cores and not an actual dose above the 10 mrem/yr limit. The best estimate of dose is calculated using the mean nuclide fractions of the seven-core data set, i.e., the BOP nuclide mixture.

Results of Evaluation of Eight Additional Cores

The analytical results for the eight additional cores are provided in Table 5. The data was reduced in the same manner as the primary seven core data set. Each radionuclide was decay corrected to 2004; the nuclide fractions were normalized to 1.0; the nuclide fractions were then multiplied by the unitized dose factors based on 1.0 dpm/100 cm^2 (to convert the fractions to dose); and finally, the dose was converted to that which would result from 18,000 dpm/100 cm^2 measurable gross beta. The data for the individual cores was then compared to the three evaluation criteria described above.

Inspection of Table 5 shows that four of the cores clearly meet the 3 evaluation criteria and are considered to be sufficiently represented by the BOP nuclide mixture. These cores were collected from the Containment loops 1, 2, and 3, and the PAB evaporator cubicle and show good agreement with the BOP mixture.

Of the remaining four cores, two of the cores were from the O/A trench and two are from the PAB pipe tunnel. Three of the four cores contained TRU's that were above the MDA. Table 6 contains the radionuclide mixture and dose data for the four TRU affected cores after removal of the MDA results. Table 7 contains the dose summaries for the individual core and average of the fractions methods for the TRU affected cores. The individual core doses range from 0.18 to 0.25 mrem/yr assuming 18,000 dpm/100 cm^2 observable beta. This is a very low fraction of the unrestricted use criteria and is much less than the 1.0 mrem/yr individual core dose criteria used in the BOP nuclide mixture decision rule. As stated previously, the data reduction method for these four cores was conducted in the same manner as for the BOP cores. The dose from the four cores TRU mixture using the average of the fractions method was 0.21 mrem/yr. Based on these results, the four core "average of the fractions" nuclide mixture will be used to determine a separate DCGL for "TRU Affected" areas hereafter referred to as "Special Areas".

After the identification of Special Areas through the analyses of the eight additional cores, a review of building basement areas was performed to determine if there were other areas that could be designated as Special Areas that were not represented by the BOP mixture. The liquid waste stream significantly impacted both the O/A trench and PAB pipe tunnel. The O/A trench captured all water released to the floor of the Containment building and routed it to the Containment sump. The PAB pipe tunnel held the pipes that carried the liquid waste water being processed by the filters and demineralizers in the PAB. Both areas had standing water and boron encrustations during plant operation. As a result of this review one additional area (letdown heat exchanger cubicle) was identified that had operating history and characteristics that were sufficiently similar to the PAB pipe tunnel and Containment O/Annulus Trench to warrant consideration as TRU affected.

The letdown heat exchanger cubicle is an area of approximately 2.5 m by 2.5 m by 3 m tall located in the PAB basement. Because of its small size, it was not specifically sampled. This is the one area that stands out as perhaps needing to be examined since it was processed high temperature liquids and had standing boron. Additional cores samples could have been collected in the letdown heat exchanger cubicle to demonstrate that the cubicle is not TRU affected and that the BOP nuclide mixture would apply. However, the decision was made to conservatively assume the area was TRU affected and to use the "Special Area" nuclide mixture to calculate the DCGL for this area. This decision is conservative since the DCGL for Special Areas is lower than the BOP areas.

Conclusion

The nuclide mixture provided in Table 4 Column 2 using the average of the fractions methods has been demonstrated to be representative of BOP areas and ensures that the established dose criterion will be satisfied with sufficient confidence. Three TRU affected areas have been identified that are represented by a unique nuclide mixture as listed in Table 7 Column 2. Finally a decision rule has been developed through the cooperative efforts of the State of Maine and Maine Yankee (i. e., TIRP). This rule will be used to evaluate the impact and use of any future core information obtained with regard to nuclide mixture and the associated DCGL.

References

1. Maine Yankee License Termination Plan Settlement Agreement, ASLNP No. 00-870-03-0LA, August 29, 2001.
2. Participant Consensus Agreement, State of Maine – Maine Yankee Settlement Agreement, Technical Issue Resolution Process, Dated December 13, 2001.

Table 1
Nuclide Fractions and Dose (mrem/y) for Balance of Plant Core Samples
(Table 1 page 1 of 2)

Column # ==>	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
		PAB 11' West Pipe Trench 1FL1 2004		Dose For 1.80E+04 dpm/100 cm^2 detectable β		Fuel Bldg Decon Room 01FL31 2004		Dose For 1.80E+04 dpm/100 cm^2 detectable β		Spray Bldg 11' 01FL41 2004		Dose For 1.80E+04 dpm/100 cm^2 detectable β		RCA Bldg 21' 01FL61 2004		Dose For 1.80E+04 dpm/100 cm^2 detectable β
Nuclide		**nf**	**1 dpm dose Times nf**	**2004**		**nf**	**1 dpm dose Times nf**	**2004**		**nf**	**1 dpm dose Times nf**	**2004**		**nf**	**1 dpm dose Times nf**	**2004**
H-3		3.997E-04	1.34E-08	9.59E-04		6.265E-04	2.10E-08	4.84E-04		1.211E-03	4.06E-08	4.85E-03		1.025E-03	3.43E-08	7.08E-04
C-14	<	3.433E-06	6.19E-11	4.43E-06	<	1.067E-04	1.92E-09	4.44E-05	<	1.531E-04	2.76E-09	3.30E-04	<	3.590E-05	6.47E-10	1.33E-05
Mn-54	<	1.369E-04	2.88E-10	2.06E-05	<	1.326E-04	2.79E-10	6.43E-06	<	1.533E-04	3.22E-10	3.86E-05	<	1.098E-04	2.31E-10	4.76E-06
Fe-55		2.855E-03	1.67E-09	1.19E-04		4.571E-04	2.67E-10	6.16E-06		3.800E-03	2.22E-09	2.65E-04		2.896E-03	1.69E-09	3.49E-05
Co-57		2.861E-05	7.26E-12	5.20E-07	<	9.019E-05	2.29E-11	5.27E-07	<	9.928E-05	2.52E-11	3.01E-06	<	5.719E-05	1.45E-11	2.99E-07
Co-58	<	5.437E-10	4.40E-16	3.15E-11	<	4.137E-10	3.34E-16	7.71E-12	<	5.137E-10	4.15E-16	4.97E-11	<	3.427E-10	2.77E-16	5.71E-12
Ni-59	<	7.594E-03	9.18E-11	6.57E-06	<	2.152E-03	2.60E-11	6.00E-07	<	8.586E-03	1.04E-10	1.24E-05	<	1.223E-03	1.48E-11	3.04E-07
Co-60		1.539E-01	9.70E-07	6.95E-02		4.057E-03	2.56E-08	5.90E-04		3.125E-02	1.97E-07	2.36E-02		2.692E-02	1.70E-07	3.50E-03
Ni-63		7.357E-01	8.22E-08	5.89E-03		2.085E-01	2.33E-08	5.37E-04		8.318E-01	9.29E-08	1.11E-02		1.184E-01	1.32E-08	2.73E-04
Zn-65	<	1.324E-04	1.40E-09	1.00E-04	<	1.383E-04	1.46E-09	3.37E-05	<	2.214E-04	2.34E-09	2.80E-04	<	8.986E-05	9.50E-10	1.96E-05
Sr-90		1.316E-04	8.30E-09	5.95E-04	<	1.198E-03	7.56E-08	1.74E-03		5.420E-04	3.42E-08	4.09E-03		1.267E-03	8.00E-08	1.65E-03
Nb-94	<	5.978E-03	1.00E-08	7.17E-04	<	3.825E-03	6.40E-09	1.48E-04	<	9.049E-03	1.51E-08	1.81E-03	<	3.867E-03	6.47E-09	1.33E-04
Tc-99	<	1.520E-05	4.89E-09	3.50E-04	<	1.409E-04	4.53E-08	1.04E-03	<	1.547E-05	4.97E-09	5.95E-04	<	1.558E-04	5.01E-08	1.03E-03
Ru-106	<	1.742E-03	2.15E-08	1.54E-03	<	3.616E-03	4.45E-08	1.03E-03	<	3.178E-03	3.91E-08	4.68E-03	<	2.158E-03	2.66E-08	5.48E-04
Ag-110m	<	1.003E-04	3.46E-10	2.48E-05	<	7.361E-05	2.54E-10	5.85E-06	<	1.228E-04	4.23E-10	5.06E-05	<	5.515E-05	1.90E-10	3.92E-06
Sb-125	<	3.596E-03	7.22E-09	5.17E-04	<	1.165E-02	2.34E-08	5.39E-04	<	7.372E-03	1.48E-08	1.77E-03	<	7.898E-03	1.59E-08	3.27E-04
I-129	<	3.032E-08	1.97E-10	1.41E-05	<	2.810E-07	1.82E-09	4.20E-05	<	3.084E-08	2.00E-10	2.39E-05	<	3.107E-07	2.02E-09	4.15E-05
Cs-134		1.720E-03	4.10E-08	2.94E-03	<	1.264E-03	3.01E-08	6.95E-04	<	2.039E-03	4.86E-08	5.81E-03		1.835E-03	4.38E-08	9.01E-04
Cs-137		8.049E-02	1.29E-06	9.25E-02		7.461E-01	1.20E-05	2.76E-01		8.188E-02	1.31E-06	1.57E-01		8.250E-01	1.32E-05	2.73E-01
Ce-144	<	3.222E-04	3.34E-10	2.39E-05	<	9.153E-04	9.49E-10	2.19E-05	<	9.014E-04	9.34E-10	1.12E-04	<	5.599E-04	5.80E-10	1.20E-05
Pm-147	<	5.086E-06	1.03E-11	7.39E-07	<	2.762E-04	5.60E-10	1.29E-05	<	2.629E-05	5.33E-11	6.38E-06	<	2.761E-05	5.60E-11	1.15E-06
Eu-154	<	2.749E-03	2.11E-09	1.51E-04	<	3.889E-03	2.99E-09	6.89E-05	<	9.220E-03	7.09E-09	8.48E-04	<	1.685E-03	1.30E-09	2.67E-05
Eu-155	<	2.347E-03	2.31E-10	1.65E-05	<	7.394E-03	7.27E-10	1.68E-05	<	7.950E-03	7.81E-10	9.34E-05	<	4.368E-03	4.29E-10	8.85E-06
Pu-238	<	8.588E-07	1.20E-10	8.56E-06	<	1.537E-05	2.14E-09	4.93E-05	<	1.768E-05	2.46E-09	2.94E-04	<	5.462E-06	7.60E-10	1.57E-05
Pu-239	<	4.261E-07	6.56E-11	4.70E-06	<	5.260E-06	8.10E-10	1.87E-05	<	7.773E-06	1.20E-09	1.43E-04	<	2.728E-06	4.20E-10	8.65E-06
Pu-240	<	4.259E-07	6.56E-11	4.70E-06	<	5.258E-06	8.10E-10	1.87E-05	<	7.771E-06	1.20E-09	1.43E-04	<	2.727E-06	4.20E-10	8.65E-06
Pu-241	<	6.122E-05	1.83E-10	1.31E-05	<	3.324E-03	9.92E-09	2.29E-04	<	3.164E-04	9.44E-10	1.13E-04	<	3.324E-04	9.92E-10	2.04E-05
Am-241	<	1.768E-06	8.24E-11	5.90E-06	<	3.481E-05	1.62E-09	3.74E-05	<	2.304E-05	1.07E-09	1.28E-04	<	8.914E-06	4.15E-10	8.55E-06
Cm-242	<	6.341E-10	4.44E-16	3.18E-11	<	1.225E-08	8.58E-15	1.98E-10	<	8.724E-09	6.11E-15	7.31E-10	<	3.500E-09	2.45E-15	5.05E-11
Cm-243	<	3.641E-07	5.57E-12	3.99E-07	<	5.775E-06	8.84E-11	2.04E-06	<	4.878E-06	7.47E-11	8.93E-06	<	1.823E-06	2.79E-11	5.75E-07
Cm-244	<	3.416E-07	4.21E-12	3.01E-07	<	5.418E-06	6.67E-11	1.54E-06	<	4.576E-06	5.64E-11	6.74E-06	<	1.710E-06	2.11E-11	4.34E-07
sum		1.000E+00				1.000E+00				1.000E+00				1.000E+00		
obs. β fraction		2.513E-01				7.809E-01				1.505E-01				8.736E-01		

Table 1
Nuclide Fractions and Dose (mrem/y) for Balance of Plant Core Samples
(Table 1 page 2 of 2)

Column # =>	18	19	20	21	22	23	24	25	26	27	28	29	33
Nuclide		**PAB 11' Pipe Trench 01FL81 2004 nf**	**1 dpm dose Times nf**	**Dose For 1.80E+04 dpm/100 cm^2 detectable β 2004**		**CTMT -2' Loop 2 02FL21 2004 nf**	**1 dpm dose Times nf**	**Dose For 1.80E+04 dpm/100 cm^2 detectable β 2004**		**CTMT -2' Loop 1 02FL51 2004 nf**	**1 dpm dose Times nf**	**Dose For 1.80E+04 dpm/100 cm^2 detectable β 2004**	**1.0 dpm dose factor**
H-3		3.583E-03	1.20E-07	2.65E-03		4.279E-03	1.43E-07	2.65E-03		1.214E-01	4.07E-06	1.26E-01	3.351E-05
C-14	<	9.586E-05	1.73E-09	3.82E-05	<	1.329E-04	2.40E-09	4.42E-05	<	2.419E-03	4.36E-08	1.35E-03	1.803E-05
Mn-54	<	1.198E-04	2.52E-10	5.56E-06	<	2.051E-04	4.32E-10	7.97E-06	<	1.356E-03	2.85E-09	8.85E-05	2.104E-06
Fe-55		2.372E-03	1.39E-09	3.06E-05	<	5.234E-04	3.06E-10	5.65E-06		1.646E-02	9.61E-09	2.98E-04	5.843E-07
Co-57	<	6.477E-05	1.64E-11	3.63E-07	<	1.157E-04	2.94E-11	5.42E-07	<	1.409E-03	3.57E-10	1.11E-05	2.537E-07
Co-58	<	4.361E-10	3.53E-16	7.78E-12	<	6.556E-10	5.30E-16	9.79E-12	<	1.135E-08	9.18E-15	2.85E-10	8.086E-07
Ni-59	<	1.792E-03	2.17E-11	4.78E-07	<	1.593E-04	1.93E-12	3.56E-08	<	2.648E-03	3.20E-11	9.94E-07	1.209E-08
Co-60		5.049E-02	3.18E-07	7.03E-03		5.989E-03	3.78E-08	6.97E-04		1.034E-01	6.52E-07	2.02E-02	6.305E-06
Ni-63		1.736E-01	1.94E-08	4.28E-04		1.543E-02	1.72E-09	3.18E-05		2.565E-01	2.87E-08	8.90E-04	1.117E-07
Zn-65	<	1.469E-04	1.55E-09	3.43E-05	<	1.508E-04	1.59E-09	2.94E-05	<	3.794E-03	4.01E-08	1.25E-03	1.058E-05
Sr-90	<	4.383E-04	2.77E-08	6.10E-04	<	3.843E-04	2.42E-08	4.48E-04	<	1.227E-02	7.74E-07	2.40E-02	6.310E-05
Nb-94	<	5.305E-03	8.88E-09	1.96E-04	<	4.703E-03	7.87E-09	1.45E-04	<	5.249E-02	8.79E-08	2.73E-03	1.674E-06
Tc-99	<	1.398E-04	4.50E-08	9.93E-04	<	1.763E-04	5.67E-08	1.05E-03	<	4.958E-05	1.59E-08	4.95E-04	3.216E-04
Ru-106	<	2.533E-03	3.12E-08	6.89E-04	<	3.326E-03	4.10E-08	7.56E-04	<	4.993E-03	6.15E-08	1.91E-03	1.232E-05
Ag-110m	<	9.030E-05	3.11E-10	6.88E-06	<	7.082E-05	2.44E-10	4.51E-06	<	1.319E-03	4.55E-09	1.41E-04	3.449E-06
Sb-125	<	7.779E-03	1.56E-08	3.45E-04	<	1.364E-02	2.74E-08	5.06E-04	<	8.338E-02	1.67E-07	5.19E-03	2.007E-06
I-129	<	2.788E-07	1.81E-09	3.99E-05	<	3.515E-07	2.28E-09	4.21E-05	<	9.887E-08	6.42E-10	1.99E-05	6.489E-03
Cs-134		1.552E-03	3.70E-08	8.17E-04		1.534E-03	3.66E-08	6.75E-04	<	1.750E-02	4.17E-07	1.29E-02	2.384E-05
Cs-137		7.402E-01	1.19E-05	2.62E-01		9.332E-01	1.50E-05	2.77E-01		2.625E-01	4.21E-06	1.31E-01	1.605E-05
Ce-144	<	6.548E-04	6.79E-10	1.50E-05	<	1.209E-03	1.25E-09	2.31E-05	<	1.379E-02	1.43E-08	4.44E-04	1.036E-06
Pm-147	<	2.584E-05	5.24E-11	1.16E-06	<	8.857E-05	1.80E-10	3.32E-06	<	8.550E-04	1.73E-09	5.38E-05	2.028E-06
Eu-154	<	3.556E-03	2.73E-09	6.04E-05	<	4.704E-03	3.62E-09	6.68E-05	<	2.092E-02	1.61E-08	4.99E-04	7.690E-07
Eu-155	<	5.055E-03	4.97E-10	1.10E-05	<	8.838E-03	8.69E-10	1.60E-05	<	8.969E-03	8.82E-10	2.74E-05	9.828E-08
Pu-238	<	1.476E-05	2.05E-09	4.54E-05	<	1.045E-05	1.45E-09	2.69E-05	<	2.307E-04	3.21E-08	9.96E-04	1.392E-04
Pu-239	<	7.040E-06	1.08E-09	2.39E-05	<	4.601E-06	7.08E-10	1.31E-05	<	8.748E-05	1.35E-08	4.18E-04	1.540E-04
Pu-240	<	7.037E-06	1.08E-09	2.39E-05	<	4.599E-06	7.08E-10	1.31E-05	<	8.745E-05	1.35E-08	4.18E-04	1.540E-04
Pu-241	<	3.110E-04	9.28E-10	2.05E-05	<	1.066E-03	3.18E-09	5.88E-05	<	1.029E-02	3.07E-08	9.54E-04	2.985E-06
Am-241	<	3.241E-05	1.51E-09	3.33E-05	<	2.233E-05	1.04E-09	1.92E-05	<	5.790E-04	2.70E-08	8.37E-04	4.658E-05
Cm-242	<	1.318E-08	9.23E-15	2.04E-10	<	6.351E-09	4.45E-15	8.21E-11	<	2.622E-07	1.84E-13	5.70E-09	7.002E-07
Cm-243	<	6.663E-06	1.02E-10	2.25E-06	<	3.750E-06	5.74E-11	1.06E-06	<	1.304E-04	2.00E-09	6.20E-05	1.531E-05
Cm-244	<	6.250E-06	7.70E-11	1.70E-06	<	3.518E-06	4.33E-11	8.00E-07	<	1.224E-04	1.51E-09	4.68E-05	1.232E-05
sum		1.000E+00				1.000E+00				1.000E+00			
obs. β fraction		8.154E-01				9.747E-01				5.799E-01			
avg. β fraction		6.324E-01											

Table 2
Calculation of Dose From MDA Nuclides

Column # ==>	2	3	4	5	6	7	8	9	10	11
	Dose Results (in mrem/y) for Average of the Fractions (1.0 dpm)									
Nuclide	**1FL1**	**01FL31**	**01FL41**	**01FL61**	**01FL81**	**02FL21**	**02FL51**	**mean***	**stdev mean**	**(stdev Mean)^2**
H-3	1.34E-08	2.10E-08	4.06E-08	3.43E-08	1.20E-07	1.43E-07	4.07E-06	1.81E-02	1.629E-02	2.65E-04
C-14	6.19E-11	1.92E-09	2.76E-09	6.47E-10	1.73E-09	2.40E-09	4.36E-08	2.16E-04	1.712E-04	2.93E-08
Mn-54	2.88E-10	2.79E-10	3.22E-10	2.31E-10	2.52E-10	4.32E-10	2.85E-09	1.89E-05	1.040E-05	1.08E-10
Fe-55	1.67E-09	2.67E-10	2.22E-09	1.69E-09	1.39E-09	3.06E-10	9.61E-09	6.98E-05	3.488E-05	1.22E-09
Co-57	7.26E-12	2.29E-11	2.52E-11	1.45E-11	1.64E-11	2.94E-11	3.57E-10	1.92E-06	1.377E-06	1.90E-12
Co-58	4.40E-16	3.34E-16	4.15E-16	2.77E-16	3.53E-16	5.30E-16	9.18E-15	4.69E-11	3.574E-11	1.28E-21
Ni-59	9.18E-11	2.60E-11	1.04E-10	1.48E-11	2.17E-11	1.93E-12	3.20E-11	1.19E-06	4.261E-07	1.82E-13
Co-60	9.70E-07	2.56E-08	1.97E-07	1.70E-07	3.18E-07	3.78E-08	6.52E-07	9.64E-03	3.765E-03	1.42E-05
Ni-63	8.22E-08	2.33E-08	9.29E-08	1.32E-08	1.94E-08	1.72E-09	2.87E-08	1.06E-03	3.816E-04	1.46E-07
Zn-65	1.40E-09	1.46E-09	2.34E-09	9.50E-10	1.55E-09	1.59E-09	4.01E-08	2.01E-04	1.569E-04	2.46E-08
Sr-90	8.31E-09	7.56E-08	3.42E-08	8.00E-08	2.77E-08	2.42E-08	7.74E-07	4.16E-03	2.992E-03	8.95E-06
Nb-94	1.00E-08	6.40E-09	1.51E-08	6.47E-09	8.88E-09	7.87E-09	8.79E-08	5.80E-04	3.217E-04	1.04E-07
Tc-99	4.89E-09	4.53E-08	4.97E-09	5.01E-08	4.50E-08	5.67E-08	1.59E-08	9.06E-04	2.408E-04	5.80E-08
Ru-106	2.15E-08	4.45E-08	3.91E-08	2.66E-08	3.12E-08	4.10E-08	6.15E-08	1.08E-03	1.427E-04	2.03E-08
Ag-110m	3.46E-10	2.54E-10	4.23E-10	1.90E-10	3.11E-10	2.44E-10	4.55E-09	2.57E-05	1.731E-05	3.00E-10
Sb-125	7.22E-09	2.34E-08	1.48E-08	1.59E-08	1.56E-08	2.74E-08	1.67E-07	1.10E-03	6.138E-04	3.77E-07
I-129	1.97E-10	1.82E-09	2.00E-10	2.02E-09	1.81E-09	2.28E-09	6.42E-10	3.65E-05	9.688E-06	9.39E-11
Cs-134	4.10E-08	3.01E-08	4.86E-08	4.38E-08	3.70E-08	3.66E-08	4.17E-07	2.66E-03	1.537E-03	2.36E-06
Cs-137	1.29E-06	1.20E-05	1.31E-06	1.32E-05	1.19E-05	1.50E-05	4.21E-06	2.40E-01	6.364E-02	4.05E-03
Ce-144	3.34E-10	9.49E-10	9.34E-10	5.80E-10	6.79E-10	1.25E-09	1.43E-08	7.73E-05	5.500E-05	3.03E-09
Pm-147	1.03E-11	5.60E-10	5.33E-11	5.60E-11	5.24E-11	1.80E-10	1.73E-09	1.08E-05	6.750E-06	4.56E-11
Eu-154	2.11E-09	2.99E-09	7.09E-09	1.30E-09	2.73E-09	3.62E-09	1.61E-08	1.46E-04	5.560E-05	3.09E-09
Eu-155	2.31E-10	7.27E-10	7.81E-10	4.29E-10	4.97E-10	8.69E-10	8.82E-10	1.80E-05	2.671E-06	7.14E-12
Pu-238	1.20E-10	2.14E-09	2.46E-09	7.60E-10	2.05E-09	1.45E-09	3.21E-08	1.67E-04	1.248E-04	1.56E-08
Pu-239	6.56E-11	8.10E-10	1.20E-09	4.20E-10	1.08E-09	7.08E-10	1.35E-08	7.22E-05	5.203E-05	2.71E-09
Pu-240	6.56E-11	8.10E-10	1.20E-09	4.20E-10	1.08E-09	7.08E-10	1.35E-08	7.22E-05	5.201E-05	2.71E-09
Pu-241	1.83E-10	9.92E-09	9.44E-10	9.92E-10	9.28E-10	3.18E-09	3.07E-08	1.91E-04	1.196E-04	1.43E-08
Am-241	8.24E-11	1.62E-09	1.07E-09	4.15E-10	1.51E-09	1.04E-09	2.70E-08	1.33E-04	1.059E-04	1.12E-08
Cm-242	4.44E-16	8.58E-15	6.11E-15	2.45E-15	9.23E-15	4.45E-15	1.84E-13	8.74E-10	7.261E-10	5.27E-19
Cm-243	5.58E-12	8.84E-11	7.47E-11	2.79E-11	1.02E-10	5.74E-11	2.00E-09	9.57E-06	7.887E-06	6.22E-11
Cm-244	4.21E-12	6.67E-11	5.64E-11	2.11E-11	7.70E-11	4.33E-11	1.51E-09	7.22E-06	5.953E-06	3.54E-11

Mean
2.80E-01
Standard Deviation of the Mean
6.59E-02

* 1 dpm average value times 18,000/0.6324 (obs average beta fraction) = 2.846E+04

Mean Dose from non-detectable nuclides: 5.07E-03

Mean Dose from detectable **(Bolded)** nuclides: 2.75E-01

Table 3
Nuclide Fractions and Dose After Removal of MDA Nuclides

Column # ==>	2	3	4	5	6	7	8	9	10	11	12	13	23
	PAB 11' West		**Dose For**	**Fuel Bldg**		**Dose For**			**Dose For**			**Dose For**	
	Pipe Trench		1.80E+04	**Decon Room**		1.80E+04	**Spray Bldg 11'**		1.80E+04	**RCA Bldg 21'**		1.80E+04	
	1FL1		**dpm/100 cm^2**	**01FL31**		**dpm/100 cm^2**	**01FL41**		**dpm/100 cm^2**	**01FL61**		**dpm/100 cm^2**	
	2004	**1 dpm dose**	**detectable β**	**2004**	**1 dpm dose**	**detectable β**	**2004**	**1 dpm dose**	**detectable β**	**2004**	**1 dpm dose**	**detectable β**	**1.0 dpm**
Nuclide	**nf**	**Times nf**	**2004**	**nf**	**Times nf**	**2004**	**nf**	**Times nf**	**2004**	**nf**	**Times nf**	**2004**	**dose factor**
H-3	4.099E-04	1.373E-08	1.021E-03	6.510E-04	2.181E-08	5.020E-04	1.271E-03	4.260E-08	6.313E-03	1.049E-03	3.514E-08	7.230E-04	3.351E-05
Fe-55	2.928E-03	1.711E-09	1.271E-04	4.750E-04	2.775E-10	6.387E-06	3.988E-03	2.330E-09	3.453E-04	2.963E-03	1.731E-09	3.563E-05	5.843E-07
Co-57	2.934E-05	7.443E-12	5.531E-07	9.373E-05	2.378E-11	5.472E-07	1.042E-04	2.644E-11	3.918E-06	5.851E-05	1.484E-11	3.054E-07	2.537E-07
Co-60	1.578E-01	9.948E-07	7.393E-02	4.216E-03	2.658E-08	6.118E-04	3.280E-02	2.068E-07	3.065E-02	2.754E-02	1.736E-07	3.573E-03	6.305E-06
Ni-63	7.544E-01	8.428E-08	6.263E-03	2.167E-01	2.420E-08	5.571E-04	8.732E-01	9.755E-08	1.446E-02	1.212E-01	1.354E-08	2.786E-04	1.117E-07
Sr-90	1.349E-04	8.514E-09	6.327E-04	1.245E-03	7.857E-08	1.808E-03	5.690E-04	3.590E-08	5.320E-03	1.296E-03	8.180E-08	1.683E-03	6.310E-05
Cs-134	1.763E-03	4.205E-08	3.125E-03	1.313E-03	3.131E-08	7.206E-04	2.140E-03	5.102E-08	7.561E-03	1.877E-03	4.476E-08	9.211E-04	2.384E-05
Cs-137	8.254E-02	1.325E-06	9.847E-02	7.753E-01	1.245E-05	2.865E-01	8.595E-02	1.380E-06	2.045E-01	8.440E-01	1.355E-05	2.788E-01	1.605E-05
sum	1.000E+00			1.000E+00			1.000E+00			1.000E+00			
obs. β fraction	2.422E-01			7.821E-01			1.215E-01			8.748E-01			
avg. β fraction	5.051E-01												

Nuclide Fractions and Dose After Removal of MDA Nuclides (Continued from above)

Column # ==>	14	15	16	17	18	19	20	21	22	23
	PAB 11'		**Dose For**	**CTMT**		**Dose For**	**CTMT**		**Dose For**	
	Pipe Trench		1.80E+04	**-2' Loop 2**		1.80E+04	**-2' Loop 1**		1.80E+04	
	01FL81		**dpm/100 cm^2**	**02FL21**		**dpm/100 cm^2**	**02FL51**		**dpm/100 cm^2**	
	2004	**1 dpm dose**	**detectable β**	**2004**	**1 dpm dose**	**detectable β**	**2004**	**1 dpm dose**	**detectable β**	**1.0 dpm**
Nuclide	**nf**	**Times nf**	**2004**	**nf**	**Times nf**	**2004**	**nf**	**Times nf**	**2004**	**dose factor**
H-3	3.685E-03	1.235E-07	2.726E-03	4.450E-03	1.491E-07	2.742E-03	1.533E-01	5.138E-06	1.850E-01	3.351E-05
Fe-55	2.440E-03	1.425E-09	3.147E-05	5.443E-04	3.180E-10	5.848E-06	2.079E-02	1.215E-08	4.374E-04	5.843E-07
Co-57	6.662E-05	1.690E-11	3.731E-07	1.203E-04	3.053E-11	5.614E-07	1.780E-03	4.515E-10	1.625E-05	2.537E-07
Co-60	5.193E-02	3.274E-07	7.228E-03	6.228E-03	3.927E-08	7.221E-04	1.307E-01	8.240E-07	2.967E-02	6.305E-06
Ni-63	1.785E-01	1.994E-08	4.403E-04	1.605E-02	1.793E-09	3.298E-05	3.241E-01	3.621E-08	1.304E-03	1.117E-07
Sr-90	4.508E-04	2.844E-08	6.280E-04	3.997E-04	2.522E-08	4.638E-04	1.550E-02	9.779E-07	3.521E-02	6.310E-05
Cs-134	1.596E-03	3.806E-08	8.404E-04	1.595E-03	3.803E-08	6.993E-04	2.211E-02	5.271E-07	1.898E-02	2.384E-05
Cs-137	7.613E-01	1.222E-05	2.699E-01	9.706E-01	1.558E-05	2.866E-01	3.317E-01	5.325E-06	1.917E-01	1.605E-05
sum	1.000E+00			1.000E+00			1.000E+00			
obs. β fraction	8.153E-01			9.788E-01			5.000E-01			

Table 4
Mean and Standard Deviation of Dose
Using the Average of Fractions and Individual Core Methods

Column # ==>	1	2	3	4	5	6	7	8	9	10	11
	7 Cores	**Dose Results (in mrem) for Average of the Fractions (1.0 dpm)**									
Nuclide	**Mean nf**	**1FL1**	**01FL31**	**01FL41**	**01FL61**	**01FL81**	**02FL21**	**02FL51**	**mean***	**stdev mean**	**(stdev Mean)^2**
H-3	2.36E-02	1.37E-08	2.18E-08	4.26E-08	3.51E-08	1.23E-07	1.49E-07	5.14E-06	2.30E-02	2.117E-02	4.48E-04
Fe-55	4.81E-03	1.71E-09	2.78E-10	2.33E-09	1.73E-09	1.43E-09	3.18E-10	1.21E-08	8.32E-05	4.602E-05	2.12E-09
Co-57	3.06E-04	7.44E-12	2.38E-11	2.64E-11	1.48E-11	1.69E-11	3.05E-11	4.51E-10	2.38E-06	1.802E-06	3.25E-12
Co-60	5.84E-02	9.95E-07	2.66E-08	2.07E-07	1.74E-07	3.27E-07	3.93E-08	8.24E-07	1.08E-02	4.253E-03	1.81E-05
Ni-63	3.55E-01	8.43E-08	2.42E-08	9.76E-08	1.35E-08	1.99E-08	1.79E-09	3.62E-08	1.16E-03	4.055E-04	1.64E-07
Sr-90	2.80E-03	8.51E-09	7.86E-08	3.59E-08	8.18E-08	2.84E-08	2.52E-08	9.78E-07	5.16E-03	3.912E-03	1.53E-05
Cs-134	4.56E-03	4.20E-08	3.13E-08	5.10E-08	4.48E-08	3.81E-08	3.80E-08	5.27E-07	3.22E-03	2.029E-03	4.12E-06
Cs-137	5.50E-01	1.33E-06	1.24E-05	1.38E-06	1.36E-05	1.22E-05	1.56E-05	5.32E-06	2.58E-01	6.632E-02	4.40E-03

* 1 dpm average value times 18,000/0.6164 (obs average beta fraction) = 2.920E+04

Mean
3.01E-01
Standard Deviation Of the Mean
6.99E-02

7 Core Dose Results From Individual Cores 18,000 dpm/100 cm^2 Observable Beta

	01FL1	**01FL31**	**01FL41**	**01FL61**	**01FL81**	**02FL21**	**02FL51**
Nuclide	**2004**	**2004**	**2004**	**2004**	**2004**	**2004**	**2004**
H-3	1.02E-03	5.02E-04	6.31E-03	7.23E-04	2.73E-03	2.74E-03	1.85E-01
Fe-55	1.27E-04	6.39E-06	3.45E-04	3.56E-05	3.15E-05	5.85E-06	4.37E-04
Co-57	5.53E-07	5.47E-07	3.92E-06	3.05E-07	3.73E-07	5.61E-07	1.63E-05
Co-60	7.39E-02	6.12E-04	3.06E-02	3.57E-03	7.23E-03	7.22E-04	2.97E-02
Ni-63	6.26E-03	5.57E-04	1.45E-02	2.79E-04	4.40E-04	3.30E-05	1.30E-03
Sr-90	6.33E-04	1.81E-03	5.32E-03	1.68E-03	6.28E-04	4.64E-04	3.52E-02
Cs-134	3.12E-03	7.21E-04	7.56E-03	9.21E-04	8.40E-04	6.99E-04	1.90E-02
Cs-137	9.85E-02	2.86E-01	2.05E-01	2.79E-01	2.70E-01	2.87E-01	1.92E-01
dose sum	1.84E-01	2.91E-01	2.69E-01	2.86E-01	2.82E-01	2.91E-01	4.62E-01

Individual Core (7) Propagation of Error
Mean
2.95E-01
Standard Deviation of the Mean
3.14E-02

Table 5
Nuclide Fractions and Dose From Eight Additional Cores
(Table 5 page1 of 2)

Column # ==>	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
Nuclide		**Containment Loop 2 CA9900 12-C003-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm^2 detectable β 2004**		**Containment Loop 1 CA9900 12-C004-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm^2 detectable β 2.00E+03**		**Containment Loop 3 CA9900 12-C005-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm^2 detectable β 2004**		**PAB Evaporator Cubicle CA9900 13-C001-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm^2 detectable β 2004**
H-3		**6.299E-03**	2.11E-07	4.62E-03		**3.601E-02**	1.21E-06	2.58E-02		**2.101E-03**	7.038E-08	1.60E-03		**7.211E-02**	2.42E-06	5.54E-02
C-14	<	3.284E-04	5.92E-09	1.30E-04	<	8.893E-04	1.60E-08	3.42E-04	<	5.168E-05	9.318E-10	2.11E-05	<	3.268E-02	5.89E-07	1.35E-02
Mn-54		**1.329E-05**	2.80E-11	6.12E-07		**2.928E-05**	6.16E-11	1.32E-06	<	5.172E-05	1.088E-10	2.47E-06	<	2.222E-04	4.67E-10	1.07E-05
Fe-55		**3.593E-02**	2.10E-08	4.60E-04		**8.846E-03**	5.17E-09	1.10E-04		**1.188E-03**	6.941E-10	1.58E-05		**8.482E-03**	4.96E-09	1.14E-04
Co-57	<	**2.762E-05**	7.01E-12	1.53E-07	<	4.709E-05	1.19E-11	2.55E-07	<	2.070E-04	5.250E-11	1.19E-06	<	7.788E-04	1.98E-10	4.53E-06
Co-58	<	3.161E-09	2.56E-15	5.59E-11	<	3.862E-09	3.12E-15	6.67E-11	<	2.284E-08	1.847E-14	4.19E-10	<	8.250E-08	6.67E-14	1.53E-09
Ni-59	<	1.359E-03	1.64E-11	3.60E-07	<	1.122E-03	1.36E-11	2.90E-07	<	2.046E-03	2.473E-11	5.61E-07	<	1.234E-03	1.49E-11	3.42E-07
Co-60		**8.522E-02**	5.37E-07	1.18E-02		**4.633E-02**	2.92E-07	6.24E-03		**4.069E-02**	2.566E-07	5.82E-03		**8.315E-02**	5.24E-07	1.20E-02
Ni-63		**1.330E-01**	1.49E-08	3.25E-04		**1.098E-01**	1.23E-08	2.62E-04		**2.002E-01**	2.237E-08	5.08E-04		**1.208E-01**	1.35E-08	3.10E-04
Zn-65	<	6.124E-06	6.48E-11	1.42E-06	<	1.277E-05	1.35E-10	2.88E-06	<	6.208E-05	6.566E-10	1.49E-05	<	2.748E-04	2.91E-09	6.67E-05
Sr-90		**9.839E-04**	6.21E-08	1.36E-03		**1.330E-03**	8.39E-08	1.79E-03		**1.131E-03**	7.135E-08	1.62E-03	<	4.323E-03	2.73E-07	6.26E-03
Nb-94	<	5.375E-05	9.00E-11	1.97E-06	<	6.424E-05	1.08E-10	2.30E-06	<	4.061E-04	6.798E-10	1.54E-05	<	1.471E-03	2.46E-09	5.65E-05
Tc-99	<	1.338E-04	4.30E-08	9.42E-04	<	1.443E-04	4.64E-08	9.91E-04	<	1.363E-04	4.382E-08	9.94E-04	<	1.177E-04	3.79E-08	8.68E-04
Ru-106	<	9.500E-05	1.17E-09	2.56E-05	<	1.483E-04	1.83E-09	3.90E-05	<	7.091E-04	8.733E-09	1.98E-04	<	2.747E-03	3.38E-08	7.76E-04
Ag-110m	<	1.072E-04	3.70E-10	8.09E-06	<	1.847E-04	6.37E-10	1.36E-05	<	4.383E-05	1.512E-10	3.43E-06	<	2.733E-03	9.43E-09	2.16E-04
Sb-125		**2.306E-04**	4.63E-10	1.01E-05	<	2.742E-04	5.50E-10	1.18E-05	<	1.229E-03	2.467E-09	5.60E-05	<	4.156E-03	8.34E-09	1.91E-04
I-129	<	2.673E-07	1.73E-09	3.80E-05	<	2.876E-07	1.87E-09	3.99E-05	<	2.718E-07	1.763E-09	4.00E-05	<	2.354E-07	1.53E-09	3.50E-05
Cs-134		**1.658E-03**	3.95E-08	8.65E-04		**2.244E-03**	5.35E-08	1.14E-03		**1.196E-03**	2.851E-08	6.47E-04	<	8.118E-04	1.94E-08	4.44E-04
Cs-137		**7.332E-01**	1.18E-05	2.58E-01		**7.910E-01**	1.27E-05	2.71E-01		**7.465E-01**	1.198E-05	2.72E-01		**6.465E-01**	1.04E-05	2.38E-01
Ce-144	<	3.068E-05	3.18E-11	6.96E-07	<	5.077E-05	5.26E-11	1.12E-06	<	2.371E-04	2.457E-10	5.58E-06	<	8.396E-04	8.70E-10	2.00E-05
Pm-147	<	9.913E-05	2.01E-10	4.40E-06	<	6.856E-05	1.39E-10	2.97E-06	<	1.939E-05	3.933E-11	8.92E-07	<	7.612E-04	1.54E-09	3.54E-05
Eu-154	<	1.007E-04	7.75E-11	1.70E-06	<	1.202E-04	9.24E-11	1.97E-06	<	7.820E-04	6.014E-10	1.36E-05	<	3.462E-03	2.66E-09	6.11E-05
Eu-155	<	1.068E-04	1.05E-11	2.30E-07	<	1.767E-04	1.74E-11	3.71E-07	<	7.920E-04	7.784E-11	1.77E-06	<	3.559E-03	3.50E-10	8.03E-06
Pu-238	<	1.049E-05	1.46E-09	3.20E-05	<	1.013E-05	1.41E-09	3.01E-05	<	1.550E-06	2.157E-10	4.89E-06	<	1.994E-04	2.78E-08	6.37E-04
Pu-239	<	6.719E-06	1.03E-09	2.26E-05	<	4.201E-06	6.47E-10	1.38E-05	<	1.052E-06	1.620E-10	3.68E-06	<	8.581E-05	1.32E-08	3.03E-04
Pu-240	<	6.718E-06	1.03E-09	2.26E-05	<	4.200E-06	6.47E-10	1.38E-05	<	1.052E-06	1.620E-10	3.68E-06	<	8.579E-05	1.32E-08	3.03E-04
Pu-241	<	8.625E-04	2.57E-09	5.64E-05	<	5.965E-04	1.78E-09	3.80E-05	<	1.687E-04	5.036E-10	1.14E-05	<	6.623E-03	1.98E-08	4.54E-04
Am-241	<	9.807E-05	4.57E-09	1.00E-04	<	3.344E-04	1.56E-08	3.33E-04	<	2.645E-05	1.232E-09	2.80E-05	<	1.383E-03	6.44E-08	1.48E-03
Cm-242	<	5.750E-07	4.03E-13	8.81E-09	<	1.944E-06	1.36E-12	2.91E-08	<	1.638E-07	1.147E-13	2.60E-09	<	8.893E-06	6.23E-12	1.43E-07
Cm-243	<	1.801E-05	2.76E-10	6.04E-06	<	6.190E-05	9.48E-10	2.02E-05	<	4.282E-06	6.555E-11	1.49E-06	<	2.123E-04	3.25E-09	7.46E-05
Cm-244	<	1.725E-05	2.13E-10	4.65E-06	<	5.929E-05	7.30E-10	1.56E-05	<	4.102E-06	5.053E-11	1.15E-06	<	2.034E-04	2.51E-09	5.75E-05
sum		1.000E+00	**1.27E-05**	**2.78E-01**		1.000E+00	**1.44E-05**	**3.08E-01**		1.000E+00	**1.25E-05**	**2.84E-01**		1.000E+00	**1.45E-05**	**3.32E-01**
obs. β fraction		8.222E-01				8.429E-01				7.932E-01				7.845E-01		

Table 5
Nuclide Fractions and Dose From Eight Additional Cores
(Table 5 page 2 of 2)

	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34
Nuclide		**PAB Pipe Tunnel CA9900 13-C002-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm² detectable β 2004**		**PAB Pipe Tunnel CA9900 13-C003-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm² detectable β 2004**		**O/A Trench CA9900 12-C001-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm² detectable β 2004**		**O/A Trench CA9900 12-C002-A 2004 *nf***	**1 dpm dose Times nf**	**Dose For** 1.80E+04 **dpm/100 cm² detectable β 2004**	**1.0 dpm dose factor**
H-3	<	3.936E-03	1.319E-07	7.02E-03	<	2.169E-02	7.269E-07	2.88E-02	<	5.168E-03	1.732E-07	3.29E-03	<	4.438E-03	1.487E-07	2.88E-03	3.351E-05
C-14	<	3.618E-03	6.525E-08	3.48E-03	<	1.058E-02	1.908E-07	7.56E-03	<	3.978E-02	7.174E-07	1.36E-02	<	2.851E-02	5.141E-07	9.95E-03	1.803E-05
Mn-54	<	1.579E-04	3.321E-10	1.77E-05	<	9.359E-04	1.969E-09	7.80E-05		**3.607E-04**	7.587E-10	1.44E-05		**4.206E-05**	8.848E-11	1.71E-06	2.104E-06
Fe-55		**2.348E-02**	1.372E-08	7.31E-04		**5.639E-02**	3.295E-08	1.31E-03		**1.015E-03**	5.929E-10	1.13E-05		**2.553E-03**	1.492E-09	2.89E-05	5.843E-07
Co-57	<	3.182E-04	8.073E-11	4.30E-06	<	1.881E-03	4.772E-10	1.89E-05	<	3.853E-04	9.774E-11	1.86E-06	<	4.736E-04	1.202E-10	2.33E-06	2.537E-07
Co-58	<	5.662E-08	4.578E-14	2.44E-09	<	3.615E-07	2.923E-13	1.16E-08	<	4.889E-08	3.953E-14	7.52E-10	<	6.306E-08	5.099E-14	9.87E-10	8.086E-07
Ni-59	<	6.317E-03	7.636E-11	4.07E-06	<	4.415E-03	5.337E-11	2.11E-06	<	4.305E-04	5.204E-12	9.90E-08	<	5.810E-04	7.023E-12	1.36E-07	1.209E-08
Co-60		**9.653E-02**	6.086E-07	3.24E-02		**2.071E-01**	1.306E-06	5.17E-02		**5.186E-01**	3.270E-06	6.22E-02		**5.516E-01**	3.478E-06	6.73E-02	6.305E-06
Ni-63		**6.183E-01**	6.908E-08	3.68E-03		**4.322E-01**	4.828E-08	1.91E-03		**4.215E-02**	4.708E-09	8.96E-05		**5.687E-02**	6.354E-09	1.23E-04	1.117E-07
Zn-65	<	2.096E-04	2.216E-09	1.18E-04	<	1.271E-03	1.344E-08	5.32E-04	<	3.116E-04	3.295E-09	6.27E-05	<	4.043E-04	4.276E-09	8.27E-05	1.058E-05
Sr-90	**<**	2.305E-03	1.455E-07	7.75E-03		**1.651E-02**	1.042E-06	4.13E-02		**3.599E-03**	2.271E-07	4.32E-03		**3.117E-03**	1.967E-07	3.81E-03	6.310E-05
Nb-94	<	1.065E-03	1.782E-09	9.49E-05	<	7.745E-03	1.296E-08	5.14E-04	<	1.990E-03	3.331E-09	6.33E-05	<	2.535E-03	4.243E-09	8.21E-05	1.674E-06
Tc-99	<	4.053E-05	1.303E-08	6.94E-04	<	3.150E-05	1.013E-08	4.01E-04	<	6.755E-05	2.172E-08	4.13E-04	<	6.028E-05	1.939E-08	3.75E-04	3.216E-04
Ru-106	<	1.619E-03	1.993E-08	1.06E-03	<	8.987E-03	1.107E-07	4.38E-03	<	2.258E-03	2.781E-08	5.29E-04	<	2.832E-03	3.487E-08	6.75E-04	1.232E-05
Ag-110m	<	1.256E-03	4.333E-09	2.31E-04	<	6.093E-03	2.102E-08	8.33E-04	<	1.708E-03	5.890E-09	1.12E-04	<	2.147E-03	7.404E-09	1.43E-04	3.449E-06
Sb-125	<	2.264E-03	4.544E-09	2.42E-04	<	8.713E-03	1.749E-08	6.93E-04		**2.827E-03**	5.674E-09	1.08E-04	<	3.093E-03	6.208E-09	1.20E-04	2.007E-06
I-129	<	8.091E-08	5.250E-10	2.80E-05	<	6.277E-08	4.073E-10	1.61E-05	<	1.346E-07	8.733E-10	1.66E-05	<	1.201E-07	7.791E-10	1.51E-05	6.489E-03
Cs-134		**4.255E-03**	1.014E-07	5.40E-03	<	3.114E-03	7.424E-08	2.94E-03		**1.951E-03**	4.651E-08	8.85E-04		**1.361E-03**	3.246E-08	6.28E-04	2.384E-05
Cs-137		**2.222E-01**	3.567E-06	1.90E-01		**1.725E-01**	2.769E-06	1.10E-01		**3.701E-01**	5.942E-06	1.13E-01		**3.303E-01**	5.303E-06	1.03E-01	1.605E-05
Ce-144	<	3.600E-04	3.731E-10	1.99E-05	<	2.155E-03	2.233E-09	8.85E-05	<	4.181E-04	4.333E-10	8.24E-06	<	5.199E-04	5.389E-10	1.04E-05	1.036E-06
Pm-147	<	7.735E-04	1.569E-09	8.36E-05	<	1.766E-03	3.581E-09	1.42E-04	<	1.674E-04	3.395E-10	6.46E-06	<	1.755E-04	3.560E-10	6.89E-06	2.028E-06
Eu-154	<	2.060E-03	1.584E-09	8.44E-05	<	9.923E-03	7.631E-09	3.02E-04	<	3.405E-03	2.619E-09	4.98E-05	<	4.419E-03	3.398E-09	6.58E-05	7.690E-07
Eu-155	<	1.245E-03	1.224E-10	6.52E-06	<	8.223E-03	8.081E-10	3.20E-05	<	1.787E-03	1.756E-10	3.34E-06	<	2.288E-03	2.249E-10	4.35E-06	9.828E-08
Pu-238	<	6.927E-05	9.641E-09	5.14E-04		**2.898E-04**	4.033E-08	1.60E-03		**3.849E-05**	5.357E-09	1.02E-04		**3.581E-05**	4.984E-09	9.64E-05	1.392E-04
Pu-239	<	2.271E-05	3.497E-09	1.86E-04		**2.607E-04**	4.015E-08	1.59E-03		**1.411E-05**	2.172E-09	4.13E-05		**2.565E-05**	3.949E-09	7.64E-05	1.540E-04
Pu-240	<	2.270E-05	3.496E-09	1.86E-04		**2.607E-04**	4.014E-08	1.59E-03		**1.411E-05**	2.172E-09	4.13E-05		**2.564E-05**	3.948E-09	7.64E-05	1.540E-04
Pu-241	<	6.730E-03	2.009E-08	1.07E-03		**1.536E-02**	4.584E-08	1.82E-03		**1.457E-03**	4.348E-09	8.27E-05		**1.527E-03**	4.557E-09	8.82E-05	2.985E-06
Am-241	<	6.356E-04	2.961E-08	1.58E-03		**1.557E-03**	7.252E-08	2.87E-03		**1.681E-05**	7.832E-10	1.49E-05	<	2.242E-06	1.045E-10	2.02E-06	4.658E-05
Cm-242	<	3.505E-06	2.454E-12	1.31E-07	<	9.076E-07	6.356E-13	2.52E-08	<	4.845E-09	3.393E-15	6.45E-11	<	4.715E-09	3.302E-15	6.39E-11	7.002E-07
Cm-243	<	1.086E-04	1.663E-09	8.86E-05	<	6.707E-05	1.027E-09	4.07E-05		**1.208E-06**	1.850E-11	3.52E-07	<	2.123E-07	3.251E-12	6.29E-08	1.531E-05
Cm-244	<	1.041E-04	1.282E-09	6.83E-05	<	6.425E-05	7.914E-10	3.14E-05		**1.158E-06**	1.426E-11	2.71E-07	<	2.034E-07	2.505E-12	4.85E-08	1.232E-05
sum		1.000E+00		**2.57E-01**		1.000E+00	**6.63E-06**	**2.63E-01**		1.000E+00	**1.05E-05**	**1.99E-01**		1.000E+00	**9.78E-06**	**1.89E-01**	
obs. β fraction		3.379E-01				4.544E-01				9.464E-01				9.302E-01			

Table 6
Nuclide Fraction and Dose After Removal of MDA Nuclides
O/A Trench and PAB Pipe Tunnel
(Table 6 Page 1 of 2)

Column # ==>	1	2	3	4	5	6	7	8	9	10
7/30/02		**Pipe Tunnel**	**Pipe Tunnel**		**Dose For**		**Pipe Tunnel**	**Pipe Tunnel**		**Dose For**
		CA9900	**CA9900**		1.80E+04		**CA9900**	**CA9900**		1.80E+04
		13-C002-A	**13-C002-A**		**dpm/100 cm2**		**13-C003-A**	**13-C003-A**		**dpm/100 cm2**
		2004	**2004**	**1 dpm dose**	**detectable β**		**2004**	**2004**	**1 dpm dose**	**detectable β**
Nuclide		**initial nf**	**normalized nf**	**Times nf**	**2004**		**initial nf**	**normalized nf**	**Times nf**	**2004**
Mn-54	<	1.579E-04	1.616E-04	3.40E-10	1.83E-05	<	9.359E-04	1.023E-03	2.151E-09	8.689E-05
Fe-55		**2.348E-02**	**2.403E-02**	1.40E-08	7.54E-04		**5.639E-02**	6.162E-02	3.600E-08	1.454E-03
Co-60		**9.653E-02**	**9.878E-02**	6.23E-07	3.34E-02		**2.071E-01**	2.262E-01	1.426E-06	5.762E-02
Ni-63		**6.183E-01**	**6.328E-01**	7.07E-08	3.80E-03		**4.322E-01**	4.722E-01	5.275E-08	2.131E-03
Sr-90	<	2.305E-03	2.359E-03	1.49E-07	7.99E-03		**1.651E-02**	1.804E-02	1.138E-06	4.597E-02
Sb-125	<	2.264E-03	2.317E-03	4.65E-09	2.50E-04	<	8.713E-03	9.520E-03	1.911E-08	7.718E-04
Cs-134		**4.255E-03**	**4.354E-03**	1.04E-07	5.58E-03	<	3.114E-03	3.402E-03	8.112E-08	3.276E-03
Cs-137		**2.222E-01**	**2.274E-01**	3.65E-06	1.96E-01		**1.725E-01**	1.884E-01	3.025E-06	1.222E-01
Pu-238	<	6.927E-05	7.088E-05	9.87E-09	5.30E-04		**2.898E-04**	3.166E-04	4.407E-08	1.780E-03
Pu-239	<	2.271E-05	2.324E-05	3.58E-09	1.92E-04		**2.607E-04**	2.849E-04	4.386E-08	1.772E-03
Pu-240	<	2.270E-05	2.323E-05	3.58E-09	1.92E-04		**2.607E-04**	2.848E-04	**4.385E-08**	1.771E-03
Pu-241	<	6.730E-03	6.888E-03	2.06E-08	1.10E-03		**1.536E-02**	1.678E-02	5.009E-08	2.023E-03
Am-241	<	6.356E-04	6.505E-04	3.03E-08	1.63E-03		**1.557E-03**	1.701E-03	7.924E-08	3.201E-03
Cm-243	<	1.086E-04	1.112E-04	1.70E-09	9.14E-05	<	6.707E-05	7.328E-05	1.122E-09	4.531E-05
Cm-244	<	1.041E-04	1.065E-04	1.31E-09	7.05E-05	<	6.425E-05	7.020E-05	8.647E-10	3.493E-05
sum		9.772E-01	1.000E+00	4.686E-06	2.556E-01		9.152E-01	1.000E+00	6.061E-06	2.517E-01
obs. b fraction		3.28E-01	3.35E-01				4.08E-01	4.46E-01		

Table 6
Nuclide Fractions and Dose After Removal of MDA Nuclides
O/A Trench and PAB Pipe Tunnel
(Table 6 page 2 of 2)

Column # ==> 11	12	13	14	15	16	17	18	19	20
Nuclide	O/A Trench CA9900 12-C001-A 2004 initial nf	O/A Trench CA9900 12-C001-A 2004 normalized nf	1 dpm dose Times nf	Dose For 1.80E+04 dpm/100 cm2 detectable β 2004		O/A Trench CA9900 12-C002-A 2004 initial nf	O/A Trench CA9900 12-C002-A 2004 normalized nf	1 dpm dose Times nf	Dose For 1.80E+04 dpm/100 cm2 detectable β 2004
Mn-54	3.607E-04	3.828E-04	8.053E-10	1.522E-05		4.206E-05	4.425E-05	9.308E-11	1.790E-06
Fe-55	1.015E-03	1.077E-03	6.293E-10	1.190E-05		2.553E-03	2.685E-03	1.569E-09	3.018E-05
Co-60	5.186E-01	5.504E-01	3.470E-06	6.561E-02		5.516E-01	5.803E-01	3.659E-06	7.038E-02
Ni-63	4.215E-02	4.473E-02	4.998E-09	9.448E-05		5.687E-02	5.982E-02	6.684E-09	1.286E-04
Sr-90	3.599E-03	3.820E-03	2.411E-07	4.557E-03		3.117E-03	3.279E-03	2.069E-07	3.980E-03
Sb-125	2.827E-03	3.000E-03	6.022E-09	1.138E-04	<	3.093E-03	3.253E-03	6.530E-09	1.256E-04
Cs-134	1.951E-03	2.071E-03	4.937E-08	9.333E-04		1.361E-03	1.432E-03	3.415E-08	6.568E-04
Cs-137	3.701E-01	3.928E-01	6.307E-06	1.192E-01		3.303E-01	3.475E-01	5.579E-06	1.073E-01
Pu-238	3.849E-05	4.085E-05	5.686E-09	1.075E-04		3.581E-05	3.767E-05	5.243E-09	1.008E-04
Pu-239	1.411E-05	1.498E-05	2.306E-09	4.359E-05		2.565E-05	2.698E-05	4.154E-09	7.991E-05
Pu-240	1.411E-05	1.497E-05	2.305E-09	4.358E-05		2.564E-05	2.697E-05	4.153E-09	7.989E-05
Pu-241	1.457E-03	1.546E-03	4.615E-09	8.724E-05		1.527E-03	1.606E-03	4.794E-09	9.222E-05
Am-241	1.681E-05	1.785E-05	8.313E-10	1.571E-05	<	2.242E-06	2.359E-06	1.099E-10	2.114E-06
Cm-243	1.208E-06	1.283E-06	1.963E-11	3.712E-07	<	2.123E-07	2.234E-07	3.419E-12	6.577E-08
Cm-244	1.158E-06	1.229E-06	1.513E-11	2.861E-07	<	2.034E-07	2.140E-07	2.636E-12	5.070E-08
sum	9.421E-01	1.000E+00	1.010E-05	1.909E-01		9.506E-01	1.000E+00	9.512E-06	1.830E-01
obs. b fraction	8.97E-01	9.52E-01				8.90E-01	9.36E-01		

Table 7
Mean and Standard Deviation of Dose
For TRU Affected Cores
Using the Average of the Fractions and Individual Core Method

Column # ==>	2	3	4	5	6	7	8	9
	4 Cores	Dose Results (in mrem) for Average of the Fractions (1.0 dpm)						
Nuclide	**Mean nf**	**13-C002-A**	**13-C003-A**	**12-C001-A**	**12-C002-A**	**mean***	**stdev mean**	**(stdev Mean)^2**
Mn-54	4.028E-04	3.40E-10	2.151E-09	8.053E-10	9.308E-11	2.286E-05	1.238E-05	1.533E-10
Fe-55	2.235E-02	1.40E-08	3.600E-08	6.293E-10	1.569E-09	3.523E-04	2.222E-04	4.935E-08
Co-60	3.639E-01	6.23E-07	1.426E-06	3.470E-06	3.659E-06	6.190E-02	2.030E-02	4.120E-04
Ni-63	3.024E-01	7.07E-08	5.275E-08	4.998E-09	6.684E-09	9.114E-04	4.464E-04	1.993E-07
Sr-90	6.874E-03	1.49E-07	1.138E-06	2.411E-07	2.069E-07	1.170E-02	6.355E-03	4.039E-05
Sb-125	4.523E-03	4.65E-09	1.911E-08	6.022E-09	6.530E-09	2.449E-04	9.084E-05	8.252E-09
Cs-134	2.815E-03	1.04E-07	8.112E-08	4.937E-08	3.415E-08	1.811E-03	4.226E-04	1.786E-07
Cs-137	2.890E-01	3.65E-06	3.025E-06	6.307E-06	5.579E-06	1.252E-01	2.097E-02	4.395E-04
Pu-238	1.165E-04	9.87E-09	4.407E-08	5.686E-09	5.243E-09	4.375E-04	2.520E-04	6.352E-08
Pu-239	8.752E-05	3.58E-09	4.386E-08	2.306E-09	4.154E-09	3.636E-04	2.735E-04	7.479E-08
Pu-240	8.750E-05	3.58E-09	4.385E-08	2.305E-09	4.153E-09	3.635E-04	2.734E-04	7.476E-08
Pu-241	6.705E-03	2.06E-08	5.009E-08	4.615E-09	4.794E-09	5.399E-04	2.886E-04	8.330E-08
Am-241	5.929E-04	3.03E-08	7.924E-08	8.313E-10	1.099E-10	7.452E-04	5.015E-04	2.515E-07
Cm-243	4.649E-05	1.70E-09	1.122E-09	1.963E-11	3.419E-12	1.920E-05	1.137E-05	1.292E-10
Cm-244	4.454E-05	1.31E-09	8.647E-10	1.513E-11	2.636E-12	1.480E-05	8.760E-06	7.674E-11

* 1 dpm average value times 18,000/0.6672 (obs. average beta fraction) = 2.698E+04

Mean
2.046E-01
Standard Deviation Of the Mean
2.988E-02

Core Dose Results From Individual Cores 18,000 dpm/100 cm^2 Observable Beta

Nuclide	**13-C002-A**	**13-C003-A**	**12-C001-A**	**12-C002-A**
Mn-54	1.83E-05	8.689E-05	**1.522E-05**	**1.790E-06**
Fe-55	7.54E-04	1.454E-03	**1.190E-05**	**3.018E-05**
Co-60	3.34E-02	5.762E-02	**6.561E-02**	**7.038E-02**
Ni-63	3.80E-03	2.131E-03	**9.448E-05**	**1.286E-04**
Sr-90	7.99E-03	4.597E-02	**4.557E-03**	**3.980E-03**
Sb-125	2.50E-04	7.718E-04	**1.138E-04**	**1.256E-04**
Cs-134	5.58E-03	3.276E-03	**9.333E-04**	**6.568E-04**
Cs-137	1.96E-01	1.222E-01	**1.192E-01**	**1.073E-01**
Pu-238	5.30E-04	1.780E-03	**1.075E-04**	**1.008E-04**
Pu-239	1.92E-04	1.772E-03	**4.359E-05**	**7.991E-05**
Pu-240	1.92E-04	1.771E-03	**4.358E-05**	**7.989E-05**
Pu-241	1.10E-03	2.023E-03	**8.724E-05**	**9.222E-05**
Am-241	1.63E-03	3.201E-03	**1.571E-05**	**2.114E-06**
Cm-243	9.14E-05	4.531E-05	**3.712E-07**	**6.577E-08**
Cm-244	7.05E-05	3.493E-05	**2.861E-07**	**5.070E-08**
dose sum	2.517E-01	2.441E-01	1.909E-01	1.830E-01

Individual Core (4) Propagation of Error
Mean
2.17E-01
Standard Deviation of the Mean
1.77E-02

ATTACHMENT 2G

Supplemental Information Regarding Concrete Core Data Use

Supplemental Information Regarding Concrete Core Data Use

To characterize contaminated concrete surfaces, there were three sets of concrete cores obtained and analyzed. The resulting data was used to establish the appropriate nuclide fractions and support the dose assessment in Section 6. Each core set was taken for different reasons and was analyzed by methods appropriate to each set's purpose. The following discussion summarizes the purpose of each set and key elements of the analysis for each.

A. Initial Set of Concrete Cores (Initial Site Characterization)

The first set of cores were collected during initial site characterization by GTS Duratek and were used to represent typical concrete nuclide data. Seven of these cores with the highest total activity were selected for off-site analysis to determine the amount of HTD nuclides present. (Using the highest activity cores offered the best chance of detection for low activity HTD nuclides.) The HTDs were determined using radiochemical analytical techniques; gamma emitting nuclides were determined by gamma spectroscopy with the cores counted 21 inches above the detector to approximate a point source. The results from these cores formed the basis for the establishment of the contaminated concrete surface nuclide fraction for the majority of basement concrete surfaces (i.e., the "balance of plant" concrete surfaces). Certain subsequent core samples and analyses would lead to establishing a separate, unique nuclide fraction for limited areas warranting such treatment. This is discussed below. See Section 2.5.3a and Attachment 2F for additional detail.

B. Second Set of Concrete Cores

Forty three (43) additional cores were collected during continuing site characterization. This data was used primarily for establishing E_t for contaminated concrete. The number of cores obtained was established so that each building or plant area would have several cores included in the data analysis with the goal that the sample population, as a whole, would more accurately represent the nuclide ratios for concrete surfaces. These 43 core samples were processed for the E_t determination by initially gross counting the cores, followed by gamma spectroscopy analysis. The cores were counted initially onsite; six cores were later recounted at an offsite vendor's laboratory (DES[1]). The onsite HPGe detectors had been calibrated using a concrete standard of uniform activity. The samples were counted at DES using a similar geometry, and the results showed good agreement. In order to determine total activity for the E_t calculation, six of the cores were dissolved, and the dissolved material was again counted using the geometry specific to the analytical technique. The counting results for the dissolved cores showed that the activity was mostly on the surface of the concrete. (Later evaluation of the data using Microshield modeling verified that the Co-60 activity was located on the surface of the concrete and had a correction factor of approximately 0.5 while the Cs-137 activity was as deep as 1 mm in the core

[1] Duke Engineering and Services Environmental Laboratory, now referred to as "Framatome ANP DE&S Environmental Laboratory."

and had a correction factor of 0.73.) An average correction factor was determined to convert the activity from a surface count to the total activity in the core. The value of the correction factor was determined to be 0.68 from the DES data, as compared to 0.67 based on the onsite data.

C. Third Set of Concrete Cores

Upon reviewing some of the original GTS data, a question remained concerning the possible presence of TRUs on concrete surfaces. A specific area of concern was the containment outer annulus trench. A decision was made to obtain cores on either side of an original trench sample to confirm or disprove the presence of TRUs. At the same time, additional cores were obtained to replace those destroyed by sample analysis. Three additional cores were collected from within the loops of containment, and three additional samples were collected in the PAB. Thus, the third set of cores totaled 8.

This set of core samples was analyzed by gross counting, gamma spectroscopy, and offsite analysis for HTD nuclides. This data formed the basis for the development of the alternate concrete nuclide fraction for trenches, pipe tunnel and other unique ("special") areas, as discussed in Section 2.5.3.a and Attachment 2F. Background information related to the development of this nuclide fraction was described in a special report from the Technical Issue Resolution Process (TIRP). The report addressed a number of concerns related to the presence of TRUs in certain plant locations. (See Section 2.7, References.)

D. Core Data Adjustments

The nuclide fraction given in Table 2-7 of the LTP was derived from the data provided by the seven original cores. Four of the additional eight cores were confirmed to be included in the "balance of plant" concrete surfaces, as represented by the initial seven cores. The remaining four cores in the "third" set supported the establishment of a nuclide fraction for the "special areas" involving the various trenches and areas which were confirmed (or expected) to contain TRUs. (See Table 2-8.)

The data reported in Table 2C-2 of the LTP is a combination of the 43 additional cores plus the eight cores from the "third" data set. The core activities were reported with no geometry correction in Attachment 8 of EC 010-01. The core activities were then geometry corrected for use in the E_t calculation (Attachment. 5 of EC 010-01), and the geometry corrected data were presented in the LTP Table 2C-2 except for the activated concrete samples (Sample # 3-1A, 3-2A, and 3-3A) which were used only for activated concrete characterization.

E. Net Count Rate

The net count rate data were determined by counting the cores in a low background area following their removal from the building floors. The count rate values were adjusted for ambient area background, and the "net cpm" was reported in Table 2C-2.

ATTACHMENT 2H

Forebay and Diffuser Characterization Discussion

Forebay and Diffuser Characterization Discussion

1. Physical Description of the Forebay/ and Diffuser

The principal forebay structure consists of the forebay basin which is approximately 400 feet in length with a granite floor, rock and soil walls (or dikes), and concrete structures at both ends. The forebay is aligned generally in a north-south direction such that the concrete structures are located at the north and south ends with the dikes forming the east and west sides. The seal pit is at the northern end, and the diffuser intake structure is located at the southern end. During operations, plant cooling water discharged into the seal pit and then flowed over a concrete seal pit weir wall, into the forebay basin. With the cooling water system permanently secured, the flow in and out of the forebay is influenced primarily by tidal fluctuations. The forebay connects to the Back River through the diffuser piping. The intake to the diffuser piping is at the southern end of the forebay. See Figure 2H-1.

The forebay dikes were designed and constructed to achieve structural stability and minimize leakage by the choice, dimensions, and placement of pervious, impervious, and protective materials. On the interior sides of the dikes (that is, on the forebay side), the exterior layer consists of two feet (or greater) of large protective "coarse rock" (rip-rap). Beneath the rip-rap is about two feet of cobble stones[1]. Underneath the cobble stone layer is about two feet of gravel ("pervious fill"). Finally, beneath the gravel layer is impervious fill material. The dike walls are inclined at a slope of approximately 1.75:1 (that is, 1.75 feet horizontal run for every 1 foot of vertical drop) which results in a slope angle of about 30 degrees from the horizontal plane. See Figure 2H-2.

The diffuser system consists of large fiberglass pipes which connect the forebay basin to the diffuser discharge, submerged in the Back River. At the forebay's southern end, the diffuser supply piping is nine feet in diameter. Downstream sections continually decrease to a diameter of approximately 5 feet with nozzles of 18 inches in diameter, spaced in the diffuser discharge piping. The diffuser at its discharge is submerged at a depth of over 40 feet below MSL.

The characterization of the forebay identified the following principal contaminated media:

- Floors of the forebay and seal pit. This includes other concrete surfaces, such as the seal pit weir wall. (This weir wall will be demolished down to 3' below grade.)
- Rip-rap, contaminated on the rock surfaces.
- Marine sediment (primary organic material), deposited on floors of forebay basin and seal pit and around the rip-rap.

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[1] This 2 foot thick layer is specified to be "6 inch minus," i.e., containing material no greater than 6" in diameter. In Figure 2H-2, this layer is referred to as "fine rock cover."

- Dike "soil", that is, any material interior to dike below the rip-rap covering, including cobble, gravel, and other soil materials, as well as sediment deposited around the cobble.[2]

Remediation plans call for the removal of a majority of the accessible marine sediment in the forebay. Once the sediment remediation is accomplished, the principal contamination source term is expected to be the dike "soil" beneath the rip-rap, based on the assessment of activity levels in the various media. As noted above, the other contaminated media that would remain are the rip-rap (with surface contamination) and whatever sediment and other surface contamination that may remain on forebay/seal rock and concrete floors. See Section 6.6.9 for the discussion of the dose assessment and contribution of each of these remaining contaminated media.

The characterization of the diffuser identified two principal contaminated media, namely:

- Marine sediment that has been re-deposited internal to the diffuser piping by tidal action (following the permanent shutdown of the plant's cooling water system).
- Contaminated internal surfaces of the diffuser fiberglass piping.

Seaweed is also considered in the diffuser dose assessment; therefore, characterization information is discussed in this attachment. See Section 6.6.9 for the dose assessment related to diffuser source terms.

2. Forebay (and Seal Pit): Contaminated Media Characterization

As part of the site's initial characterization (by GTS-Duratek), several forebay samples were obtained and analyzed. Subsequent to that sampling (late 2000), an additional set of 15 sediment samples were obtained by Maine Yankee (see EC 004-01), composited, and analyzed for HTDs. The LTP Rev. 1 nuclide fraction for forebay sediment (Section 2.5.3.e) was established based on this sampling and analysis (decay corrected to 1/1/2004). No TRU's were detected in this 2000 composite sample.[3] This nuclide fraction is presented in Table 2H-1 below.

In 2001, an expanded sampling program was developed and implemented to support further characterization and remediation planning. This effort involved more extensive sampling of the forebay and principal forebay features to gain insight regarding spatial variations in activity, sediment deposition, and the activity depth profile interior to the forebay dikes. At the same time, remediation planning was involved in a number of studies and field tests to determine the optimum remediation techniques. These studies and tests also included the evaluation of material handling equipment required to address the somewhat unique challenges of the forebay, given the marine environment, variety of material sizes (from rip-rap to glacial till), and

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[2] An additional, extensive dike coring program was completed in the third quarter of 2002 to better define remediation requirements of the dike soil beneath the rip-rap. See Section 2.4 of this attachment for additional detail.

[3] The composite forebay sediment sample was analyzed for a standard suite of TRU nuclides. See Attachment 1 of EC-041-01 for identification of specific nuclides.

relatively steep slopes (of the forebay dikes).

Table 2H-1. Forebay Sediment Nuclide Fraction (Decay corrected to 1/1/2004)

Nuclide	Fraction
Co-60*	0.567
Cs-137*	0.030
Sb-125	0.005
Fe-55	0.165
Ni-63	0.233

* The resulting Co-60/Cs-137 from this data is 18.9.

The 2001 sampling program included the following principal tasks:

- Sampling of organic sediment around the rip-rap on both the east and west dikes;
- Sampling of sediment material accumulated on exposed rock surfaces in the vicinity of the weir wall at the northern end of the forebay;
- Sampling of underwater sediment on forebay basin floor and on the bottom (floor) of the seal pit.
- Subsequent, depth profile sampling into the dike material or "soil."

In addition, as part of work directly related to remediation planning, rip-rap surface samples were analyzed for material composition and activity concentration.

The results of the characterization efforts are summarized below. See EC-041-01 for additional detail on sample locations, individual sample results, analysis of results, and use in the dose assessment

2.1 Dike Spatial Activity Distribution

A total of forty (40) sediment samples were taken to provide information of spatial variance of activity in the sediment deposited in the tidal zone around the rip-rap on the forebay dike interior surfaces. Twenty (20) samples were obtained on each dike, i.e., ten samples along the high tide line and ten (10) samples along the low tide line. See Table 2H-2.

Table 2H-2. Sediment Around Rip-Rap at Forebay High & Low Tide Lines									
	Co-60 (pCi/g)				Cs-137 (pCi/g)				
Sample Location	Max	Min	Avg	Std Dev	Max	Min	Avg	Std Dev	Co/Cs Ratio
High Tide Line (East & West Combined)	92.6	1.8	16.9	21.8	6.5	0.2	1.2	1.4	14.6
Low Tide Line (East & West Combined)	62.7	4.5	22.5	15.2	1.9	0.3	1.0	0.5	22.0
High & Low Tide Line Combined	92.6	1.8	19.7	18.8	6.5	0.2	1.1	1.0	18.0

As shown in Table 2H-2, the sediment samples collected at the low tide line reported a higher Co-60 average than those collected at the high tide line. (The Cs-137 values for both high and low tide were relatively low by comparison.)

The two tidal area sediment samples with the highest reported Co-60 activity were 63.6 and 92.6 pCi/g, collected on the northern portion of the west dike at high tide. See Figure 2H-1. Because of the high concentrations, these particular locations were chosen for additional sampling to explore the activity profile interior to the dikes. The results from this effort are described below in Section 2.4 (of this attachment).

While these levels in the tidal area sediment are high relative to remediation levels (i.e., the DCGL proposed in Section 6 dose modeling), the profile sampling confirmed at these locations that a large portion of the contamination is near the dike soil surface, that is, the material immediately beneath the rip-rap covering. Later, more extensive sampling of the dike soil beneath the rip-rap demonstrated that the contaminated material has not penetrated beneath the rip-rap to any significant extent. (See Section 2.4 for additional discussion.) Since the contaminated sediment is generally accessible, loose, and concentrated near the surface, measures under consideration for sediment remediation around the rip-rap and on the basin/seal pit floors are expected to be quite effective.

Dose modeling addresses each of the contaminated media (described in Section 1 of this attachment) including separate treatment of contaminated floors and the interior dike soil. See Section 6.6.9.

2.2 Exposed Sediment Material (in vicinity of weir wall)

Nine (9) samples were collected from material (sediment, soil, and other material) available on the exposed rock, i.e., having no rip-rap layer, at the northern end of the forebay/seal pit structure in the area of the seal pit weir. Most of these samples were obtained on the west side to provide appropriate coverage of the area in the path of the emergency spillway.[4]

This set of exposed sediment samples exhibited the highest activity concentrations of all samples obtained in this particular sampling campaign of Spring 2001. See Table 2H-3 a summary of these results.

Table 2H-3. Sample Results: Sediment from Exposed Rock Surfaces and Underwater Sediment

	Co-60 (pCi/g)					Cs-137 (pCi/g)			
Sample Location	**Max**	**Min**	**Avg**	**Std Dev**		**Max**	**Min**	**Avg**	**Std Dev**
Exposed Sediment Material	445.0	0.2	65.9	148.3		23.8	0.3	3.3	7.7
Underwater Sediment (Forebay and Seal Pit)	62.7	5.5	19.0	16.4		7.0	0.2	1.9	2.1

a. The average activities of the exposed sediment material samples were: 65.9 pCi/g Co-60 and 3.3 pCi/g Cs-137. The maximum reported activity, 445 pCi/g Co-60 and 23.8 pCi/g Cs-17, was associated with a sample collected on the western side, near the weir. See Figure 2H-1 for approximate location. The second highest sample, collected from an area immediately adjacent to the above sample (on the exposed rock), reported 130 pCi/g Co-60 and 3.3 pCi/g Cs-137.

b. Not only did these samples report the maximum activity for any location sampled in this campaign, but also they were particularly high relative to the other exposed sediment samples. For sample the Co-60 concentrations ranged from 0.2 to 10.7 pCi/g and Cs-137 from 0.3 to 0.5 pCi/g for the other seven (7) exposed sediment samples.

c. The average exposed sediment sample activities (excluding the two highest samples) were 2.64 pCi/g Co-60 and 0.4 pCi/g Cs-137. The average activities for all nine (9) exposed sediment samples were 65.9 pCi/g and 3.3 pCi/g Cs-137. The average Co/Cs ratio was 19.8 (using the data from all nine samples).

d. The two highest exposed sediment samples were sent to an outside laboratory for HTD analyses. The nuclide fraction results from these HTD analyses were

4 From 1972 until late 1974, cooling water discharge passed over the weir and directly into Bailey Cove. During that time period, the flow path included portions of exposed rock now part of the western dike (at the northern end). Construction of the west dike and diffuser system was completed in 1975. The western exposed rock then became part of an emergency spillway to provide a pathway in the event the diffuser system was not operating properly.

comparable to the 2000 composite sediment HTD results with the exception that the exposed sediment sample analysis identified the presence of TRU nuclides in very low concentrations. The MDC values of the 2000 composite sediment sample analyses would have been low enough to detect the TRU nuclides had they been present at the levels found in the later exposed sediment samples.[5] The original and later HTD data sets were compared and evaluated. The TRU nuclides, reported in the exposed sediment samples, were determined to represent less than 1% of the total dose associated with forebay media and were, therefore, eliminated from the nuclide fraction. Overall, it was determined that the original nuclide fraction for sediment (reported in LTP Rev. 1) was conservative due to the its higher proportion of dose-significant gamma emitters (i.e., Co-60, Cs-137, and Sb-125). The original nuclide fraction was, therefore, used in the dose assessment.

e. Lastly, as mentioned above, the exposed rock area, by its nature, contains only a small amount of material. While two of the exposed sediment samples reported very high activity, it is expected that remediation measures in this area will be quite effective because the total volume of material on these exposed rock surfaces is relatively small and because the contamination is loose and accessible.

2.3 Underwater Forebay (and Seal Pit) Sediment

Thirteen (13) sediment samples were taken from underwater areas in the forebay and seal pit. Activity levels for underwater sediment were comparable to that of sediment deposited on the dikes around the rip-rap, presented in Table 2H-2 above. The overall average activities (combining forebay and seal pit samples) are 19.0 pCi/g Co-60 and 1.9 pCi/g Cs-137. Table 2H-3 summarizes the results from this sampling. Since this sediment is accessible (by diving operation) and can be vacuumed by any number of proven techniques, remediation measures for this contaminated media are expected to be quite effective.

2.4 Dike "Soil" Activity Profile

As discussed in Section 2.1 above, depth profile samples were taken at the two locations exhibiting the highest activity levels in the rip-rap tidal zone. This sampling was undertaken to gain further insight regarding the penetration of activity into the dike interior (and to support remediation planning). See Figure 2H-1 for the surface (starting) location for these profile samples.

The depth profile samples were taken in 6" intervals down to a depth of 24." The dike

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[5] See Attachment 1 of EC-041-01 for the listing of MDC values obtained in the subject sediment analyses by Duke Engineering and Services Laboratory, i.e., the "2000 composite" forebay sediment sample and the more recent, higher activity exposed sediment samples (Sample numbers: H059 and H060).

soil material for each 6" interval was composited. Both series demonstrated a generally decreasing activity concentration with depth. See Table 2H-4, which provides the average Co-60 and Cs-137 activities values (average of the two profile samples at a given profile location). Overall, this initial data indicated that the majority of the contamination was concentrated near the surface of the dike soil. This initial information on potential dike soil activity, while limited, was used in the forebay dose assessment.

It was recognized that additional sampling of the dike soil was appropriate for remediation planning and to confirm activity level assumptions used in the dose assessment. This sampling effort involved the use of coring into the area beneath the rip-rap (parallel to the slope) by way of inclined drilling from the top of the dike, as well as several vertical corings near the centerline of each dike. This dike coring campaign was completed in the third quarter of 2002.

The dike soil samples taken from both vertical and inclined corings revealed very low levels of contamination, much lower than that assumed in the forebay dose assessment. The sampling program was quite extensive and involved a total of 19 corings (total vertical and inclined), including corings at the approximate locations at which the previous two profile samples (presented in Table 2H-4) were taken. The 19 corings were made down to the bedrock layer beneath the dikes and varied in depth from approximately 12 to 80 feet. Samples were taken by compositing material from approximately each meter of depth. This sampling density resulted in over 270 individual samples, with approximately 210 coming from the inclined corings.

The samples were analyzed by gamma spectroscopy on site (i.e., using a HPGe detector). Most of the 270 dike samples from the later campaign were analyzed to be less than the MDA. The averages of all positively detected Co-60 (six positives) and Cs-137 (38 positives) were 0.071 pCi/g and 0.082 pCi/g, respectively. These levels are much lower than the values used in the forebay dose assessment for dike soil (Section 6.6.9), as well as the surface soil DCGLs for Co-60 and Cs-137. These dike characterization results show that contaminated material has not, in general, penetrated to any significant extent into the dike material beneath the rip-rap. As noted above, the forebay dose assessment was based on the limited results from the two profile samples (shown in Table 2H-4). This later dike coring campaign is considered to be a more complete and representative characterization of dike soil contamination. However, since the values presented in Table 2H-4 are conservatively higher, the dose assessment (for dike soil) will continue to be based on Table 2H-4 and requires no change. Additional discussion on the dike coring results is provided in Maine Yankee's letter to the NRC, dated December 12, 2002 (Reference 2.7.26).

Table 2H-4. Depth Profile Sample Results[6] (Average activity values for samples collected at the listing location)			
Location	**Co-60 pCi/g**	**Cs-137 pCi/g**	**Co/Cs Ratio**
"Surface" sediment[7]	78.2	4.6	17.0
6" (Composite)[8]	15.3	0.9	16.7
12" (Composite)	6.7	0.6	11.0
18" (Composite)	2.6	0.3	8.2
24" (Composite)	2.8	0.2	12.0

2.5 Rip-Rap Rock, Surface Activity

As part of other remediation planning activities (mentioned above), material samples were obtained from rip-rap rock surfaces. The contamination was noted to adhere to the rip-rap rock surface much like that on diffuser piping surface, i.e., by being incorporated into an organic film. The surface material adhering to the rip-rap (in areas exposed to tidal action) exhibited the same general appearance as that found on the piping coupons retrieved for analysis from the diffuser piping. The surface activity concentrations (on rip-rap and diffuser piping) were also comparable. For these reasons, the rip-rap surface data and the information from the diffuser piping surfaces were used to establish the average rip-rap rock surface activities of 0.1 pCi/g Co-60 and 0.1 pCi/g Cs-137. Table 2H-5 lists the rip-rap surface activities and offers comparison to other media contamination levels.

2.6 Forebay/Seal Pit Floors and other Forebay Concrete Surfaces

No contamination data is available for the forebay/seal pit floors (or other forebay concrete surfaces). The largest surface area is represented by the forebay basin floor which consists of a granite ledge with a relatively low permeability and rock fill. Remediation methods expected for these surfaces are expected to be highly effective. Contamination levels for these surfaces will be confirmed as part of the remediation

6 Depth profile samples were collected at the location of the highest reported activities for sediment collected beneath the rip-rap in the tidal zone, i.e., "surface" sediment. See Section 2.1 in this attachment.

7 "Surface" sediment activities, presented here for comparison, are the averages of the two sediment samples, collected immediately beneath the rip-rap, which reported the highest activity.

8 These activities represent an average of the two samples taken at the listed interval, for example, dike soil collected and composited from the 0" to 6" interval.

process. From a dose assessment standpoint, a conservative surface contamination level (DCGL) was established to bound any contamination that may remain on the forebay/seal pit floor surfaces. See LTP Section 6.6.9.

Table 2H-5. Summary Media Activity Data for the Forebay/Seal Pit (for the Principal Nuclides)

	Co-60 pCi/g	Cs-137 pCi/g	Comment
Forebay floor (and limited concrete surfaces)	TBD	TBD	Expected to be largely remediated with remediation of marine sediment. Conservative surface contamination level assumed in dose assessment.
Rip-rap rock surface	0.1	0.1	Based on both diffuser and rip-rap rock surface samples. (Co-60/Cs-137 ratio: Approx. 1.0)
Marine sediment near rip-rap[9]	19.7	1.1	Marine sediment is expected to be largely remediated in the initial stage of forebay/seal pit remediation. (Co-60/Cs-137 ratio: Approx. 18.0)
Dike "soil" material[10]	0.071	0.082	Material beneath rip-rap (Co-60/Cs-137 ratio: Approx. 0.9)

3. Diffuser, Contaminated Media Characterization

As noted above, the principal diffuser contaminated media included: (1) marine sediment likely redeposited back into the diffuser discharge piping (following the permanent shutdown of the plant circulating water system) and (2) the diffuser piping internal surfaces. From a dose standpoint, the principal dose contributor is the marine sediment entrained in the diffuser. The plant derived activity in this sediment originated in the plant's licensed liquid effluent releases (via the forebay). Then, with the securing of plant operations and the cooling water system, the tidal action transported benthic silt back into the diffuser system. Plant derived activity concentrations reported for marine sediment now inside the diffuser piping are higher than that measured in sediment outside the piping.[11] The higher sediment activity inside the piping is believed to be due to activity absorbed or incorporated into the sediment inside the piping from

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9 Average of sediment samples collected beneath rip-rap. See Table 2H-2.

10 Sample data from the 2002 forebay dike coring campaign. Values shown are averages from the samples that resulted in a positive detection. See Section 2.4.

11 Per LTP Table 2B-5, Package R2000, samples taken near the diffuser reported a maximum Co-60 activity of 0.12 pCi/g.

the liquid effluent discharges since the end of plant operations. Although the dose consequences of the licensed liquid effluent releases which resulted in the activity in the diffuser have already been accounted for and reported in the routine effluent release reports, a dose assessment of the activity conservatively assumed to remain in the diffuser is discussed in Section 6.6.9.

As a matter of completeness in this discussion, seaweed characterization data is also included here since it is considered as a potential contaminated media in the dose pathway analysis. See the discussion below.

3.1 Diffuser: Marine Sediment Inside Diffuser Piping

During diving operations and inspections of diffuser discharge piping, sediment samples were obtained and analyzed by gamma spectroscopy. This analysis provided the following average activities are included in Table 2H-6.

Table 2H-6. Diffuser Related Characterization Summary[12]

	Co-60 pCi/g	Cs-137 pCi/g	Comment
Sediment inside diffuser discharge piping	1.1	0.15	Average activity. These sediment samples were also analyzed for HTDs. No HTD nuclides were detected. See EC 041-01. (Co-60/Cs-137 ratio: Approx. 7.3)
Diffuser inside piping surface	0.1	0.1	Average diffuser piping coupon activity. (Co-60/Cs-137 ratio: Approx. 1.0)
Seaweed	76.8	5.63	Average activity from forebay samples (as a conservative measure). See discussion in text. (Co-60/Cs-137 ratio: 13.6)

3.2 Diffuser Surfaces

During the above mentioned diving inspections of diffuser piping, coupons of the fiberglass piping were obtained and analyzed for surface contamination. The nuclides detected were Co-60 and Cs-137 at nearly equal activity. The activity levels detected were very near the MDA of 0.1 pCi/g for each nuclide and appeared to be present on the surface as a tightly adhered, thin film of organic material. The physical appearance of this material on the piping surface was similar to that noted on the contaminated rip-rap surfaces. The activity levels of the diffuser piping surface was also comparable to that on the rip-rap, suggesting similar physical mechanisms for adhering and incorporation of

12 See Attachment 3 of EC-041-01 for additional detail regarding diffuser characterization sampling, such as number of samples and individual results.

contamination at work.

3.3 Seaweed Activity, Relevant to the Diffuser Dose Assessment

Seaweed is present in the forebay and shoreline areas around Bailey Point. Dose contributions via contaminated seaweed were considered in the diffuser dose model as a matter of completeness, even though the dose contribution was expected (and confirmed) to be low. Seaweed samples taken from shoreline locations have shown sporadic and low activity levels of radionuclide uptake. Seaweed samples taken from the forebay were used in the dose assessment as a conservative measure of any seaweed related dose.[13] See Section 6.6.9 for seaweed use, pathway assumptions, and dose results. The seaweed activity values presented in Table 2H-6 are associated with forebay samples but were applied to the diffuser dose assessment.

4. Nuclide Fraction for Forebay/Diffuser Material

In summary, characterization samples were obtained and analyzed from contaminated media associated with the forebay/seal pit structures, including sediment under water and around the rip-rap, material on exposed rock (near the weir), dike "soil" beneath the rip-rap, and rip-rap surfaces. Additional samples were taken and analyzed from sediment inside the diffuser piping, as well as material deposited on diffuser piping internal surfaces. HTD analyses were performed on 3 collections of sediment sampling sets: an earlier (MY) composite of 15 samples, two high activity samples from the exposed sediment material, and sediment collected from inside the diffuser piping. An examination of these results concluded that the original HTD sample set, used to establish the LTP Rev. 1 nuclide fraction are appropriate and conservative nuclide fractions. The sample analyses also consistently confirmed that Co-60 and Cs-137 were the principal nuclides of interest. As noted in Table 2H-7, the Co/Cs ratios for the various contaminated media are comparable, spanning the range of 10.1 to 19.8.

The Co/Cs ratios were, in general, found to be lower for lower activity samples, as would be expected. This was seen in the assessment of contamination on rip-rap and diffuser piping surfaces, as well as deeper dike soil samples. However, the use of a nuclide fraction with a much higher Co/Cs ratio, such as that in Table 2H-1, is conservative from a dose standpoint. See EC 041-01 for additional discussion.

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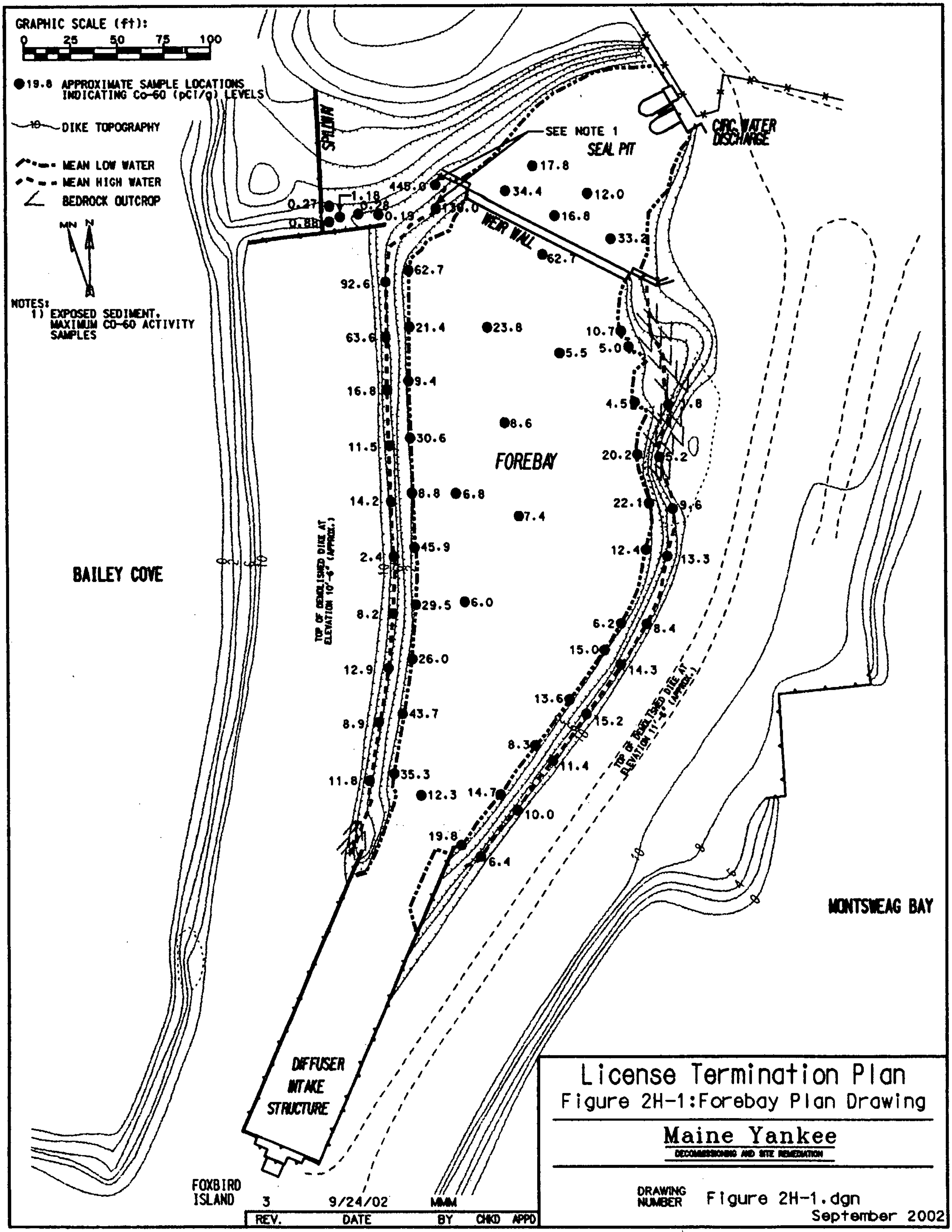
13 Seaweed and other vegetative matter in the forebay will be removed during the sediment remediation work.

Table 2H-7. Comparison of Co/Cs Ratios			
	pCi/g Co-60	pCi/g Cs-137	Co/Cs Ratio
LTP Rev. 1 forebay sediment NF (Table 2H-1)	NA	NA	18.9
Sediment around rip-rap in tidal zone (Table 2H-2)	19.7	1.1	18.0
Exposed sediment material (Section 2H-2.2a)	65.9	3.3	19.8
Underwater sediment, forebay and seal pit (Section 2H-2.3)	19.0	1.9	10.1
Dike "Soil," underneath rip-rap (Data from 2002 dike coring campaign. See Section 2.4)	0.071	0.082	0.9

The forebay dose assessment confirmed that nuclides other than Co-60 and Cs-137 represent only a small fraction of the dose contribution.

Thus, considering the overall dominance of Co-60 and Cs-137 nuclides in the dose impact, the comparable Co/Cs ratios for forebay/diffuser materials, and the effective absence of TRU nuclides, an overall evaluation of this characterization data concluded that a single nuclide fraction, determined by HTD analyses was appropriate for application to forebay/diffuser media. Further assessment and comparison of the HTD analyses concluded that the originally determined nuclide fraction, established in the LTP Rev. 1 analysis of forebay sediment, remained appropriate and conservative for dose assessment application to forebay and diffuser contaminated media. See EC 041-01 for additional detail and discussion of the data evaluation.



License Termination Plan

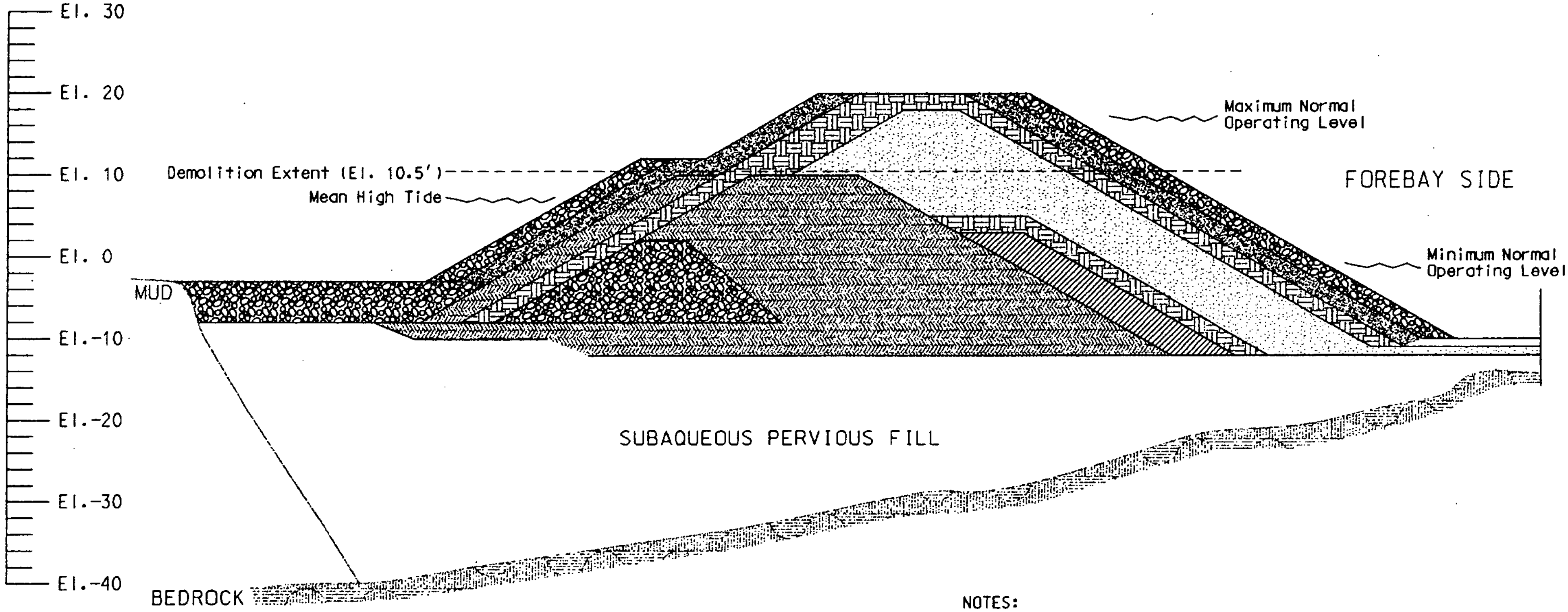
Figure 2H-1: Forebay Plan Drawing

Maine Yankee

DECOMMISSIONING AND SITE REMEDIATION

DRAWING NUMBER Figure 2H-1.dgn

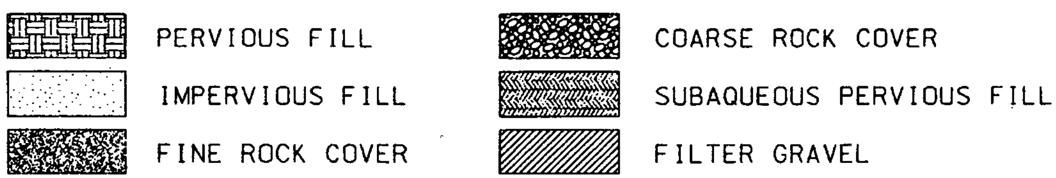
September 2002



NOTES:

This figure provides a typical forebay cross-section primarily to illustrate the "forebay side" outer layer construction arrangment. Actual interior construction and the dikes relationship to bedrock varies with location in the forebay. See engineering drawings for actual construction details.

The outer dike layers are typically 2' thick.



License Termination Plan
Fig. 2H-1: Forebay General Section

Maine Yankee

Figure 2h-2.dgn

ATTACHMENT 2I

Soil Sampling and Radionuclide Fraction

Soil Sampling and Radionuclide Fraction[1]

Introduction

Multiple soil samples representing areas of the site known to have high activity soil contamination were collected. Several samples from each area were composited to provide the most representative contaminated soil values and provide the highest probability to detect and quantify hard-to-detect (HTD) radionuclides that could be associated with the contaminated soil. Specific instructions were included for composition and analysis of these samples so as to insure the representation of the samples to be submitted for HTD vendor lab analysis. Since the final status surveys for soil include gamma spectroscopy analysis of each soil sample, the HTD data set is useful for establishing the surrogate relationship to Cs-137. These HTD nuclides (H3 and Ni-63) contribute to less than one percent of the total soil dose.

Sample Analysis

A comparison of specific soil nuclide parameters over time (1999- 2030) was made to determine how the soil Cs-137 surrogate DCGL value changes with time. The DCGL ranges from about 4.2 to 4.4 pCi/g. The change is mostly due to the fact that Co-60 decays at a faster rate than Cs-137, which results in higher allowable surrogate DCGL levels at later times. This variation with respect to time shows that the effect of conducting final status surveys significantly sooner or later than the currently proposed time is insignificant. In practice, the Co-60 will be measured by gamma spectroscopy and the only nuclides included in the Cs-137 surrogate calculations will be H-3 and Ni-63. The total dose from H-3 and Ni-63 in soil is about 0.1 mrem/y as calculated for the year 2004. Any changes in dose from these radionuclides over time will be negligible (<0.02 mrem/y) relative to the unrestricted use criteria.

Sample Selection and Composition

To determine the best representation of Industrial and Restricted Area samples the soil samples and respective locations collected during the GTS Site Characterization were examined. Emphasis was placed on samples collected from areas of principle spill or contamination incident. These areas of significance were the RWST, PWST and the Shielded Radiological Waste Storage Area (SRWSA). Examination of all other site characterization soil samples showed that these three areas contained the maximum concentration of elevated soil activity. The available GTS vendor laboratory results for some samples from these areas showed relatively high MDA values for several HTD nuclides. Any positive TRU results were at or very near MDA values and those near the MDA value did not appear in the ratios of one nuclide to another, which would be expected in power reactor TRU inventory. From these observations it was decided to composite biased samples of maximum concentration from the regions of the most significant incidents. The twelve samples that were composited for these areas originated

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[1] The soil sample analysis results and general methodology are presented in Engineering Calculation EC 013-01, Rev. 0. This calculation reviews the associated sample results and encompasses the features and nuclides associated with Engineering Calculation EC 007-00, Rev 1.

from the archived GTS site characterization soil samples and are expected to represent greater location diversity and better estimate of distribution than individual sample locations.

Inspection and composite instructions were developed in the form of a technical evaluation document so that all samples were systematically processed in the same manner and using the same methodology. Once the twelve archived samples were located, the samples were assigned a new chain-of-custody form and a minimum of one sample from each group (RWST, PWST and SRWSA) was analyzed in the original GTS retrant beakers using the Maine Yankee gamma spectroscopy system. As one of the instruction steps, the Maine Yankee spectroscopy Cs-137 analysis results were compared to the original GTS Cs-137 results and found to reasonably agree. As stated, the Maine Yankee analysis results conclude that the principle gamma emitters associated with the original GTS soil sample containers were within reasonable agreement of the concentrations reported in the GTS Characterization Report.

Following this comparison and per the composite instructions each of the GTS samples for each of the three regions (RWST, PWST and SRWSA) were thoroughly mixed and a predetermined sample mass collected of the composite representing each region. From the original Cs-137 concentration associated with each sample, the concentration per unit mass and total mass of the sample was estimated. These results were compared to the composited sample results. This comparison provides both a final check of the reported concentrations to the current analysis and insight into the distribution of associated radionuclides in the media. The narrow range of concentration variation associated with the RWST estimated and final composite values is indicative of contaminants associated with liquids where the concentration would expectedly be more uniform. The wider range of variation for the PWST and Shielded Storage areas estimated and final composite values are indicative of non-uniform contaminants and for a given sample group the range variation would represent the spatial distribution of the activity in the media. Table 2I-1 presents these findings.

Table 2I-1
Original and Composite Cs-137 Soil Concentrations and Comparison

	GTS Samples	Estimated Collective Value		Final Composite Value	
Sample Location	Cs-137 Range	Weight (g)	Cs-137 (pCi/g)	Weight (g)	Cs-137 (pCi/g)
RWST	11.0 -114.0	1440.0	61.6	1475.0	60.5
PWST	14.6 - 156.0	1500.0	86.1	1532.0	99.4
Shielded Storage	18.3	800.0	18.3	1023.0	22.1
All RWST samples represent surface soil; Two PWST soil samples (Cs-137 ranging from 14.6 - 57.6 pCi/g) represent soil at 6-18 inch depths. Three PWST sample represent Cs-137 surface soil ranging from 69.1-156 pCi/g. The single Shielded Storage sample is surface soil.					

Additional Confirmatory Sample Collection

The continued characterization soil samples (442 samples) collected in 1999 and 2001 support the observed composite results obtained from the samples associated with the RWST, PWST and SRWSA.

These investigations focused primarily in the Industrial and Restricted Area of the site. Only Co-60 and Cs-137 were identified in the 442 samples. These samples represented both surface and subsurface investigations to a depth of nearly 4 meters (~12 feet). The concentration range of all these samples was significantly lower than the samples used for the soil profile provided in EC-013-01 (See Table 2I-2 below).

A total of 442 samples from 107 locations were collected and analyzed. The sample analysis results (442 samples) showed that Cs-137 was reported at >MDA 35.5 percent of the time while Co-60 was reported at >MDA only 2.0 percent of the time. The results of these samples provide additional support for Cs-137's predominate presence in contaminated site soils. DCGL values show that the surrogate DCGL changes little over time (~2.6% from 2004 to 2030). The maximum observed soil concentrations for Cs-137 and Co-60 in the 1999 and 2001 sample results (442 samples) were considerable lower (Cs-137: 34.7 and Co-60 12.4 pCi/g) than the composited samples used to determine the HTD soil constituents. These results indicate that the analyzed composites conservatively address the HTD and gamma emitters associated with the site soils. Of the 422 samples 79 were determined to exceed the action level estimated for the sampling plan.

Table 2I-2 presents the range of Co-60 and Cs-137 for the 79 samples that were found to exceed the sample plan respective Action Levels of 1.0 and 3.1 pCi/g for Co-60 and Cs-137. It is important to note that for the Co-60 data in Table 2I-2 only seven Co-60 sample results are above the MDC for the analysis parameters (The reported Co-60 MDC's for the remaining 72 samples ranged from 0.05 to 0.40 pCi/g). For the Cs-137 data in Table 2I-2 a total of 58 (73.4%) of the 79 samples are above the MDC for the analysis parameters (The 21 Cs-137 samples less than the MDC ranged from 0.06 to 0.41 pCi/g). The results of Table 2I-2 show that none of the 442 additional samples collected approached the soil concentrations reported for the RWST and PWST composite samples. As previously stated the radionuclide results of the RWST and PWST samples conservatively characterize the Maine Yankee site soils.

Table 2I-2
Cs-137 and Co-60 Range for Continued Characterization ≥Action Level*

Cs-137 Range (pCi/g)	Number of Observations	Co-60 Range (pCi/g)	Number of Observations
>0.34 -2.0	33	>0.06 - 0.50	36
>2.0 - 5.0	22	>0.50 - 1.0	27
>5.0 - 10.0	15	>1.0 - 2.0	15
>10.0 - 20	3	12.4	1
>20 - 34.7	6		
Total	79	Total	79

*Sample Plan Action Level 1.0 and 3.1 pCi/g for Co-60 and Cs-137 respectively.

Summary

- The soil characterization by GTS and sample locations throughout the Restricted Area (RA) and Industrial Area (IA) were reviewed. Sample locations were selected that reflected locations of historic primary contamination incidents and highest soil contamination.
- The concentrations of the selected samples increase the probability of detecting and quantifying HTD nuclides.
- The composite method used resulted in composite soil concentrations conservatively higher than any of the GTS characterizations soil samples and the 442 continued characterization samples acquired in the RA and IA in1999 and 2001.
- All FSS soil samples will be analyzed using gamma spectroscopy.
- The 442 continued characterization soil samples collected in 1999 and 2001 support the composite results.
- The Cs-137 surrogate DCGL for soil varies no more than 2.6% from 2004 through 2030 (~4.0% from 1999 through 2030).

MAINE YANKEE

LTP SECTION 3

IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

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3.0 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction

3.1.1 Purpose

This section of the LTP describes the remaining dismantlement activities at MY pursuant to 10 CFR 50.82(a)(9)(ii)(B) and following the guidance of NUREG 1700 and Regulatory Guide 1.179. Information is presented to demonstrate that these activities will be performed in accordance with 10 CFR Part 50 and will not be inimical to the common defense and security or to the health and safety of the public pursuant to 10 CFR 50.82(a)(10). Information which demonstrates that these activities will not have a significant effect on the quality of the environment is provided in LTP Section 8.

The dismantlement activities described in this section provide the NRC the information to support their determination to terminate the license pursuant to 10 CFR 50.82(a)(11)(i). Therefore, this section was written to clearly indicate each dismantlement activity which remains to be completed prior to qualifying for license termination. Furthermore, information is provided on the final state of the site including structural remnants, basement foundations and buried piping and conduits. This information ensures that the scope of any possible residual contaminated materials associated with the final state of the site are considered in dose modeling, survey design and environmental assessment. Any changes to the dismantlement activities described in this section which are made pursuant to 10 CFR 50.59 must also consider the impact of those changes on the final state of the site and any impacts on dose assessment, survey design or environmental assessment.

Information related to the remaining decontamination and dismantlement tasks is also provided. This information includes an estimate of the quantity of radioactive material to be released to unrestricted areas, a description of proposed control mechanisms to ensure areas are not recontaminated, estimates of occupational exposures, and characterization of radiological conditions to be encountered and the types and quantities of radioactive waste. This information supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify inspection or technical resources needed during the remaining dismantlement activities. Many of these dismantlement tasks require coordination with other federal, state or local regulatory agencies or groups. Maine Yankee's coordination with these agencies and groups is generally described.

An evaluation of the remaining decontamination and dismantlement activities is described in this section. This evaluation presents summary supporting justification for the conclusion that, pursuant to 10 CFR 50.59, activities may be conducted without obtaining a license amendment pursuant 10 CFR 50.90. Where activities require Maine Yankee to obtain a license amendment, such activities are identified along with the corresponding schedule for the proposed license amendment and the schedule for needed approval.

3.1.2 Decommissioning Progress Update

Shortly after the submittal of the 10 CFR 50.82(a)(1) certifications, Maine Yankee assembled a System Evaluation Review Team (SERT) to evaluate each plant system, structure and component (SSC) against applicable regulatory and design basis requirements. These evaluations resulted in the classification of SSCs as available and/or abandoned. Applicable systems were drained, de-energized and deactivated as appropriate for turnover to the Decommissioning Operations Contractor (DOC). The reactor coolant system was chemically decontaminated to reduce source term in preparation for dismantlement.

Systems and functions required to support the safe storage of spent fuel were redesigned, as necessary and consolidated into the Spent Fuel Pool Island (SFPI). Electrical power was provided from the 115KV incoming line with a back up diesel generator specifically for security, but available for the SFPI. An industrial water-to-air cooling system replaced the primary component cooling /service water systems that serviced the spent fuel pool cooling and clean up system. Makeup water is supplied from the PWST with back up from the Wiscasset water supply and the fire protection service system. A portable mix tank and pump batches borated water when required in the make up for the spent fuel pool.

During the fall of 1997 and spring of 1998, Maine Yankee conducted a radiological characterization of the site through GTS Duratek. Appropriate historical information was compiled into the Historical Site Assessment (HSA). This site characterization, which is summarized in LTP Section 2, was conducted to assist companies bidding for a contract to decommission the site with additional characterization to be conducted as necessary thereafter. During the fall of 1998, Maine Yankee reviewed bids and selected Stone & Webster as the DOC. Under Maine Yankee oversight, Stone & Webster conducted various decontamination and dismantlement activities until May 2000 when the contract was cancelled.

The overall project schedule defines the current status and remaining activities. Four phases of site dismantlement, some of which run in parallel, were defined by Stone & Webster's contract. As part of preparing the site, Phase 1 removed structures to increase the free area needed for large vehicles and equipment. The removal work has been completed and involved the removal of guard towers, some tanks and other structures. Efforts to release non-impacted areas are ongoing. Phase 2 initiated activities for commodity removal, dismantlement and structure decontamination. This phase is currently in-progress. Phase 3 consists primarily of demolition activities as well as site restoration activities. Phase 4 consists of the construction of an Independent Spent Fuel Storage Installation (ISFSI) and movement of spent fuel to dry storage.

The construction of the ISFSI has been completed, and movement of spent fuel was completed in the first quarter of 2004. In preparation for constructing the ISFSI, final status surveys of the land area and the ISFSI Security Operation Building (SOB), formerly the Low Level Waste Storage Building (LLWSB), were initiated in the fall of 1999 through summer of 2000. In preparation for fuel transfer, Maine Yankee conducted a complete inventory and inspection of the contents of the spent fuel pool during 2000.

Some major decommissioning activities have been completed and others are in-progress. Reactor coolant system piping, reactor coolant pumps and motors, steam generators and the pressurizer have been removed and shipped offsite for processing and/or waste disposal as appropriate. Other small commodities have also been removed and shipped offsite. Reactor vessel internals were segmented using an abrasive water jet (AWJ) system. Greater-than-class-C (GTCC) waste generated as a result of the segmentation project were loaded into NAC UMS casks and stored onsite at the ISFSI. During 2003 and 2004, remaining above grade structures in the Industrial Area were surveyed and demolished including the Spray Building, Primary Auxiliary Building, Fuel Building and Containment Building.

On January 3, 2001, Maine Yankee submitted an application to amend the license to release a portion of the site classified as non-impacted. This application provides the NRC with the information specified in LTP Section 1.4.2. This land area contains a few structures including the Eaton farmhouse. While some non-radiological remediation was conducted on the farmhouse, no dismantlement activities are required to be completed prior to removing this land area from the jurisdiction of the Part 50 license as requested in the proposed license amendment On April 10, 2001, Maine Yankee submitted a second application to amend the license to release an additional portion of the site classified as non-impacted. On August 16, 2001, Maine Yankee resubmitted its application to release these lands, combining the previous two applications into one application and revising the presentation of the characterization data and results. Statistical analyses were presented to demonstrate that the residual activity, if any, in these lands is indistinguishable from background. On November 19, 2001, Maine Yankee supplemented its combined application, making certain clarifications including land survey information. The NRC granted this request for the release of these lands in July 2002. See Section 1.4.2.

3.1.3 Decontamination & Dismantlement Process Summary

Decontamination & dismantlement activities will be supported by detailed project planning and scheduling. This planning supports as low as reasonably achievable (ALARA) reviews, estimation of labor and resource requirements, while tracking cost and schedule. Work packages are used to implement the detailed plans and provide instructions for actual field implementation. The work packages address described units of work and include appropriate hold and inspection points. Administrative procedures control work package format and content, as well as the review and approval process.

Systems and components removed and released from the secondary side of the plant for commercial disposal are surveyed in accordance with plant procedures based upon a no detectible radioactivity standard. The controlling procedure specifies that the instrumentation must be capable of detecting beta/gamma (and alpha if suspected) radioactivity[1] at or below the levels listed below:

a. Total surface beta/gamma contamination @5000 dpm/100 cm^2
b. Loose surface beta/gamma contamination @1000dpm/100 cm^2
c. Fixed alpha contamination @100 dpm/100 cm^2
d. Loose surface alpha contamination @ 20 dpm/100 cm^2
e. Gamma dose rates of 10 micro rem/hr

This procedure requires that material be evaluated for probability of radioactive contamination by utilizing "knowledge of process" which may include review of surveys, the historical site assessment, characterization surveys or other knowledge of material history. Survey requirements are increased according to the greater probability for contamination. For instance, materials with no probability for contamination are subject to an aggregate dose rate survey and a validation survey which includes a truck monitor or acceptable alternative. Materials with a low probability for contamination are subject to a biased direct frisk (typically approx. 10% of surface area) and loose surface contamination survey prior to packaging as well as the above requirements for aggregate and validation survey following packaging. Materials with a high probability for contamination are subject to the above requirements for low probability materials however the entire accessible surface area of the materials are subject to a direct frisk prior to packaging. Additional or alternative survey requirements are also specified for special situations including volumetric materials, difficult to survey items, systems or components and samples.

A separate procedure has been implemented with the same detection levels, augmented by additional controls for the release of material from the radiologically restricted area. Generally, systems and components removed from the primary (radiologically controlled) side of the plant are packaged and either transported to an offsite processing facility, a low-level radioactive waste (LLRW) disposal facility, or an appropriate disposal facility.

Decontamination of structures will include a variety of techniques ranging from water washing to surface material removal. Structural material may be packaged and either transported to an offsite processing facility, a LLRW disposal facility, or an appropriate disposal facility.

Following the removal or decontamination of systems, components, and structures, a comprehensive final status survey (FSS) will be completed as described in Section 5 of this LTP.

As referred to above, the dismantlement activities will be carried out in the following four phases:

[1] In accordance with NRC Circular 81-07 and IE Information Notice No. 85-92

Phase 1: Prepare Site & Release Non-Impacted Areas
Phase 2: Dismantle Commodities & Decontaminate Structures
Phase 3: Demolish Buildings & Restore Site
Phase 4: Establish Independent Spent Fuel Storage Installation (ISFSI)

These phases may be implemented in parallel and are not necessarily sequential. A brief discussion of the four phases follows:

Phase 1: Prepare Site & Release Non-Impacted Areas

The preparations period began with permanent plant closure on August 7, 1997.

This phase involved the demolition of miscellaneous tanks, buildings, fences and vehicle barriers, etc. to allow ease of access to the site. During this phase, as demonstrated by this LTP, no radiological contaminants were found North of Old Ferry road, or West of Bailey Cove, and these areas are therefore designated and expected to be released on an early basis in accordance with 10 CFR Part 20 Subpart E (Radiological Criteria for Unrestricted Use), the enhanced state clean-up standards, and 10 CFR 50.82 (a)(11)(i) and (ii).

This phase also included site characterization activity, license basis document revision, spent fuel pool island construction and system evaluation, re-classification and, as appropriate, deactivation as described above.

Phase 2: Dismantle Commodities & Decontaminate Structures

Commodities are dismantled and removed during this phase. Following commodity removal, applicable portions of structures are decontaminated as necessary. Maine Yankee intends to demolish structures, with few exceptions, down to three feet below grade. For structures on the secondary side of the plant, sufficient surveys are conducted prior to demolition to ensure that any applicable portions of the structure are decontaminated. For structures on the primary (radiologically controlled) side of the plant, those portions of the structure above three feet below grade will generally be demolished, packaged and either transported to a LLRW disposal facility or an alternate disposal facility. Some metals, such as rebar, may be recycled, as appropriate, if the metals can be released using a no detectable radioactivity standard. Basement surfaces below three feet below grade will be decontaminated and remediated (paint removal, chemical stain removal, etc.) as necessary and a final status survey will be performed before the basement is filled with soil.

Phase 3: Demolish Buildings & Restore Site

During this phase, structures will be demolished to an elevation corresponding to three feet below grade. These demolition activities will be reviewed during planning to ensure no adverse effect on the SFPI (i.e. walls of adjacent buildings that have a support function of the SFP will remain intact). Concrete buildings will be demolished to 3 foot below grade. Other buildings are designated for either industrial reuse, recycling, or offsite disposal; and are dispositioned accordingly.

Activated portions of remaining foundations above the Activated Concrete DCGLs will be removed.

Several options exist for sequencing building demolition activities with FSS. These demolition sequences will be evaluated and selected with the objective of minimizing the potential for recontamination of surfaces that have already received a final status survey and maximizing the quality of the final status survey. For all options, a final status survey will be performed on the basement surfaces before fill material is placed and on the remaining building footprint after fill material is placed. The status of dismantlement, remediation and FSS activities will be frequently communicated to state and NRC authorities to ensure adequate time for confirmatory measurements, if necessary, prior to the basement being filled. Listed below are some options for sequencing building demolition activities with FSS.

Option 1 Complete Building Demolition prior to FSS:
Will require special attention to keeping the elements (weather) out of the basement during FSS and confirmatory measurements

Option 2 FSS prior to Building Demolition:
Will require special attention for preventing the building demolition from re-contaminating surfaces below where FSS has been completed.

Option 3 Building Demolition not including a floor above grade (nominally 21 ft.). This option would:

- Seal openings in the upper floor to act as a roof.
- Perform FSS on basement floors, walls and ceilings (if applicable).
- Allow MY to notify NRC and State of their opportunity to perform confirmatory measurements.
- Fill basement areas with soil fill material.
- Complete the demolition of the building down to the 17 ft. elevation.
- Perform confirmatory measurements on the surface of the soil fill material after super structure demolition.

This option would require actions to:

♦ Keep the elements out of foundation area which do not have a 21 ft floor (e.g. SFP portion of the Fuel Building).
♦ Prevent recontamination pathways to the basement during

above 21 ft building demolition and during basement soil fill operations.

Other demolition sequencing options may be developed, as necessary, to achieve the objectives described above.

Phase 4: Establish an Independent Spent Fuel Storage Installation (ISFSI)

The ISFSI is designed, constructed and loaded with fuel stored in casks during this Phase. Maine Yankee's storage of spent fuel in the ISFSI will be conducted under a general Part 72 license pursuant to 10 CFR Part 72, Subpart K. Therefore, Maine Yankee will store fuel only in fuel casks approved by the NRC as listed in 10 CFR 72.214.

Following complete transfer of the spent fuel from the spent fuel pool to the ISFSI, Maine Yankee will dismantle and demolish the spent fuel pool. Maine Yankee has submitted a license amendment, pursuant to 10 CFR 50.90, to add an applicability statement to certain technical specifications that describe requirements associated with the spent fuel pool. This license amendment has been approved prior to demolition of the spent fuel pool.

3.2 Remaining Dismantlement Activities

The purpose of this section of the LTP is to indicate each dismantlement activity which remains to be completed prior to qualifying for license termination. This information is provided to support the NRC in making their determination to terminate the license pursuant to 10 CFR 50.82(a)(11)(i). In addition to identifying the dismantlement activities, information is provided on the final state of the site including structural remnants, basement foundations and buried piping and conduits. This information ensures that the scope of possible contaminated materials associated with the final state of the site are considered in dose modeling, survey design and environmental assessment. Any changes to the dismantlement activities described in this section which are made pursuant to 10 CFR 50.59 must also consider the impact of those changes on the final state of the site and any impacts on dose assessment, survey design or environmental assessment.

3.2.1 Major Decommissioning Activities

10 CFR 50.2 defines "major decommissioning activity" as any activity that results in permanent removal of major radioactive components, permanently modifies the structure of the containment, or results in dismantling components (separating and packaging GTCC waste) for shipment in accordance with 10 CFR 61.55.

Those activities are summarized as follows:

a. Removal of the steam generators and the pressurizer. The external surfaces decontaminated as required, and all openings sealed-welded. These components serve as their own transport containers. This activity was completed in 2000.

b. The reactor internals have been segmented such that the components with the lowest activity (upper guide structure and the uppermost and lowermost portions of the core support barrel assembly) will be shipped in the RPV,

c. The segments with intermediate levels of activity (the center section of the core support barrel assembly) will be shipped in casks for disposal in a near surface disposal site, and

d. The segments that exceed class C limits (the core support plate and the core shroud) are stored on site for later transport with the spent fuel to a USDOE disposal facility.

e. Remove the RPV and place it into transport/disposal container, for shipment and disposal intact.

f. Segment the neutron shield tank structure formerly surrounding the reactor vessel, and place the segments into shielded containers.

g. Segment the RCS and other large-bore piping, decontaminate to acceptable limits, if necessary, for offsite direct disposal, size reduction/disposal, or offsite recycle as appropriate considering the residual activity level. This was mostly completed in 2000.

h. The containment equipment access was modified (with closure capability) to facilitate moving a multi-wheeled transporter into containment for loading/ removal of large components. This task was completed.

i. Once all spent fuel is removed from the spent fuel pool, the spent fuel facility will be decontaminated and dismantled.

The containment polar crane, and/or a crane set-up inside containment loads each large component onto a multi-wheeled transporter for removal through the modified containment equipment hatch. The transporter moves the component(s) to the designated preparation/ temporary storage area within the industrial area. Reactor coolant pumps and motors were shipped via truck, and rail. Rail or barge will be used for the reactor vessel head for transport offsite. The large components such as RPV, pressurizer, and steam generators were or will be transferred via the multi-wheeled transporter onto barges for shipment either directly to the disposal facility (as in the case of the reactor vessel), or to an offsite facility for additional decontamination and/or volume reduction prior to final disposal or recycle (for the other components).

During 1999 and 2000, Maine Yankee removed containment piping and many components. Reactor vessel internals and the reactor vessel itself (with some internals) were processed and removed, in 2001 and 2002, from the containment building and stored for later shipment . The reactor vessel was loaded into a transport/disposal container (DOT approved). The vessel and its container were moved onto a sea-going barge and transported via the Atlantic Ocean, Intercoastal Waterway, then up the Savannah River where it was offloaded at the Savannah River Site. After barge offloading, the vessel package was transported overland for disposal at the GTS DURATEK low level radioactive waste disposal facility, near Barnwell, South Carolina. These activities were coordinated with the State of Maine, US DOT, US Coast Guard, NRC, South Carolina Department of Health and Environmental Control (SCDHEC), and the other States requiring notification of the shipment.

3.2.2 Dismantlement Activity Schedule

In relation to plant commodities and internal structures, the project's approach to dismantlement is to expeditiously remove these items, and transport for processing or disposal. It provides a safe, productive, and cost-effective means for commodity removal and accelerates access to the building surfaces for decontamination efforts.

NSSS component removal should be completed approximately three and one half years following cessation of operations. It is expected that the majority of plant structures and facilities will be decontaminated and dismantled within seven years of cessation of operations as listed below in Table 3-1.

The few facilities and structures required to support the ISFSI (spent fuel and GTCC waste storage) will be decontaminated, as necessary, and dismantled after USDOE has removed the stored materials.

Table 3-1 Major MY Area/Systems, Structures, and Components Removed (By Year)	
2001	
	PZR quench tank
	Rx vessel head
	Rx vessel internals
	Regenerative heat exchanger
	SI tank #2
	Evaporators
	Turbine steam system
	Circulating Water Pumphouse
	Circulating water system
	Primary drain tank
	Neutron shield tank
	Rx pressure vessel
	Turbine building (Phase I)
2002	
	Spray Building
	Steam and Valve House
	Primary Vent Stack
	115 KV yard limited (Switchyard remains)
2003	
	Emergency Feedwater
	Spent fuel pool (limited)
	Boron waste storage tank (BWST)
	Primary water system
	Fire protection system limited
	Spent fuel pool

Table 3-1 Major MY Area/Systems, Structures, and Components Removed (By Year)
Spent fuel building
Fire Protection System
Primary Auxiliary Building
2004
PW - potable water connection (plant site)
Forebay
Containment
2026 (or after DOE removes the stored materials)
ISFSI site D&D with remediation as required

The remaining decommissioning schedule represented in Table 3-2 will be revised during the project. However, the LTP does not require revision to describe the schedule changes since this section is a general description of D&D activities and options. Existing lines of communication (i.e. weekly telecom) will be utilized to inform the NRC of any significant changes to major milestones in the schedule.

Equipment and materials will be removed from areas unless the radiation surveys indicate that the structures can be released for unrestricted access and conventional demolition. By the Winter of 2002, after the fuel is transferred to the ISFSI, the SFP and its supporting systems are scheduled for D&D.

Table 3-2
Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)

Activity Number		Activity Description	Completion Date
TG0102-2	TG01	Demo Main Generator	2Q 01
Z-0139	SBHP	HP Checkpoint / Chem Lab Commodity Ripout	2Q 01
Z-0141	SB01	SBP Service Building Proper	2Q 01

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
SPRB01-8	HV7	HV-7 & 9 Room Commodity Ripout	2Q 01
0194X	P21E	P21E - PAB El 21' Evap Cubicle Segment EV-2	2Q 01
PULG01-1	PULG	PAB 36' General Area Commodity Ripout	2Q 01
PP-2 DWST	DWST	Foundation Pkg./Proc. C&D (Subgrade)	2Q 01
CB1501-5	CBQT	Quench Tank Area -2' Commodity Removal	2Q 01
0192	P21H	PAB 21' Letdown HX Room Commodity Removal	2Q 01
PULG01-4	PULG	PAB 36' - Boric Acid Storage Tank Demolition	2Q 01
0130	SPRB	Spray Building Commodity Ripout	2Q 01
PUDD01-1	PUDD	Remove Block Wall from Decay Drum Cube Opening	2Q 01
SPRB01-10	SPRB	Spray Pump Rebuild Room Commodity Removal	2Q 01
PUDDO1-3	PUDD	PAB 36' Remove Waste Gas Surge Tank TK-10	2Q 01
PUDDO1-4	PUDD	PAB 36' Remove Decay Drum Tank 60A	2Q 01
PUDD01-5	PUDD	PAB 36' Remove Decay Drum Tank 60B	2Q 01
PUDD01-6	PUDD	PAB 36' Remove Decay Drum Tank 60C	2Q 01
0144X	CB14	SIT #2 Cut-Up	2Q 01
PUDD01-7	PUDD	PAB 36' Remove Decay Drum Tank 60D	2Q 01
PUDD01-8	PUDD	PAB 36' Remove Decay Drum Tank 60E	2Q 01
0245	XS00	Remove/Rig out Spare Transformer (X-1S)	2Q 01
TB0006-7X	TB00	Turbine Building - ROOFING REMOVAL	3Q 01
PERH	PH	Personnel Hatch Area Commodity Ripout	3Q 01
TB0006-7	TB00	Turb. Bldg. - Galbestos Siding Removal	3Q 01

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
0141X	CB14	Regen HX Commodity Removal	3Q 01
	CB11	Demo RCP #1 Pump Pedestal	3Q 01
PUDD01-2	PUDD	PAB 36' Decay Drum Cubicle Comm Rmvl	3Q 01
0053	CICI	CTMT ICI Sump Commodity Removal	3Q 01
0083X	CB13	SIT #3 Cut-Up	3Q 01
Z-0156P	SVH1	Steam & Valve House Commodity Removal	3Q 01
RCP2PED	CB12	Demo RCP #2 Pump Pedestal	3Q 01
SG1PED	CB11	Demo SG #1 Pedestal Base	3Q 01
PUTC01-4	PUTC	PAB 36' VCT Cubicle Commodity Ripout	3Q 01
P21L01-2	P21L	PAB 21' General Area Commodity Removal	3Q 01
CB1501-11	CBQT	Quench Tank Area Final Ripout	3Q 01
RCP3PED	CB13	Demo RCP #3 Pump Pedestal	3Q 01
PUTC01-2	PUTC	PAB 36' VCT Cubicle - Remove VCT TK-6	3Q 01
SG3PED	CB13	Demo SG #3 Pedestal Base	3Q 01
Z-0157P	RMCC	Reactor MCC Room Commodity Removal	3Q 01
CB1201-7	CB12	CTMT Loop 2 Final Rip Out	3Q 01
LOOP1C	CB11	CTMT Loop 1 Commodity Removal Complete	3Q 01
LOOP1F	CB11	CTMT Loop 1 Final Ripout	3Q 01
0211	PLAD	PLAD PAB 11' ADT - Segment TK-12A	3Q 01
PUWG01-1	PUWG	PAB 36' Waste Gas Cube Commodity Removal	3Q 01
0147	CB31	Remove Commodities/Cut Liners Rx Cavity Area	3Q 01

Table 3-2
Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)

Activity Number		Activity Description	Completion Date
CW0006-7	CCW1	Circ Water Demo- Walls & Roof El. 21'	3Q 01
WH5BF	MSW5	Warehouse #5 Subgrade Foundation Backfill	3Q 01
Z-0397Y	345K	Electrical Tower Demolition (X-1A/B Area)	3Q 01
Z-0397S	345K	Electrical Tower Subgrade Demo (B parking lot)	3Q 01
Z-0377	MSCL	Collection Site Demolition	3Q 01
Z-0447	YMET	Demo - MET (fences, house, tower, conc pads)	3Q 01
Z-0357	YDWW	Well Water House Demolition	3Q 01
Z-0357S	YDWW	Well Water House Subgrade Demolition	3Q 01
TB0006-20	TB00	Turbine Hall Crane Demolition	4Q 01
0066X	CB11	SIT #1 Cut-Up	4Q 01
P21E01-2	P21E	P21E El 21' Evap Cubicle Grating	4Q 01
CB1301-8	CB13	CTMT Loop 3 Final Ripout	4Q 01
LOOP3C	CB13	CTMT Loop 3 Commodity Removal Complete	4Q 01
TB0006-21	TG01	Turbine Pedestal	4Q 01
TB0006-22	TB00	Turbine Building Demo (Ph I - Excl. North Wall)	4Q 01
Z-0152X	SB00	Service Building Demo (Phase I) (Above Grade)	4Q 01
CW0006-8	CCW1	Circ Water Pump House Demo - Slab @ 21'	4Q 01
PLAD01-2	PLAD	PLAD PAB 11' ADT - Segment TK-12B	4Q 01
CW0006-10	CCW1	Circ Water Pump House Demolition Retaining Walls	4Q 01
0226	TK33	Remove CWPH Ferrous Sulfate Storage Tk (TK-33)	4Q 01
CW0006-2	CCW1	Circ Water Pump Shaft Backfill	4Q 01

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
CW0006-9	CCW1	CW Pump House Demo Structure to High Water	4Q 01
CW0006-11	CCW1	Circ Water Pump House Concrete Demolition	4Q 01
0714	STP1	Sewage Treatment Plant Demolition	4Q 01
TB1BF	TB00	Turbine Building Demo Phase 1 Backfill	4Q 01
PUEC01-1	PUEC	PAB 36' Evap Cubicle Commodity Removal	4Q 01
STPBF	STP1	Sewage Treatment Plant Demolition Backfill	4Q 01
PUHV01-1	PUHV	PAB El 36' HVAC Rmv Comp Below HV-1,2	4Q 01
PUHV01-2	PUHV	PAB El 36' HVAC Remove Filters	4Q 01
CWBF	CCW1	CW Pump House Demo Backfill & Landscape	4Q 01
PUHV01-3	PUHV	Remove Tubing, Piping, Conduit & HVAC Duct	4Q 01
PUHV01-4	PUHV	PAB El 36' HVAC Remove Walkway	4Q 01
0311	PLPD	PAB 11' PDT - Segment TK-11 (Primary Drain)	4Q 01
TB0006BG	TB00	Turb Bldg Foundation Subgrade Demo	4Q 01
Z-0152XS	SB00	Service Building Subgrade Demolition (Phase 1)	4Q 01
0114	NSRV	Segment Neutron Shield Tank	1Q 02
PLPA01-2	PLPA	PAB 11' - PAB Sump Area Commodity Removal	1Q 02
0153	CCG1	CTMT 46' Charging Floor Commodity	1Q 02
0150	CB32	CTMT Upender - Remove Commodities/Cut Liners	1Q 02
CB3301-3	CCOA	Remove CTMT 46' Outer Annulus Jib Crane	1Q 02
0156	CPLE	Remove CTMT Elevator Room Commodities 1	1Q 02
P21L01-5	P21L	P21L - PAB El 21' Remove Monorail	1Q 02

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
CB3301-2	CCOA	Remove CTMT 46' Outer Annulus Monorail	1Q 02
Z-0337	YDPH	MPH - Montsweag - Remove Commodities & Backfill	1Q 02
Z-0397	345K	Electrical Tower Demolition (B Parking Lot)	1Q 02
CCOAF	CCOA	CTMT 46' Outer Annulus Final Ripout	1Q 02
HV7DEMO	HV7	HV-7 & 9 Building Demolition	1Q 02
0713S	SBTT	Service Bldg Test Tank Subgrade Foundation Demo	2Q 02
CB18F	CB18	CTMT EL -2' OA Final Ripout	2Q 02
0726S	DWST	DWST -Subgrade Foundation Demolition	2Q 02
0731	X MSGH	Demo Gas House Foundation	2Q 02
SBTTBF	SBTT	SBTT Subgrade Foundation Demo Backfill	2Q 02
0727S	RWST	RWST/SCAT Tank Subgrade Foundation Demo	2Q 02
DWSTBF	DWST	DWST Subgrade Foundation Demo Backfill	2Q 02
GHBF	MSGH	Gas House Foundation Backfill	2Q 02
PP-4	RWST	RWST/SCAT Foundation Packaging/Processing C&D	2Q 02
RWSTBF	RWST	RWST/SCAT Subgrade Foundation Demo Backfill	2Q 02
0569	SPRB	Spray Building Demolition	2Q 02
SBFD	SBF	Softball Field Demolition (Dugouts & Fencing)	2Q 02
SPRAY	CCG1	Remove CTMT Spray Rings	2Q 02
0719	YDBG	Barge Slip / Road Demolition	2Q 02

Table 3-2
Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)

Activity Number		Activity Description	Completion Date
0159	CB34	Remove Commodities CTMT Polar Crane (CR-1)	2Q 02
Z-0156	SVH1	Steam Valve House Demolition (Above Grade)	2Q 02
CB2101F	CB21	Final Commodity Removal CTMT El 20'	3Q 02
0136	CPHO	CTMT Personnel Hatch Commodity Removal	3Q 02
CB3301-1	CCOA	Remove CTMT 46' Outer Annulus Grating	3Q 02
0712	CPHO	CTMT Personnel Hatch Demolition	3Q 02
CCFFR	CCG1	CTMT 46' Charging Floor Final Ripout	3Q 02
0233DR	PWST	Drain Primary Water System	3Q 02
PUFN01-2	PUFN	PAB El 36' Fan FN-1A Removal	3Q 02
Z-0152	SB00	Service Building Demo (Phase 2) Above Grade	3Q 02
PUFN01-3	PUFN	PAB El 36' Fan FN-1B Removal	3Q 02
PAB11FR	PA00	PAB 11' Final Ripout	3Q 02
HV7DEMOS	HV7	HV-7 & 9 Subgrade Demolition	3Q 02
PUFN01-4	PUFN	PAB El 36' Install Temporary Roof	3Q 02
Z-0132	RCAD	RCA Drumming Room Commodity Removal	3Q 02
Z-0135	SFP1	Fuel Bldg Proper Commodities Ripout	3Q 02
Z-0156S	SVH1	Steam Valve House Subgrade Foundation Demo	3Q 02
Z-0157	RMCC	Reactor MCC Room Demolition	3Q 02
Z-0197	AD00	Admin Bldg / Gatehouse Demolition	4Q 02
0513	PWST	PWST - Primary Water Storage Tank Comm Rmvl	4Q 02
Z-0197S	AD00	Admin Bldg / Gatehouse Subgrade Demo	4Q 02

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		Activity Description	Completion Date
LSACOM	LSAB	LSA Building Commodity Ripout	4Q 02
0711	CEHO	Containment Equipment Hatch Demolition	4Q 02
PUFN01-1	PUFN	PAB El 36' Cut Opening in Roof	4Q 02
ADBF	AD00	Admin Bldg / Gatehouse Demolition Backfill	4Q 02
Z-0152S	SB00	Serv Bldg Subgrade Demo (Phase 2) Below Grade	4Q 02
PAB21FR	PA00	PAB 21' Final Ripout	4Q 02
TB0006-22X	TB00	Turbine Building Demo (Ph II North Wall)	4Q 02
0215	PABR	PAB Roof Area Commodity Removal	4Q 02
0728	WART	Wart Building Demolition (Above Grade)	4Q 02
PUFN01-5	PUFN	PAB El 36' Fan FN-1A/B Area Commodities	4Q 02
PU4801-1	PU48	PAB El 36' Remove Duct	4Q 02
PU4801-2	PU48	PAB El 36' Remove Supports	4Q 02
Z-0131	SFP3	SFP Heat Exchanger Cubicle Commodity Removal	4Q 02
0253	SFP2	Spent Fuel Pool Area Commodity Removal	4Q 02
MYM16	SB00	Service Building Demolition Complete	4Q 02
TB0006BG2	TB00	Turb Bldg Subgrade Demo (Below North Wall)	4Q 02
0728S	WART	Wart Building Subgrade Demolition	4Q 02
0569S	SPRB	Spray Building Subgrade Demolition	4Q 02
PERHD	PH	Personel Hatch Area Demolition	4Q 02
Z-0071	NFLA	NFLA / Vault Commodity Demolition	4Q 02
PU4801-3	PU48	PAB El 36' Fan FN-48 Area Commodities	4Q 02

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
0230	X140	XFMRS X-14/16 Area Commodity Removal	4Q 02
Z-0140	RCAW	RCA Bldg Commodity Removal	4Q 02
PERHDS	PH	Personnel Hatch Area Subgrade Demolition	4Q 02
Z-0137	SFP4	SFP Ventilation Room Commodity Removal	4Q 02
MONO	PUSA	PAB 36' Monorail Demolition	4Q 02
0738	X140	X-14/16/16A/16B Transformer Demolition	4Q 02
TB2BF	TB00	Turbine Building Demo Phase 2 Backfill	4Q 02
PT001	SFP6	RCA Pipe Tunnel Commodity Removal	4Q 02
PAB36FR	PA00	PAB 36' Final Ripout	4Q 02
0245FD	XALL	Transformer Foundations Demolition (All)	4Q 02
0236	115KV	115KV Tower Dismantlement (X-14 area)	1Q 03
Z-0207	CB00	Primary Vent Stack Demolition	1Q 03
0716	115KV	115KV Tower Subgrade Foundation Demo (X-14 Area)	1Q 03
0254	SFP2	Spent Fuel Pool - Cut Liner/Fuel Racks	1Q 03
0222COM	HR00	High Rad Bunker Commodity Ripout	1Q 03
0231	SFPI	Generator/Pagoda Area Commodity Removal	1Q 03
Z-0227	SFPI	Generator/Pagoda Area Demolition	1Q 03
Z-0172	PA00	Primary Aux. Building Demolition	1Q 03
0227	SEAL FI	Forebay/Seal Pit Demolition	1Q 03
MYM11	PA00	PAB Structure Demolition Complete	1Q 03
Z-0172S	PA00	Primary Aux Bldg Subgrade Demolition	1Q 03

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
0725	EFPR	Emergency Feed Pump Room Demolition	1Q 03
0227S	SEAL FI	Forebay/Seal Pit Subgrade Demolition	1Q 03
0725S	EFPR	Emergency Feed Pump Room Subgrade Demolition	1Q 03
0235	BWST	BWST Berm Demolition	2Q 03
0515	BWST	BWST Tank & Commodity Ripout	2Q 03
0235S	BWST	BWST Subgrade Demolition	2Q 03
0233	PWST	PWST Tank Demolition	2Q 03
0237	BWST	Contaminated Soil Removal (BWST)	2Q 03
0233S	PWST	PWST Subgrade Demolition	2Q 03
PP-17	BWST	BWST/Berm Demolition Packaging/Processing C&D	2Q 03
PP-16	PWST	PWST Demolition Packaging/Processing C&D	2Q 03
PWSTBF	PWST	PWST Subgrade Demolition Backfill	2Q 03
0222	HR00	Demolition High Rad Bunker	2Q 03
Z-0347X	MSW4	MSW4 - Warehouses #4 (Annex) Demolition	2Q 03
0228	DIFF	Diffuser Piping Demolition	2Q 03
Z-0347XS	MSW4	MSW4 - Warehouse #4 Subgrade Demolition	2Q 03
Z-0347X1	MSW2	WHSE - Warehouses #2/3 Demolition	2Q 03
0236L	115KV	Drop 115 KV Power Lines	2Q 03
Z-0347X2	MSW2	WHSE - Warehouses #2/3 Subgrade Foundation Demo	3Q 03
Z-0367	MSIC	Information Center Demolition	3Q 03

Table 3-2
Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)

Activity Number		Activity Description	Completion Date
Z-0482	BLPT	Bailey Farm House/BarnDemolition	3Q 03
Z-0482S	BLPT	Bailey Farm House/Barn Subgrade Demo	3Q 03
Z-0367S	MSIC	Information Center Subgrade Demolition	3Q 03
Z-0157S	RMCC	Reactor MCC Room Subgrade Foundation Demolition	3Q 03
0224	FPH1	Fire Pump House Commodities	3Q 03
0717	FR00	Fire Pump House Demolition (Above Grade)	3Q 03
0717S	FR00	Fire Pump House Demolition (Below Grade) (Water from Fire Pond will be sampled and processed as necessary for release)	3Q 03
Z-0069S	STFB	Staff Building & Tunnel Demo (Subgrade)	4Q 03
Z-0257	YDTP	Test Pit Demolition	4Q 03
Z-0387	YDLT	LIFT - Lift Station Demolition	4Q 03
0257S	SF00	RCA/Fuel Building Subgrade Demolition	4Q 03
0239	LSAB	LSA Building Demolition	4Q 03
Z-0267	YDFC	SECF - Security Fence Demolition	4Q 03
0722	YDOU	Remove Outside Utilities (3 ft. below) Demo.	4Q 03
0239S	LSAB	LSA Bldg. Subgrade Foundation Demolition	4Q 03
Z-0417	YDSH	Temp. Power Shack Demolition (Whitehouse)	4Q 03
Z-0417C	YDSH	Temp. Power Shack Foundation Demolition	4Q 03
Z-0307	YDLP	ULP - Utilities Light Poles Demolition	4Q 03
Z-0437	SEAL	SEAL - Seal Pit Outfall Demolition (Above Grade)	4Q 03

Table 3-2
Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)

Activity Number		Activity Description	Completion Date
Z-0437S	SEAL	SEAL - Seal Pit Outfall Demolition (Below Grade)	4Q 03
Z-0297	YARD	FPHS - Fire Protection Hose/STA/Hydrants Demo	4Q 03
0257	SF00	SFP / RCA Building Demolition	1Q 04
		Forebay Sediment Assessment/Remediation	1Q 04
MYM13	SFP2	Spent Fuel Pool Demolition Complete	2Q 04
Z-0148	CB00	Containment Building Demolition	2Q 04
Z-0317	YDPW	PW-Potable Water Connection Demolition	2Q 04
0724	YDSL	Sanitary Lines Demolition	2Q 04
Z-0148S	CB00	Containment Building Subgrade Demo	2Q 04
ROAD	YDRP	Site Roads & Parking Lots Demolition	2Q 04
Z-0407RT	YDRR	Railroad Tracks Demolition	2Q 04
Z-0237	MSMO	MOD - Modular Offices (2) Demolition	2Q 04
	0591	Final Doc. Submittal to NRC	2Q 04
	0287	Decommissioning Complete / Non ISFSI Land Released	3Q 04
		ISFSI Dismantlement, Decommissioning and Remediation (After removal of all spent fuel)	2026**
		Dismantlement of structures, support buildings, fences, lighting and utilities poles	2026**
		Site Remediation, planting of grass, trees, etc	2026**
		Final Facility Site Survey	2026**
		Release of the Facility Site for unrestricted use	2026**

Table 3-2 Area of Activity & Decommissioning Activities Schedule (Arranged Chronologically)			
Activity Number		**Activity Description**	**Completion Date**
		Termination of Maine Yankee Atomic Power Company's Part 50 License	2026**

** Projected date for DOE to have taken possession and removed stored materials.

3.2.3 Material Removal Sequence

Removal sequence may be dictated by access and material handling restrictions or by personnel exposure considerations. In most cases, a top-down approach will be used; materials and structures at the highest elevations are removed first to allow access to components in lower levels. In other cases, different approaches may prove more efficient.

Generally, the first items removed are those that are not, or are only slightly, contaminated to preclude contamination by other equipment. However, personnel exposure considerations may not always allow this option. Where non-contaminated equipment cannot be removed first, covers or other protection methods to ensure effective contamination control shall be used. Similarly, non-contaminated piping should be removed from pipe chases and horizontal pipeways before cutting pipes. If this is not possible, other precautions, such as covers, are used to minimize the spread of contamination.

Where rapid cutting techniques are available, pipes and equipment can be sectioned into pieces that are manageable using light rigging or by manual lifting. Where slow cutting techniques are used the largest manageable pieces will typically be freed and moved to a more convenient location for further reduction.

In the initial stages of decommissioning, most material removed from the containment building will pass through the modified equipment hatch and/or the additional 8' x 8' and 10' x 10' openings cut through the side of the containment that facilitates movement of materials when the larger opening is in use.

The plant is equipped with multiple cranes, hoists, and lifting and transport systems. These systems can be used to lift and transport components and equipment to support plant decommissioning activities. Forklifts, mobile cranes, front-end loaders, and other lifting and transport devices can also be used for plant decommissioning activities. The major installed plant cranes, hoists, and lifting and transport devices that are available to support decommissioning include:

a. Containment Building Polar Crane (360 ton, dual 185 ton hooks, 15 ton auxiliary hook)

b. Fuel Building Overhead Crane;

c. Equipment Room Monorails; and

d. Fuel Building Yard Crane (125 ton main hook /20 ton auxiliary hook)

e. Turbine Hall Overhead Crane

f. Plant Equipment Monorail Systems

The cranes continue to be maintained in accordance with applicable standards and regulations. The containment building equipment hatch modification allows the multi-wheeled transport direct access into containment.

The containment building polar crane is capable of reaching most locations inside the containment building and can handle large, heavy loads. The fuel building yard crane has access into the fuel building via a roll up door for movement of heavy components. This crane is being upgraded to be single failure proof for use in transferring spent fuel out of the spent fuel pool in spent fuel storage casks. The fuel building crane is used to some extent for movement of components in the spent fuel pool.

Installed cranes, hoists, and monorails may be used in conjunction with temporary or mobile lifting and transport devices to support decommissioning. The installed plant cranes, hoists, and other lifting devices can be decontaminated and dismantled when they no longer are required to support decommissioning activities.

3.2.4 Final State-of-the-Site Description

The purpose of this section is to present a conceptual description of the site following license termination and unrestricted release and to identify the extent of the types of media that must be considered in dose assessment, survey design and environmental assessment. Figure 3-30 shows the anticipated final state of the site. At license termination, when the site will be released for unrestricted use, the site will be a backfilled and graded land area with possibly some above grade structures remaining depending on the industrial reuse of the site. Generally speaking, all of the above grade structures will be demolished to three feet below grade and the resulting concrete demolition debris will be disposed of offsite at either a low-level waste facility or an appropriate disposal facility except for the 345 & 115 kV switchyards and possibly other administrative buildings. The remaining basement foundations will be filled with a soil fill material following any required remediation and FSS activities.

The former Low Level Waste Storage Building [now the ISFSI Security Operations Building-(SOB)] will remain in place until the fuel is transferred to the USDOE. The 115 kV switchyard and the 345 kV switchyard, will remain intact. The road that travels past the ISFSI will remain in place, terminating near the 115 kV switchyard. The original plant access road will remain but terminate between the ISFSI and the former location of the Information Building. The existing railroad that serves the ISFSI with its two spurs will remain in place, with one spur terminating near the 115 kV switchyard and the other terminating at the edge of the old road bed (formerly between the Restricted Area and the Service Building). The Old Ferry Road (a public road) and public boat ramp will remain in place.

Some below grade structures and systems will remain as described below:

Turbine and Service Building

These buildings will be demolished to three feet below grade. Concrete duct banks, building footings and foundations below this elevation will remain in place. A Final Status Survey will be performed on the remaining building footprint before it is backfilled. Piping below the foundation from the following systems will be removed: Primary Component Cooling, Secondary Component Cooling, sanitary sewer, oil lines, and floor drains. Service water intake lines may remain in place. The service water discharge line may be removed if necessary following final status survey. The service water discharge line was used as part of the radiological effluent discharge flow path from the test tanks. The circulating water discharge pipe encasement top may be left in place or may be broken and the lines backfilled following final status survey.

Containment, Primary Auxiliary Building, Fuel Building and Containment Spray Building

These buildings will be demolished to three feet below grade. Basement foundations below this level will remain in place and be backfilled with soil fill following remediation, as required, and final status survey. Some or all of the intervening walls and floors in the basements may be removed. The steel liner in the basement of the containment will remain in place. Many of the basement concrete and steel liner surfaces are covered with paint known to contain trace amounts of lead and/or PCB's. This paint will be removed prior to final status survey. The fuel transfer tube and bellows will be removed. The spent fuel pool liner will be removed due to known contamination levels. Some limited amounts of embedded pipe which penetrate basement walls will remain in place. These embedded pipes are easily accessible from either side for final status survey. Sub-mat "popcorn" concrete and its embedded drain lines around the sub-base of the containment will remain in place. These lines lead to the containment foundation sump pump which has been regularly sampled for contaminants. The containment drain sump will be demolished to three feet below grade. The foundation drain discharge line to the storm drain system may be removed

Above Grade Structures in the Radiological Controlled Area: High Radiation Bunker, Main Steam and Valve House, Emergency Feedwater Pump Room, LSA Building, Equipment Hatch/HV-7 & 9 Rooms, Ventilation Equipment Area, Reactor Motor Control Center Room.

These structures will be demolished down to three feet below grade. Building footings and foundations may remain.

Circulating Water Pump House

The Circulating Water Pumphouse (CWPH) will be demolished (demolish concrete 3' below grade with grade varying from El. 17' at the west wall to El. -3' at the east, backfill and cover with rock rip-rap). The intake structure which is below water level will remain in communication with the river. Outlet CW piping will be removed along with portions of the SW piping.

Sewage Treatment Plant

The Sewage Treatment Plant inlet pipes (coming from the TB/SB) will be removed, with outlet piping inspected, decontaminated (if required) and left buried, prior to building demolition to 3 feet below grade.

Foundations associated with Tanks, Guard Towers, Meteorological Towers, Yard Crane Footings, Vehicle Barriers, Transformers and Above Grade Structures including: Warehouse 2/3, 4 and 5, Administrative Building (Front Office), Gatehouse, Staff Building, Collection Center, Information Center

The meteorological tower will be dismantled and removed. The concrete footings and attached guide lines will be removed to three feet below grade and the area backfilled. Concrete foundations for tanks, guard towers, yard crane footings, vehicle barriers, transformers and buildings will be demolished to three feet below grade. Footings and foundations below three feet below grade may remain in place. Maine Yankee may determine that the Warehouses, Staff Building and Information Center can remain standing, after radiological release for unrestricted use.

Buried Piping

Buried fire protection and raw water piping will be left "in place." The CW and portions of the SW pipes between the CWPH and the Turbine and Service buildings will be removed. Piping between the DWST/RWST and the CSPA locations will be removed. All buried piping in the alleyway (area formerly between the service and containment buildings) will be removed. "Hot Side"storm drains will be decontaminated (if required) and left in place. Cold side storm drains will be left in place with the following caveats: All catch basins and manholes will be cleaned out, demolished to three feet below grade, and backfilled.

Forebay, Seal Pit and Diffuser Piping (see also Sections 8.2 and 8.6.4 regarding the NRPA process)

Maine Yankee evaluated the final disposition of the Forebay, seal pit, and diffuser piping. As part of a NRPA process, Maine Yankee analyzed remedial options and coordinating, as required, with the Maine Department of Environmental Protection and the U.S. Army Corps of Engineers. (Other responsible agencies coordinate through these two principal agencies). The key options evaluated included: (1) leave in place as exists; (2) secure and leave in place; (3) partial removal and; (4) complete removal. The types of impact that will be considered in the analyses include environmental impacts (water quality, marine wetlands, freshwater wetlands and land use), ecological impacts including flora, fauna and marine resources, and impacts on natural resources and navigation. The evaluation addressed the following options:

Diffuser Pipe, Foxbird Island – onshore below grade. Options include capping and leaving in place or removal, backfill and restoration to existing grade/conditions.

Diffuser Pipe, Mudflats – below the sediment/water interface. Options include capping and leaving in place or removal, backfill and restoration of the tidal flats.

Diffuser Pipe, Offshore above the sediment/water interface. The first option involves removal, possibly adding rip rap to the thrusters to form an artificial reef and augmented habitat. The second option involves leaving in place and filling with sand. This option may also include adding rip rap to form an artificial reef.

The concrete saddle supports and thrust blocks for the diffuser piping were left in place.

Forebay. All options involved demolishing the seal pit and removal of concrete down to three feet below grade, capping piping and trenches, and removing contaminated sediments, if required, consistent with the assumptions and dose assessment presented in Section 6.6.9. The first option also included removal of the west bank of the Forebay to re-establish tidal flow. The second option involved leaving the west bank of the Forebay in place, cutting down berms to above the high water elevation, and using this material as fill for the forebay. The third option involved leaving the berms in their present configuration.

Fire Pump House and Fire Pond

This pond existed solely for the purpose of holding water supplied from Montsweag Brook and pumping it to the fire water protection header surrounding the plant and adjacent buildings. It has no direct discharge path to the bodies of water surrounding Bailey Point. Demolition of the man-made, concrete lined Fire Water Pond, will return the site to conditions similar to pre-plant construction. After draining, the concrete liner will be removed and the earthen impoundment leveled. Surface water will once again flow across this area The Fire Pump House will be demolished to three feet below grade and backfilled with soil- like fill.

Bailey Farm House, Barn, Well water structure and systems

The Bailey farm house, barn and well water structure will be demolished to three feet below grade. Building footings and foundations may remain.

Restricted Area (RA)

The previously described Restricted Area (RA) will be radiologically released for unrestricted use. However, to assure compliance with non-radioactive environmental monitoring issues, it may be fenced, and the land deeded with restrictive covenants against excavating basements, drilling wells for drinking or irrigation water, or residential construction.

Independent Spent Fuel Storage Installation (ISFSI)

After the DOE transports all the stored spent fuel and GTCC wastes from the ISFSI, it will be decontaminated, if necessary, and demolished down to three feet below grade. A Final Status Survey will be performed for remaining lands and/or structures.

3.3 Methods of Decontamination and Dismantlement

3.3.1 Decontamination of Systems and Components

Systems and components removed and released from the secondary side of the plant for commercial disposal are surveyed in accordance with plant procedures based upon a no detectable radioactivity standard. Generally, systems and components removed from the primary (radiologically controlled) side of the plant are packaged and either transported to an offsite processing facility, LLRW disposal facility, or an appropriate disposal facility. Application of coatings and hand wiping may be used to stabilize or remove loose surface contamination. Potentially or slightly contaminated components (i.e., lighting ballast, mercury switches, etc) will be decontaminated onsite for release in accordance with plant radiological monitoring procedures for release.

Tanks and vessels are evaluated and, if required, flushed or cleaned to reduce contamination levels and remove sludge prior to sectioning and/or removal. The following considerations are incorporated into tank and vessel sludge removal activities:

a. Precautions are taken to ensure that in the unlikely event liquid inadvertently is discharged from the tank it will be captured (i.e., plugged lines, attached catch container, or temporary berm installation) for processing by a liquid waste processing system;

b. Sludge removed from the tank is stabilized prior to shipment in conjunction with the Maine Yankee Process Control Program (PCP); and

c. Wastewater will be processed and analyzed before being discharged.

3.3.2 Dismantlement of Systems and Components

Dismantlement methods can be divided into two basic types: non-destructive means such as disassembly, and destructive means such as cutting. Disassembly generally means removing fasteners and components in an orderly non-destructive manner (the reverse of the original assembly). Cutting methods include but are not limited to water jet, flame cutting, abrasive cutting, and cold cutting.

Water jet uses a very high-pressure stream of water to cut components (usually submerged underwater). Flame cutting includes the use of oxyacetylene and other gas torches, carbon arc torches, air or oxy arc torches, plasma arc torches, cutting electrodes, or combinations of these. Most of the torches can either be handheld or operated remotely with the appropriate devices. Abrasive cutting includes the use of grinders, abrasive saw blades, most wire saws, water lasers, grit blast, and other techniques that wear away metal. Cold cutting includes the use of band saw, blade saw, drilling, machining, shear, and bolt/pipe/tubing cutter devices.

Selection of the preferred method depends on the specific situation. Other dismantlement technologies may be considered and used if appropriate. Dismantling of systems includes the removal of valves and piping for disposal. Most valves can be removed with piping. Larger valves and valves with actuators may be removed separately for handling purposes.

Commodities are considered to be piping, HVAC, conduit, cable, cable tray, platform steel, pipe supports - basically any piece of equipment or material located within a building/structure that does not form an integral part of that structure. The removal of commodities will be based upon three general categorizations:

a. Areas within the Restricted Area (RA) of the plant

Systems and components will be removed from each area of the building/structure/yard, packaged, and either transported to an offsite processing facility, a LLRW disposal facility (Class A, B, or C), or an appropriate disposal facility.

b. Areas within the non-RA of the plant that may have some internal system contamination.

Systems and components identified during the site characterization and subsequently verified and bounded by Maine Yankee, will be remediated, and the balance of the building/structure surveyed and released for demolition. Remediation (leaving as is, removal, capping, or grouting) will depend on the level of radioactive contamination found (if any).

c. Areas outside of the Restricted Area (RA)

These areas have never been exposed to radiological contamination. Commodities will be demolished and removed with the building/structure upon completion of appropriate survey.

3.3.3 Decontamination of Structures

Structure decontamination methods typically include wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. Selection of the preferred method is based on the specific situation. Other decontamination technologies will be considered and used if appropriate.

If structural surfaces are washed to remove contamination, controls are implemented in accordance with approved plant procedures to ensure that wastewater is collected for processing by liquid waste processing systems. Airborne contamination control and waste processing systems are used as necessary to control and monitor releases.

Concrete that is activated will be removed down to the activated concrete DCGL and sent to a low level radioactive waste disposal facility. Removal of contaminated (non-activated) concrete will be performed using methods that control the removal depth to minimize the waste volume produced. Appropriate engineering controls for control of dust and debris will be used to minimize the spread of contamination and reliance on respiratory protection measures.

The following structural decontamination methods are described:

a. In-situ Concrete Decontamination by Bulk Removal

Diamond wire saw cutting may be used for the removal of volumetric concrete above the unrestricted use criteria, (or DCGL).

The removal of concrete consisting of the upper 1 or 2 feet of a thick slab such as a building foundation mat will require volumetric removals beyond the limits of scabblers or shot blasting. Whether due to activation or to leakage of liquids into concrete, the material may be removed using a mini-hoe ram or demolition robot. These have the flexibility to access congested areas and can be controlled to limit the volume of waste produced.

b. In-situ Surface Decontamination of Concrete

The expected depth of the contamination will establish the process used for the surface decontamination of concrete. Scabblers and shot blasting equipment fitted with vacuum collection systems may be used for surfaces with deeper contamination. Elsewhere, sponge blasting using one or more different media and wipe downs with solvents may be used. Cross-contamination and recontamination will be minimized using the vacuum collection systems.

c. Decontamination of Plant Concrete Structures That Are to Be Demolished (located higher than three feet below grade)

Contaminated concrete structures above three feet below grade may not be completely decontaminated. They will be packaged and shipped off site for disposal at a LLRW disposal facility or appropriate disposal facility.

d. Concrete Surfaces Located at Elevations Lower than Three Feet below Grade

Concrete surfaces below three feet below grade will be decontaminated if required to established criteria.

e. In-situ Surface Decontamination of Metals/Preparation of Metal - Surfaces for Segmentation

Most metallic wastes will not be decontaminated on site. Sponge blasting using various media ranging from non-aggressive for surface cleaning to heavy abrasive media or other methods for paint or oxide removal will be used and/or wipe downs with solvents. The contamination on exterior and/or interior metallic surfaces may be fixed prior to dismantling the structure or component.

Internal building steel within the RA will be dismantled, packaged, and shipped for processing, unless it can be easily determined that the steel can be released.

Steel located within non-RA buildings, i.e., not considered to have been exposed to radiological contamination, will be surveyed and released for demolition, i.e., Turbine Building, Circulating Water Pump Structure, etc. The external structural steel of plant buildings has been assessed during walkdowns and, depending upon the area, will either be surveyed and released for demolition or dismantled for packaging and shipment to a waste processor.

f. Embedded and Buried Piping Survey and Decontamination

There are two categories of pipe: buried pipe and embedded pipe. Buried pipe is pipe run underground, buried in a trench and surrounded by soil, whereas embedded pipe is encased in concrete. Treatment of buried pipe will depend on results from the surveys associated with the RCRA closure process as to whether it can be left in place, must be filled with inert material to be left in place, or must be removed. If buried pipe is to remain, it may be surveyed using the "pipe crawlers" to compare residual activity to the DCGLs or if the buried pipe is not expected to contain any residual activity, survey will only be conducted at accessible portions of the pipe, intakes or outfalls. The majority of embedded pipe ($\approx$1000 feet) is expected to be removed when concrete is demolished to three feet below grade. Embedded pipe remaining will also be surveyed using "pipe crawlers" or other appropriate method to compare residual activity to DCGLs.

The radiological pipe crawler allows in-situ survey, characterization, and decontamination of underground (buried) and embedded piping. By using this technology, safety risks, demolition costs, and secondary waste are reduced. (This technology has been used by other licensees to unconditionally release over 18,000 linear feet of piping with verification by the NRC). Based on survey results a decision considering the best engineering practice, will determine whether the remaining buried or embedded piping will be left as is, capped, or grouted.

Areas with activity above DCGL values will be remediated.

3.3.4 Building Demolition and Site Restoration

Table 3-3 describes the structures and facilities within the scope of the decommissioning along with the condition of release and final configuration.

Table 3-3
Structures and Facilities Within the Scope of Work for Demolition

Building or area description	Condition of release	Final configuration of structure
Containment building	1	Demo. 3 ft below grade; backfill
Steam and valve house	1	Demo. 3 ft below grade; backfill
Spray building	1	Demo. 3 ft below grade; backfill
Containment equipment hatch outer	1	Demo.
Containment personal hatch outer	1	Demo.
LSA building	1	Demo. 3 ft below grade; backfill
Reactor MCC room	1	Demo. 3 ft below grade; backfill
Emergency feed pump room	1	Demo. 3 ft below grade; backfill
Primary auxiliary building	1	Demo. 3 ft below grade; backfill
Fuel building	1	Demo. 3 ft below grade; backfill
RA building	1	Demo. 3 ft below grade; backfill
Service building - hot side	1	Demo. 3 ft below grade; backfill
Service building test tanks (TK-14A/B)	1	Complete demo.; remove pads; backfill
Demin. water storage tank (TK-21)	1	Complete demo.; remove pads; backfill
Primary water storage tank (TK-16)	1	Complete demo.; remove pads; backfill
Boron water storage tank (TK-13 A/B)	1	Complete demo.; remove pads; backfill
Refueling water storage tank (TK-4 & TK-54) and greenhouse	1	Complete demo.; remove pads; backfill

Table 3-3
Structures and Facilities Within the Scope of Work for Demolition

Building or area description	Condition of release	Final configuration of structure
Offgas stack	1	Complete demolition
Aux boiler stack	1	Complete demolition
High radiation bunker	1	Demo. 3 ft below grade; backfill
Turbine building	2	Demo. 3 ft below grade; backfill
WART/I&C building	2	Complete demolition; backfill
LLWB - Low level waste building	2	Complete demolition; backfill
FI - Foxbird island	TBD	To Be Determined (TBD) based on alternatives analysis
Security/Gatehouse building	3	Demo. 3 ft below grade; backfill
Administration building	3	Demo. 3 ft below grade; backfill
Fuel oil bunker	3	Demo. 3 ft below grade; backfill
Gas house	3	Complete demolition; backfill
Circ water pump house	3	Demo.West wall down to 3 ft below grade (17' elevation) and East wall to -3 ft elevation; backfill and add rip rap
Temporary power shack - West of fuel building	3	Complete demolition; backfill
Modular office buildings (2)	3	Complete demolition
8 Sided storage building	3	Complete demolition
Test pit	3	Complete demolition; backfill
West - area West of the containment Building including (3) guard towers	3	Complete demolition; backfill
Condensate storage tank (concrete pad only)	3	Demo. 3 ft below grade; backfill
Transformer including elect tower and concrete structures/pad	see below	see below

Table 3-3 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
1. X-IA	3	Remove transformer for disposition; demo. pads; backfill
2. X-IB	3	Remove transformer for disposition; demo. pads; backfill
3. X-24	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
4. X-26	3	Remove transformer for disposition; demo. 3 ft below grade ; backfill
5. X-28	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
6. X-IS	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
7. X- 14	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
8. X-16	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
9. X- 16A	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
10. X- 16B	3	Remove transformer for disposition; demo. 3 ft below grade; backfill
Temp generator/transformer/pagoda (cold & dark equipment)	2	Remove transformer for disposition; demo. pagoda
Outside protected area		
STPI - Sewage treatment plant	3	Demo. 3 ft below grade; backfill
Lift station	3	Demo. 3 ft below grade; backfill
Information center bldg (including concrete pad and blocks 4 ft x 4 ft x 4 ft)	3	Demo. 3 ft below grade; backfill

Table 3-3
Structures and Facilities Within the Scope of Work for Demolition

Building or area description	Condition of release	Final configuration of structure
Collection site	3	Demo. 3 ft below grade; backfill
Staff building/staff tunnel	3	Demo. 3 ft below grade; backfill
Annex - WHSE 4	3	Complete demolition; backfill
WHSE 2 & 3 - warehouse	3	Demo. 3 ft below grade; backfill
WHSE 5 - warehouse	3	Complete demolition; backfill
FPH - fire pump house/retention pond	3	Demo.; level; backfill
Seal pit (outfall)	TBD	To Be Determined (TBD) based on Alternatives Analysis
Barge slip area	3	grade area adjacent to slip
Pre-fab building (generator) and yard	3	Complete demolition; backfill
ELECT towers (inside industrial area)	3	Demo. towers and pads; backfill
Railroad cars (4) - includes track spur from fuel bldg to the property line	3	Demo. rail cars; remove rails to location designated; grade
Well water house	3	Demo. to 3 ft below grade; backfill
MET tower and equip. inside fenced area	3	Demo. 3 ft below grade; backfill
Electrical fence (around radio tower area)	3	Complete demolition; backfill
Concrete blocks - 4ft x 4ft x4ft Vehicle barriers	3	Remove initially and store for use; then demo.
Remove outside utilities - 3 ft below grade		
1. Potable water	3	Demo. 3 ft below grade; backfill

Table 3-3 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
2. Utility light poles	3	Demo. 3 ft below grade; backfill
3. Fire hydrant hose stations	3	Demo. 3 ft below grade; backfill
4. Sanitary lines	3	Demo. 3 ft below grade; backfill

Notes:
a. Number 1 denotes that the bldg/area will undergo decontamination and the commodities will be removed prior to building demolition.
b. Number 2 denotes that the bldg/area will have commodities removed and decontamination will be done as required. Remaining commodities will remain as is for building demolition.
c. Number 3 denotes that the bldg/area is clean and the commodities will remain as is for building demolition.

Property, structures and facilities will be demolished to a level three feet below present grade, with few exceptions. As a result of this approach, the following sequence of dismantlement and demolition will occur for buildings with Condition of Release 1 identified in Table 3-3:

a. Strip, package, ship commodities from the buildings (piping, steel, components, etc.) Commodities, including building steel determined to be clean may be released to the demolition contractor.
b. Perform decontamination of the building concrete surfaces (at elevations below 3 feet below grade) to meet established criteria levels. Package the debris from decontamination and ship for LLW processing and/or disposal.
c. Perform a final survey (sequence of "c" and "d" optional as described in section 3.1.3)
d. Release for demolition.
e. Demolish the building structure to 3 feet below grade. Separate the clean[2] rebar from the concrete.
f. Prepare the demolished concrete for shipment offsite.
g. Release rebar using established radiological release procedures and ship rebar to metal recycling contractor.

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[2] Clean rebar has no detectable, plant-derived radioactivity associated with it. Rebar will be surveyed in accordance with free release criteria and disposed of as scrap. If activated rebar is discovered it will be disposed of as radwaste.

The structures specified as Condition of Release 2 in Table 3-3, are those that are on the cold side of the plant and have been maintained as radiologically "clean," with the exception of some systems and equipment that may have internal contamination. Within these areas, the process for demolition will follow this process:

a. Remediate, package, and ship systems, components and, commodities identified within the site characterization report and assessed and bounded by Maine Yankee. Structural steel of plant buildings will either be surveyed and released for recycling or dismantled for packaging and shipment as LLW material.
b. Decontaminate, if required, to achieve the established radiological release criteria.
c. Perform radiation surveys to allow material release to the demolition contractor.
d. Release for demolition to the contractor.
e. Demolish structures and foundations to depth specified.
f. Subsurface piping to be handled as indicated above.
g. Perform final grade.

The buildings, structures, and facilities identified as Condition of Release 3 in Table 3-3 are those that do not have a history of contamination and are therefore classified as "presumed clean." In certain cases there are minor exceptions to this generalization, based upon the information in the site characterization report, such as a small area within the information center and the staff building, that appear to have been remediated. Also, the site characterization report identifies higher activity levels within the basement of the environmental lab (Bailey House), that may be attributed to background from the granite. However, Maine Yankee will evaluate and release these individual areas in accordance with plant radiological release procedures to allow for demolition. Procedural controls identify the monitoring requirements for construction debris release (Refer to Section 3.1.3).

Therefore, buildings, structures, and facilities identified as Condition of Release 3 in Table 3-3 will be processed as follows:

a. Remove ancillary equipment required for asset recovery (furniture, etc.) - (It is assumed that Maine Yankee will remove equipment designated for asset recovery, prior to the scheduled remediation/ demolition of the structure).
b. Perform survey in accordance with established procedures and criteria.
c. Release for demolition to the contractor.

3.4 Evaluation of Dismantlement Activities

3.4.1 Systems Review

The license basis status of Maine Yankee systems, structures, and components (SSC), is summarized in Table 3-4. As indicated by this table, the majority of radiologically contaminated systems and components not required to support the storage of spent fuel have been abandoned and accepted for decommissioning in accordance with the plant System Evaluation Reclassification Team (SERT) file 98-136. SERT designates them as ready to be "accepted for decommissioning." These SSCs will be deactivated, dismantled, and disposed of in accordance with the schedule described above. NRC and other regulatory agencies will be informed of significant schedule changes during weekly scheduled communications.

3.4.2 System Deactivation

SSCs, which are not Important to the Defueled Condition (ITDC), nor required to support the Spent Fuel Pool (SFP) are placed in an "abandoned" status per a defined program [System Evaluation Reclassification Team (SERT) file 98-136], which designates them as ready for "decommissioning". Those systems listed as "NO" in the "Required for SFP" Column in Table 3-4 have been eliminated from consideration in the license basis.

Systems or components will continue to be abandoned/deactivated prior to decontamination, if necessary and dismantlement. In general, deactivation is implemented by mechanical isolation of interfaces with operating plant systems, draining piping/components, and de-energizing electrical supplies. Combustible material (e.g., charcoal form filters, lube oil) is removed from abandoned/deactivated components where possible. Chemicals used in, or resulting from, decommissioning activities are controlled in accordance with the applicable chemical safety program. Plant drawings are revised to indicate abandoned/deactivated portions of systems. Plant procedures are modified to reflect the changes when applicable.

Abandonment/deactivation of plant systems is controlled by approved plant procedures. The deactivation plans are established to implement the desired system valve lineup changes and electrical isolations. The design change process is used to remove components, lift electrical leads, install electrical jumpers, cut and cap piping systems, or install blank flanges as appropriate.

Plant procedures provide controls over the operation of deactivated system boundary valves. As additional systems are deactivated, existing isolation boundaries are re-evaluated and changed, as necessary, to reflect the new plant condition. Mechanical boundaries of abandoned SSCs (including boundary valves) are specifically identified in accordance with Maine Yankee's procedures

Temporary liquid and solid waste processing systems may be used during decommissioning for processing plant waste. These systems may include filters and/or demineralizers and may be used at one or more locations in the waste-processing path. Localized temporary ventilation equipment and HEPA filtration may be used to supplement building ventilation and minimize the spread of radioactive particulate contamination.

Table 3-4
Status of Major MY Systems, Structures, and Components

System/Component/Structure	Required for Defueled Condition	Status
Reactor coolant system	NO	Removal mostly complete
Reactor vessel internals	NO	Segmentation and Removal complete
Reactor vessel	NO	Removal complete
Secondary component cooling water system	NO	Removal ongoing
Potable water system Wiscasset water system	YES	Preparations for partial removal are ongoing, portion remaining in service to support SFP System and site needs
Spent fuel pool and fuel handling equipment	YES	Preparations for partial removal are ongoing, portion remaining in service to support SFP System
Spent fuel pool cooling system	YES	Will remain in service as long as materials are stored in the SFP
Spent fuel pool cooling and demineralizer system	YES	Will remain in service as long as materials are stored in the SFP
Plant effluent system	YES	Rerouted to support Forebay remediation and dismantlement
Containment ventilation systems	YES	Preparations for removal are ongoing as portions of system are no longer required to support decommissioning or SFP

Table 3-4
Status of Major MY Systems, Structures, and Components

System/Component/Structure	Required for Defueled Condition	Status
Fuel building ventilation systems	YES	Will remain in service as long as materials are stored in the SFP
Instrument and service air systems	NO	Preparations for removal of Components not required for support of spent fuel pool are ongoing.
Solid radioactive waste system	YES	Will remain in service as long as materials are stored in the SFP, and support decommissioning activities after removal of SFP contents.
Liquid radioactive waste system	YES	Will remain in service as long as materials are stored in the containment Building, Auxiliary Building, or SFP, and support decommissioning activities after removal of SFP contents.
Radiation monitoring system	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing
Electrical systems	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing in Accordance with decommissioning
Fire protection systems	YES	Preparations for partial removal of components not required for support of spent fuel pool are ongoing (Back Up make-up water supply to Spent Fuel Pool)
Containment building	NO	Preparations for removal are ongoing. Impact on fuel transfer canal minimized through redesign

Table 3-4
Status of Major MY Systems, Structures, and Components

System/Component/Structure	Required for Defueled Condition	Status
Auxiliary building	YES	Preparations for partial removal of components not required for support of spent fuel pool are ongoing in accordance with decommissioning schedule
Turbine/generator building (has a wall that supports the SFPI_systems)	NO	Removal complete except for wall that supports the SFPI_systems
Low level waste storage building	NO	Converted over to the Security and Ops bldg for ISFSI

3.4.3 Nuclear Safety and Regulatory Considerations

The following general considerations, as applicable, will continue to be incorporated into packages during the decommissioning period.. During the decommissioning period, dismantlement activities will be reviewed to ensure that they do not impact safe storage of fuel and GTCC wastes in the ISFSI licensed under a general Part 72 license. Work packages are implemented in accordance with administrative controls. When applicable, decommissioning work is reviewed against the requirements of 10 CFR 50.59, 50.82(a)(6) and/or 72.48 to ensure work that is being performed without prior NRC approval does not need a license amendment.

Following complete transfer of the spent fuel from the spent fuel pool to the ISFSI, Maine Yankee will dismantle and demolish the spent fuel pool. Maine Yankee submitted a license amendment, pursuant to 10 CFR 50.90, to add an applicability statement to certain technical specifications that describe requirements associated with the spent fuel pool. On February 6, 2002, this license amendment was approved.

Dismantlement activities will be conducted to ensure the safe storage of spent fuel and to protect the public health and safety as well as the common defense and security. Maine Yankee's Quality Assurance Program (QAP) defines the mechanical SSCs in Table 3-5 as safety related.

Table 3-5
Safety Related Mechanical Components

Component	Safety Class
Spent fuel pool cooling loop suction piping (from the pool wall up to and including the siphon protection)	3
Fuel transfer tube	3
Blind flange on containment side of fuel transfer tube	3
Valve FP-21 (transfer tube isolation valve)	3

Other items are identified by the Quality Assurance Program (QAP) as requiring a degree of quality and are designated as QA Related (QAR). Following removal of fuel from the spent fuel pool, Maine Yankee will revise the Quality Assurance Program, pursuant to 10 CFR 50.54(a)(3) to delete the safety related and QAR classifications related to spent fuel storage in the spent fuel pool. Equivalent classifications for the ISFSI and the NAC UMS cask are specified in Appendix B to the Maine Yankee Quality Assurance Program.

3.5 Radiological Impacts of Decontamination and Dismantlement Activities

3.5.1 Waste Characterization

The MY Decommissioning Project Waste Management Plan includes waste disposal strategies, and addresses issues such as: estimates of the quantity of radioactive material to be released, control mechanisms, and radioactive waste characterization. Radioactive waste has been characterized by sending representative samples for 10 CFR Part 61 analysis. Table 3-6 lists the nuclides for which the samples were analyzed. Table 3-7 presents typical sample Part 61 analysis results.

3.5.2 Radioactive Waste Projections

Any data provided herein are estimated values and may or may not represent actual final volumes. The subject values shown in Table 3-7 provide relative fractions of nuclides historically present in Maine Yankee's waste streams. This and other information sources were used to identify those nuclides which were requested for Part 61 analyses. Alternate means and methods may be utilized when appropriate to reduce these volumes. The projected activities and volumes of radioactive material generated are summarized in Table 3-8 and 3-9.

Table 3-6
Nuclides Checked for by 10 CFR61 Analysis

Nuclide	Principal Emission★	Nuclide	Principal Emission★	Nuclide	Principal Emission★
Ag-110m	gamma	Zr-93	beta	*Nb-94 (in activated metal - C-14, Ni-59, Ni-63)	gamma
Am-241	alpha	Sn-126	beta	+Kr-85	gamma
C-14	beta	Iso-U	alpha	#Cr-51	gamma
Cm-242	alpha	K-40	gamma	#Fe-59	gamma
Cm-243/244	alpha	Zn-65	gamma	#Nb-95	gamma
Co-57	gamma	Eu-154	gamma	#Zr-95	gamma
–	--	Eu-155	gamma	#Mo-99	gamma
Co-60	gamma	Eu-152	gamma	#I-131	gamma
Cs-134	gamma	Tl-208	gamma	#Xe-133	gamma
Cs-137	gamma	Bi-212	gamma	#Ba-140	gamma
Fe-55	ec	Pb-212	gamma	#La-140	gamma
H-3	beta	Bi-214	gamma	#Ce-141	gamma
Mn-54	gamma	Pb-214	gamma	#Sn-113	gamma
Ni-59	ec	Ra-226	gamma	#Sb-124	gamma
Ni-63	beta	Ac-228	gamma	#Ru-103	gamma
Pu-238	alpha	Pa-234m	gamma	#Co-58	gamma
Pu-239/240	alpha	Th-234	gamma		
Pu-241	beta	U-235	gamma		
Sb-125	gamma	Be-7	gamma		
Sr-90	beta	Ce-144	gamma		
*Tc-99	beta	Sb-126	gamma		
* +I-129	beta	Sn-126	gamma		

★Analysis performed measuring this principal emission
*for waste classification
#for Rx pwr operations (short lived) + in spent fuel

Table 3-7
10 CFR61 Sample Analysis Results (Typical)
These values are shown to present relative fractions of nuclides historically present.

NUCLIDE	RESIN SAMPLE µCi/g 8/21/96	LIQUID FILTERS µCi/sample 9/4/96	SMEAR ACTIVITY µCi/sample 6/18/97	CAVITY DRAIN DOWN FILTER µCi/sample ACTIVITY 7/23/96	UPENDER SMEAR µCi/sample ACTIVITY 6/10/98	#3 SG BOWL µCi/sample SMEAR ACTIVITY 3/28/98
H-3	2.00E-01		1.86E-02	1.40E-01		15500
C-14	8.51E-02					4270
Mn-54	8.01E-01		1.63E-03	1.00E-02	1.00E-03	1260
Fe-55	9.81E+00	7.49E-01	2.46E-01	2.58E-01	4.33E-02	160000
Co-57	2.07E-02		2.37E-04	4.79E-04		
Co-58	7.70E-02		4.15E-02			674
Co-60	9.68E+00	1.64E+00	4.48E-01	3.61E-01	1.14E-01	147000
Ni-59	1.04E-01					
Ni-63	1.42E+01	1.40E+00	3.34E-01	8.97E-02	3.86E-02	18700
Zn-65						
Sr-90	2.38E-01			2.74E-02		370
Zr-93						
Nb-94						
Tc-99						6920
Ag-110m			1.37E-03			
Sb-125	2.72E-01	2.62E-03	5.81E-03	1.61E-03	2.95E-03	2110
Sn-126						
I-129						
Cs-134	2.00E+01		5.54E-03			
Cs-137	3.72E+01	3.35E-03	8.06E-02	1.03E-02		
Ce-144				2.45E-03		
Eu-152						
Eu-154						
Eu-155						
U-234						
U-235						
U-238						
Pu-238	6.67E-04	1.53E-04	1.45E-05	1.83E-04	1.20E-05	6.9
Pu-239/240	2.79E-04	1.91E-04	2.24E-05	6.02E-04	1.53E-05	5.3
Pu-241	2.05E-02	1.65E-02	1.98E-03	2.32E-02	9.60E-04	315
Am-241	3.56E-04	2.71E-04	3.00E-05	2.77E-04	1.55E-05	3.4
Cm-242	1.64E-04			3.21E-06	6.10E-06	
Cm-243/244	4.53E-04	2.34E-04	1.19E-05		8.50E-06	1.8

Table 3-8
Projected Activities and Volumes

Activity	Curies	Volume
Reactor vessel and internals	2,200,000 Ci	11,500 Cu.Ft.
Large NSSS components Steam Generators Pressurizer	1,600 Ci	27,000 Cu.Ft.
Activated Concrete	390 Ci	23,000 Cu. Ft.
Contaminated Debris (Structural Steel, etc)	0.10 Ci	163,000 Cu. Ft
Contaminated Concrete	1.75 Ci	900,000 Cu. Ft.
Radioactive Water	0.5 Ci	850,000 Gallons
Soil①	0.1 Ci	25,000 Cu. Ft

① This volume is an estimate, subsurface soil will be sampled and surveyed following commodity removal

The Total Estimated Radwaste Volumes Transported and Buried are described in Table 3-9 (reproduction of Table 1 - PMP 9.0 Rev A, Section 6.3). Although the total estimated radwaste volumes exceeds the 18,340 m^3 described in NUREG-0586 the associated impacts are bounded by those addressed in the FGEIS as discussed in detail in section 8.7.

Materials removed and/or generated during the demolition process will be disposed of based upon the origin of the material and the radiological survey findings prior to or after demolition.

Radiologically contaminated concrete materials generated from the RA (from demolition at elevations above 3 feet below grade), will be shipped off site for disposal at a LLW facility or appropriate disposal facility. Disposal of building reinforcing steel and structural steel, which has been properly released, will be performed by the demolition contractor to scrap and/or landfill areas.

Table 3-9
Total Low-level Waste Volume per Maine Yankee Decommissioning

Item	Vol ft³	Transportation Mode	Disposal/Process	Disposal Volume ft³
Reactor Pressure Vessel (RPV)	9,500	Barge	Direct Disposal	9,500
Non-GTCC RPV Hardware	1,500	Cask/Truck	Direct Disposal	1,500
RPV Head	300	Cask/Truck	Direct Disposal	300
Pressurizer	2,200	Barge	Processing VR 21.3/1*	100
Reactor Coolant Pumps & Motors	4,800	Train/Truck	Direct Disposal	4,800
Steam Generators	20,000	Barge	Processing VR 21.3/1*	940
Radioactive Contaminated Metal	150,000	Truck	Processing VR 21.3/1*	7,040
Dry Active Waste (DAW) Resin	13,000	Truck	Processing VR 21.3/1*	610
Liquid Waste Processing	400	Cask/Truck	Direct Disposal	400
Spent Fuel Pool Purification Resin	150	Cask/Truck	Direct Disposal	150
Pre-Existing Waste	200	Cask/Truck	Direct Disposal	200
Contaminated Soil	25,000	Train	Direct Disposal	25,000
Radioactive Concrete	900,000	Train	Direct Disposal	900,000
Used Oil, Radioactive	270	Truck	Processing/ Incineration	0
TOTAL VOLUME SHIPPED	**1,127,320 ft³ (31,924 m³)**			
[74% increase over NUREG value-18,340 m³]				
TOTAL DISPOSAL VOLUME @ a Volume Reduction of 176,770 **ft³**				**950,550 ft³ (26,920 m³)**
[47% increase over NUREG value-18,340 m³] **NOTE:** This would require an additional .3 acre more than the 2 acres described in the NUREG.				

* Past performance as of 01-16-01 indicates a 21.3 to 1 Volume Reduction

Table 3-10 below describes the approach to handling building materials for regulatory release.

Table 3-10
Approach to Handling of Building Materials for Regulatory Release

No.	Type of building material	Approach
1	Areas with low contamination potential	Free-release in accordance with procedures
2	Concrete with medium to high surface contamination potential (at elevations above - 3 feet below grade)	Ship offsite for disposal at Envirocare or Barnwell or an appropriate disposal facility
	Concrete with medium to high surface contamination potential (at elevations below - 3 feet below grade)	Remediate to acceptance criteria levels and leave in place, with removed material disposal at Envirocare or Barnwell
3	Contaminated metals removed	Ship to processor or for disposal at Envirocare or Barnwell
	Non-contaminated metals removed	Ship to processor for scrap or disposal
4	Built-up tar roofing, inner layer of siding (with actual or potential contamination)	Process at LLW treatment facility or directly dispose at Envirocare
	"Clean" tar roofing, siding	Ship to a processor or disposal
5	Outer layer of siding (Galbestos)	Surface release survey; send to asbestos landfill
6	Refueling cavity and spent fuel pool liners	Process at LLW treatment facility

3.5.3 Occupational Exposure

The estimated total nuclear worker exposure during decommissioning is estimated to be 946 person-rem which is below the 1215 person-rem found acceptable for decommissioning in the reference PWR NUREG-0586 Table 4.3-2.

Table 3-11 lists estimated exposure/area of activity Attachment A at the end of this document provides pictorial reference using the Acronym assigned to the decommissioning activity. Detailed planning precedes initiation of each specific activity, and includes engineering design, ALARA planning, and refinement of cost, schedule, and required resources estimates.

Table 3-11
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled

Area/Activity	Title	Exposure
DC.2 Period 2 (Decommissioning) DC.2.01 NSSS Removal DC.2.01.01 Reactor coolant piping DC.2.01.02 Pressurizer relief tank DC.2.01.03 Reactor coolant pumps and motors DC.2.01.04 Pressurizer DC.2.01.05 Steam Generators DC.2.01.06 CRDMs & service structure removal DC.2.01.07 Reactor vessel internals DC.2.01.08 Reactor vessel		93.951 REM
DC.2.03 System removal DC.2.03.01 Containment		
DC.2.03.01.01 **Cbl-1**	CTMT Loop #1	97.114 REM
DC.2.03.01.02 **Cbl-2**	CTMT Loop #2	65.745 REM
DC.2.03.01.03 **Cbl-3**	CTMT Loop #3	63.171 REM
DC.2.03.01.04 **Cbl-4**	SI Tank #2 & Regen Ht Exch E-67	11.592 REM
DC.2.03.01.05 **Cbl-5**	CTMT -2 Lvl Pressurizer Area	25.411 REM
DC.2.03.01.06 **Cbl-6**	CTMT -2 Lvl Sump Pump Area	22.608 REM
DC.2.03.01.07 **Cbl-7**	CTMT Iodine Filter Area	6.485 REM
DC.2.03.01.08 **Cbl-8**	CTMT -2' Outer Annulus	43.334 REM
DC.2.03.01.09 **CB2-1**	CTMT 20' Outer Annulus	19.313 REM
DC.2.03.01.10 **CB3-1**	Reactor Cavity Area	19.615 REM
DC.2.03.01.11 **CB3-2**	CTMT Cavity Upender Pit	26.683 REM
DC.2.03.01.12 **CB3-3**	CTMT 46' Penetration Room	6.078 REM
DC.2.03.01.13 **CB3-4**	CTMT Polar Crane (CR-1)	4.042 REM

Table 3-11
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled

Area/Activity	Title	Exposure
DC.2.03.01.14 **CCG**	CTMT Charging Floor	3.105 REM
DC.2.03.01.15 **CEHO**	CTMT Equip Hatch Outer (PE-3)	3.871 REM
DC.2.03.01.16 **CICI L DC.**	CTMT Incore Instrument Sump	6.533 REM
DC.2.03.01.17 **CPHO**	CTMT Personal Hatch Outer Area	.728 REM
DC.2.03.01.18 **CPLE**	CTMT Elevator & Room	.173 REM
DC.2.03.02 Primary Auxiliary		
Bldg	PAB 21' Level Valve Alley	.742 REM
DC.2.03.02.01 **P21A**	PAB 21' Boric Acid Pump Area	6.387 REM
DC.2.03.02.02 **P21B**	PAB 21' Charging Pump Cubicle	22.718 REM
DC.2.03.02.03 **P21C**	PAB 21' Level Degas Cubicle	9.160 REM
DC.2.03.02.04 **P21D**	PAB 21' Evap Cubicle	39.169 REM
DC.2.03.02.05 **P21E**	PAB 21' Heat Exchanger Room	16.495 REM
DC.2.03.02.06 **P21H**	PAB 21' General Area	1.418 REM
DC.2.03.02.07 **P21L**	PAB 21' Sample Sink Area	2.799 REM
DC.2.03.02.08 **P21S**	PAB 21' Level HPSI Room	.956 REM
DC.2.03.02.09 **P21V**	PAB Lower Lvl Aerated Drain Tank Area	22.184 REM
DC.2.03.02.10 **PLAD**	PAB Lower Lvl Boric Acid Mix Tank Area	13.790 REM
DC.2.03.02.11 **PLBA**	PAB Lower Lvl Aux Chrg Pump Cubicle	5.054 REM
DC.2.03.02.12 **PLCP**	PAB Lower Lvl Degas Cubicle	1.551 REM
DC.2.03.02.13 **PLDC**	PAB Lower Lvl Evap Cubicle	13.751 REM
DC.2.03.02.14 **PLEC**	PAB Lower Lvl Letdown Area	38.761 REM
DC.2.03.02.15 **PLLA**	PAB Lower Lvl Ctmt Penetration Area	28.907 REM
DC.2.03.02.16 **PLPA**	PAB Lower Lvl Primary Drain Tank Area	11.122 REM
DC.2.03.02.17 **PLPD**	PAB Lower Lvl Pipe Tunnel	30.815 REM
DC.2.03.02.18 **PLPT**	PAB Lower Lvl Primary Water Pump Area	.289 REM
DC.2.03.02.19 **PLPW**	PAB Upper Lvl FN-48 Area	.485 REM
DC.2.03.02.20 **PU48**	PAB Upper Lvl Decay Drum Cubicle	.512 REM
DC.2.03.02.21 **PUDD**	PAB Upper Lvl Evap Cubicle	5.921 REM
DC.2.03.02.22 **PUEC**	PAB Upper Lvl FN-1A/B Area	.506 REM
DC.2.03.02.23 **PUFN**	PAB Upper Lvl Heat & Ventilation	.383 REM
DC.2.03.02.24 **PUHV**	PAB Upper Lvl General	1.741 REM
DC.2.03.02.25 **PUL**	PAB Upper Lvl Radioactive Storage Area	.316 REM
DC.2.03.02.26 **PUSA**	PAB Upper Lvl VCT Cubicle	.529 REM
DC.2.03.02.27 **PUTC**	PAB Upper Lvl Waste Gas Cubicle	.279 REM
DC.2.03.02.28 **PUWG**		
DC.2.03.04 **Service/fuel**		

Table 3-11
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled

Area/Activity	Title	Exposure
building/SVH/SPRB	Demineralizer Water Storage Tank (TK-21)	.103 REM
DC.2.03.04.01 **DWST**	Emergency Feed Water Pump Room	.159 REM
DC.2.03.04.02 **EFPR**	Fuel Building Proper	
DC.2.03.04.03 **FBP**	LSA Storage Building	.628 REM
DC.2.03.04.04 **LSAB**	New Fuel Laydown Area / Fuel Vault	1.622 REM
DC.2.03.04.05 **NFLA**	RCA Drumming Room	
DC.2.03.04.06 **RCAD**	RCA Waste Solidification	8.772 REM
DC.2.03.04.07 **RCAW**	Reactor MCC Room	.046 REM
DC.2.03.04.08 **RMCC**	Service Building Decon Room	.314 REM
DC.2.03.04.09 **SBDR**	Service Building HP Checkpoint	.044 REM
DC.2.03.04.10 **SBHP**	Service Building Machine Shop	.293 REM
DC.2.03.04.11 **SBMS**	Service Building Proper	
DC.2.03.04.12 **SBP**	Service Building Seal Room	.111 REM
DC.2.03.04.13 **SBSR**	Service Building Test Tanks (TK-14 A&B)	
DC.2.03.04.14 **SBTT**	Service Building Steam & Valve House	
DC.2.03.04.15 **SBVH**	Spent Fuel Pool	32.159
DC.2.03.04.16 **SFP**	Spent Fuel Pool Heat Exchanger Room	REM
DC.2.03.04.17 **SFPH**	Spent Fuel Pool Ventilation Room	9.120 REM
DC.2.03.04.18 **SFPV**	Spray Building	.287 REM
DC.2.03.04.19 **SPRB**	Steam & Valve House	78.093 REM
DC.2.03.04.20 **SVH**		.054 REM
DC.2.03.05 Miscellaneous		
DC.2.03.05.01 **BWST**	Boron Waste Storage Tanks (TK-13 A&B)	.162 REM
DC.2.03.05.02 **CST**	Condensate Serge Tank (TK-122)	.003 REM
DC.2.03.05.03 **CWI**	Circulating Water Pump House	
DC.2.03.05.04 **FI**	Foxbird Island	
DC.2.03.05.05 **FOB**	Fuel Oil Pump House & Bunker	
DC.2.03.05.06 **FPH**	Fire Pump House	
DC.2.03.05.07 **GH L**	Gas House	
DC.2.03.05.08 **HRB**	High Radiation Bunker	.528 REM
DC.2.03.05.09 **PWST**	Primary Water Storage Tank (TK-16)	.068 REM
DC.2.03.05.10	RWST/SLAT Tanks	1.549 REM
RWST/SCAT		
DC.2.03.05.11 **STFB**	Staff Building & Tunnel	
DC.2.03.05.12 **STPI**	Sewage Treatment Plant	7.136 REM
DC.2.03.05.13 **West - RCA**	RCA Yard Area - West Side	

Table 3-11
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled

Area/Activity	Title	Exposure
Total Estimated exposure for the project		**937.543 REM**

3.5.4 Public Exposure

Continued application of Maine Yankee's Radiation Protection Program, Waste Management Plan, Radiological Effluents Controls Program and Radiological Environmental Monitoring Program assures public protection in accordance with 10 CFR 20. Details for remediation are provided in Section 4 of this LTP. LTP Section 8 contains an evaluation of estimated public exposure as a result of decommissioning activities including the transportation of radioactive waste.

3.5.5 Expected Radiological Conditions

Characterization of concrete within the Restricted Area (RA) of the site shows the following:

1. Painted concrete has surface contamination up to 1 million dpm/100 cm^2 (worst case) which is amenable to surface remediation techniques such as wiping, washing, power washing or abrasive surface removal.

2. Bare concrete has surface contamination, absorbed contamination and activation products within the concrete matrix. Surface contamination levels are similar to those for painted concrete. Absorbed activity has been found to penetrate to a depth of approximately 1 mm.

3. Concrete structures adjacent to the reactor vessel also showed activation products at levels of a few pCi/g except for the In Core Instrumentation (ICI) sump where levels were as high as 600-800 pCi/g to depths of several inches. These types of radioactivity are amenable to remediation by surface removal techniques except for the deeply deposited activation products.

4. Surface abrasive or surface removal remediation techniques may generate airborne radioactivity. Airborne activity will be controlled within the requirements of 10 CFR Part 20 and measured using standard processes and procedures existing within the radiation protection program. These processes and procedures have proven successful for controlling

decontamination and demolition activities in the past while protecting the health and safety of the workers and the public.

Maine Yankee has segmented the reactor vessel internals and loaded resulting GTCC waste into NAC UMS casks for storage at the ISFSI. This segmentation process used an abrasive water jet. Special precautions were taken to capture the residue (SWARF) resulting from this segmentation.

3.5.6 Contamination Control

Due to the large scope of the D&D and the need for some FSS activities to be performed in parallel with dismantlement activities, a systematic approach to controlling areas is established. Upon commencement of the FSS for survey areas within the Restricted Area (RA) where there is a potential for re-contamination, implementation of one or more of the following control measures will be required:

a. Personnel training
b. Installation of barriers to control access to surveyed areas
c. Installation of barriers to prevent the migration of contamination from adjacent overhead areas
d. Installation of postings requiring personnel to perform contamination monitoring prior to surveyed area access
e. Locking entrances to surveyed areas of the facility
f. Installation of tamper-evident labels
g. Upon completion of FSS, the area is placed under periodic routine survey by Radiation Protection to ensure no re-contamination occurs. If re-contamination is identified, an investigation will be initiated that would result in corrective actions up to and including re-performance of the FSS on that area.

During the D&D activities, measures will be maintained and/or established to control and monitor radwaste effluents. This consideration should not preclude the removal of penetrations and attachments to the containment building, provided that openings are closed, or can be closed in a timely manner.

<u>Airborne Controls</u>

Airborne radioactive particulate emissions will continue to be filtered, and effluent discharges sampled/monitored and quantified. Consideration is given to the following items:

a. Operation of the appropriate portions of the containment ventilation and purge system, or an alternate system, during decontamination and dismantlement activities in the containment building;
b. Operation of the appropriate portion of the auxiliary building ventilation system, as required.
c. Operation of the appropriate portion of the fuel building ventilation system to support the fuel building.
NOTE: The auxiliary building roof physically supports the fuel building ventilation ducts.
d. Use of local HEPA filtration systems for activities expected to result in the generation of airborne radioactive particulate (e.g. grinding, chemical decontamination, or thermal cutting of components)

When applicable during demolition engineering controls such as misting will be applied to concrete surfaces. Where practical for ALARA purposes, temporary shielding is used during decommissioning activities. Some dismantlement activities may be performed under water for shielding purposes as well as contamination control.

Liquid/Particle Control

Work activities are planned to minimize the spread of contamination. Liquids are contained within existing or supplemental barriers and processed by a liquid waste processing system prior to release. To minimize the potential for spread of contamination, the following considerations will continue to be addressed when planning decommissioning work activities.

a. Covering of openings in contaminated components to confine internal contamination;

b. D&D of SSCs by decontamination in place, removal and decontamination, or removal and disposal;

c. Removal of supports in conjunction with equipment removal or decontamination of supports in conjunction with building decontamination;

d. Removal of systems and components from areas and buildings prior to structural decontamination (block shield wall, portions of other walls, ceiling, or floors may be removed to permit removal of systems and components.);

e. Removal or decontamination of embedded piping, conduit, ducts, plates, channels, anchors as required, sumps, and sleeves during area and building structural decontamination activities;

f. Use of local or centralized processing and cutting stations to facilitate packaging of components removed in large pieces; and

g. Removal of small or compact plant components and parts intact, where feasible. (This includes most valves, smaller pumps, some small tanks, and heat exchangers. These components could then be decontaminated in whole or part, and reduced to smaller dimensions in preparation for disposal or release.)

3.6 Coordination with Other Regulatory Agencies

The decommissioning and termination of Maine Yankee's Part 50 license involves, in addition to the NRC, coordination with a number of federal, state and local agencies as well as several advisory groups. This section outlines the broad responsibilities of those groups and also addresses specific environmental issues raised in the FGEIS in the context of the Maine Yankee site.

3.6.1 Regulatory Agencies

The following federal, state and local agencies have some level of involvement in Maine Yankee's decommissioning. Some have direct approval authority over site activities while others serve in an advisory capacity to other agencies. Their primary functions, programs, and regulatory authority are described below.

b. US Environmental Protection Agency (EPA) - EPA has been engaged in discussions with various stakeholders about the Maine Yankee decommissioning process. The EPA is supporting the Maine Yankee decommissioning project in several areas. The EPA is enabled by 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 (MDEP) 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, painted structural steel and painted concrete surfaces.

The EPA previously administered the National Pollutant Discharge Elimination System (NPDES) permit program as authorized by the Clean Water Act in Maine. Maine Yankee maintained an NPDES permit during operation to reflect discharge of certain process wastewater during decommissioning. Effective January 12, 2001, MDEP administers the NPDES program on EPA's behalf. MDEP has issued a new discharge license to Maine Yankee

c. US Department of Transportation (DOT) - The DOT regulates the packaging, labeling and shipment of waste materials offered for interstate commerce. Waste materials that are expected to be shipped from Maine Yankee during decommissioning that are regulated by the DOT include radiological wastes, mixed waste, and hazardous waste. DOT approved the transportation of the Pressurizer and Steam Generators as their own shipping containers and the shipping container for the Reactor Vessel.

d. US Coast Guard - The Coast Guard has authority to control vessel traffic in the navigable waterways of the US. Barge shipment of large components will be coordinated with the Coast Guard to ensure that all applicable requirements for securing loads and notifying the public are met.

e. US Department of Energy (DOE) - The DOE has a contractual obligation to take receipt and dispose of Maine Yankee's GTCC waste and spent nuclear fuel.

f. Maine Department of Environmental Protection (MDEP) - The MDEP is the lead state agency responsible to prevent, abate and control the pollution of the air, water and land and prevent diminution of the natural environment of the state. The MDEP has authority in a variety of statutes and accomplishes its charge through a number of regulations. The MDEP regulates solid and hazardous waste activities, development activities at Maine Yankee through the Site Location of Development Law, industrial discharges, air emissions, and activities affecting significant natural resources including coastal and freshwater wetlands. These aspects are discussed in more detail in Section 8.6.

g. Maine Department of Human Services - The Department of Human Services through the Division of Health Engineering (DHE) has responsibility for radiological programs within the state. DHE also sponsors the two State Nuclear Inspectors that monitor activities at Maine Yankee.

h. Maine Department of Inland Fisheries and Wildlife (IF&W) - IF&W does not directly regulate activities at Maine Yankee. IF&W does however provide technical support to the MDEP for permitting activities relating to development projects and projects that may affect significant natural resources. IF&W is also responsible for the Maine threatened and endangered species protection program.

i. Maine Department of Marine Resources (DMR) - DMR does not directly regulate activities at Maine Yankee. However DMR does provides technical support to MDEP on projects involving potential impacts to coastal wetlands.

j. Maine Department of Transportation (MDOT) - MDOT has permitting authority for new development projects generating over 100 passenger car equivalent trips in the peak hour. It is not anticipated that MDOT will have active involvement in decommissioning activities.

k. Maine Historic Preservation Commission - Maine Yankee has coordinated with this organization for the preservation of the two identified archaeologic sites on Maine Yankee property. The specific location of archaeological sites is not provided to ensure their integrity is protected.

l. Town of Wiscasset (Town) - The Town has permitting authority over new development projects such as the recently permitted Independent Spent Fuel Storage Installation (ISFSI) now under construction. The Town also has permitting authority over major earthwork projects. It is expected that final site grading will trigger Town review and approval requirements.

m. The Maine Turnpike Authority- has a long standing agreement that placarded shipments of LLW will only travel on the Turnpike during daylight hours.

n. The Maine State Police- are given a courtesy call before each LLW shipment leaves the site. This is not an official requirement.

3.6.2 Advisory and Community Entities

a. The State Nuclear Safety Advisor responsibilities include advising the Governor and legislature on nuclear power issues, specifically transport and storage of nuclear waste at Maine Yankee. The Advisor also consults with relevant federal agencies and coordinates the activities of state agencies with respect to decommissioning. Another duty is to keep abreast of related activities in other states and to advise the Governor and legislature on such activities. In addition to making these recommendations and updates to the Governor, the Advisor prepares an annual report.

b. The Governor's Technical Advisory Panel (TAP) is currently comprised of four professors with expertise in radiological sciences from the University of Maine, Colorado State University, University of Michigan, and the University of Massachusetts Lowell. This panel was assembled in 1999 to provide independent evaluation of technical decommissioning issues and to advise the Governor accordingly. Panel members are Dr. C. T. Hess, Dr. F. Ward Whicker, Dr. Glenn Knoll, and Dr. George E. Chabot.

c. The Maine Advisory Commission on Radioactive Waste and Decommissioning is charged with overseeing radioactive waste activities in the state, including the decommissioning process at Maine Yankee. The Commission meets on a quarterly basis. Its members include state legislators, members of the public, waste generators and state agency staff.

d. The Maine Yankee Community Advisory Panel (CAP) was established in 1997 to enhance opportunities for public involvement in the decommissioning process of Maine Yankee. The CAP represents the community. By thoroughly reviewing the decommissioning process, the CAP is in a position to advise Maine Yankee on key issues of concern to the regional community.

e. Friends of the Coast (FOC)- Friends of the Coast Opposing Nuclear Pollution is a local environmental organization founded in 1995. Friends of the Coast participates regularly in stakeholder discussions on the full range of decommissioning issues and has a seat on the Maine Yankee CAP.

3.6.3 Environmental and Regulatory Issues

Section 8.6 of the LTP provides a detailed discussion of how non-radiological environmental and regulatory issues associated with decommissioning are being addressed with the cognizant state and federal agencies having jurisdiction over those issues.

3.7 References

3.7.1 NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans"

3.7.2 NRC Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (January 1999)

3.7.3 Post Shutdown Decommissioning Activities Report

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

3.7.5 "Site History Report," Stone and Webster Environmental Technology and Services (November 1999), transmitted via James T. Kilbreth letter to Joan Jones, State of Maine, dated November 16, 1999

3.7.6 Kim Tripp, US Fish and Wildlife Services, letter to David Asherman, dated July 21, 1999, regarding federally listed species.

3.7.7 NRC letter to Maine Yankee, dated July 30, 2002, Issuance of Amendment No. 167, license amendment approving partial release of site lands

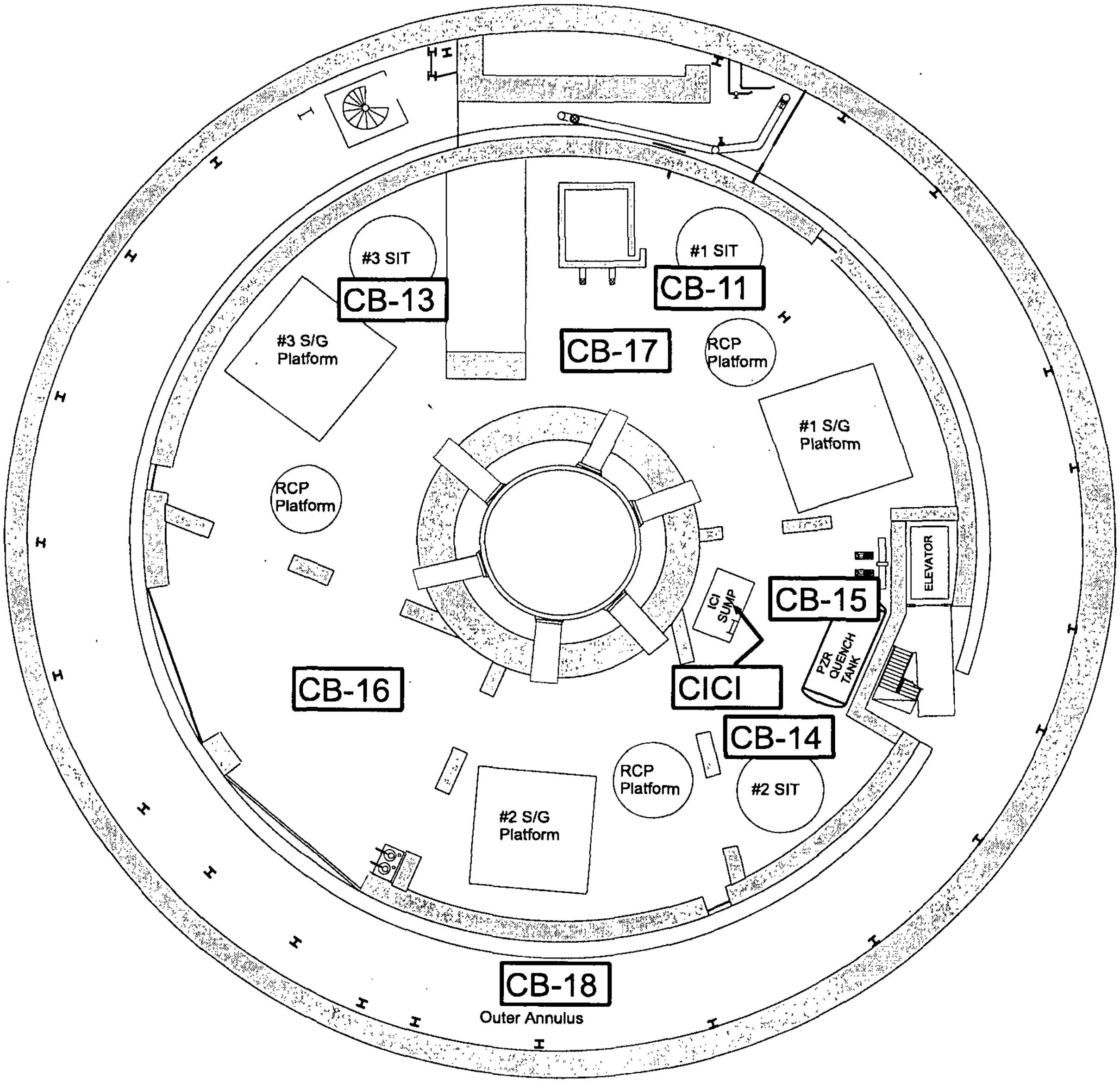
ATTACHMENT 3A

Drawings Associated with Specific Decommissioning Tasks

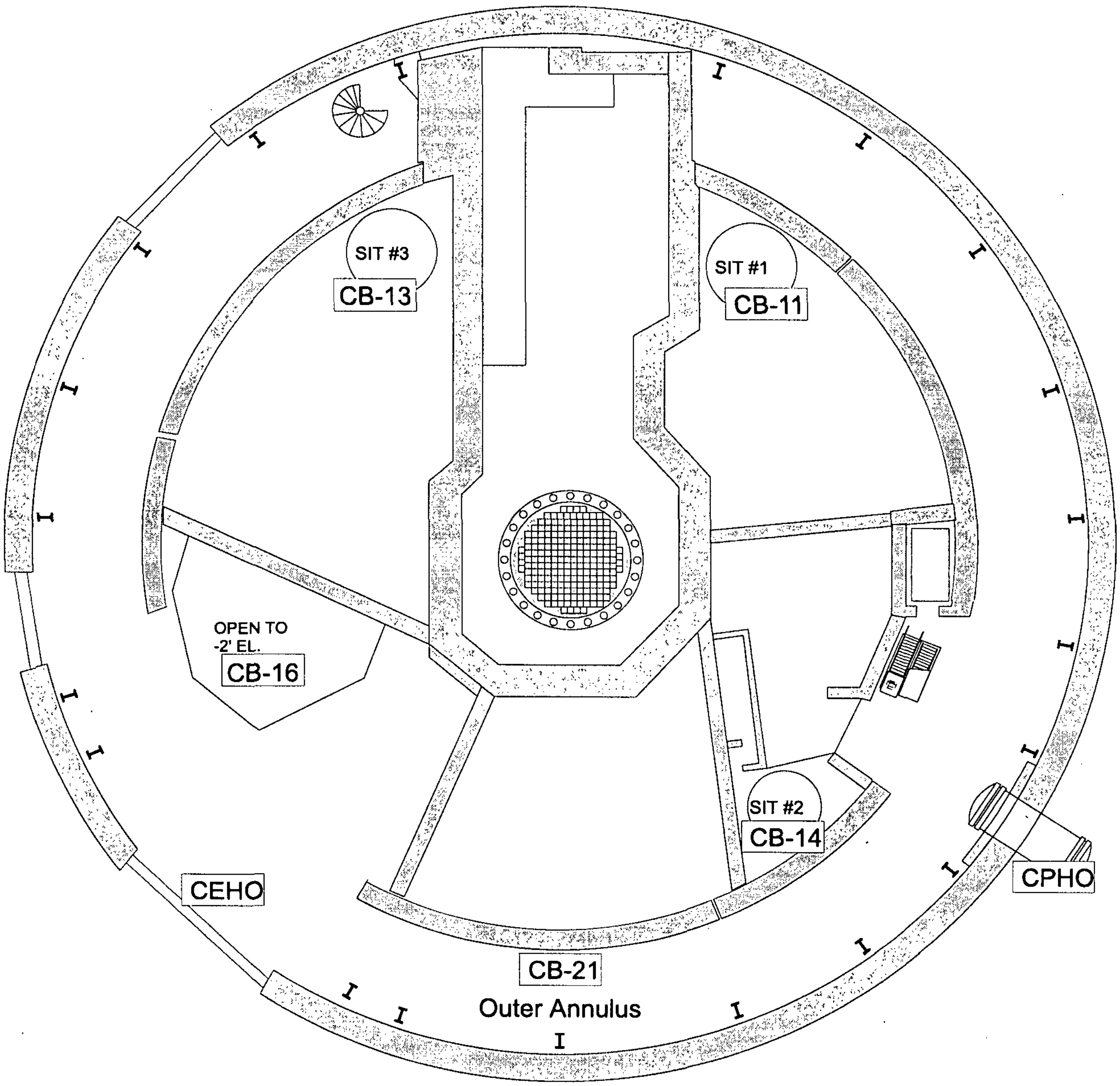
These drawings are provided "For Information Only" to support the reader's understanding and correlation of decommissioning tasks, anticipated radiation exposure, and physical locations involved in the subject tasks.

TABLE 3A-1 DECOMMISSIONING AREAS				
Figure Number	**Figure Title**	**Areas**		
Figure 3-1	Containment Building -2' Elevation General Area	CB-11 CB-13 CB-14	CB-15 CB-16 CB-17	CB-18 CICI
Figure 3-2	Containment Building 20' Elevation General Area	CB-11 CB-13 CB-14	CB-16 CB-21	CEHO CPHO
Figure 3-3	Containment Building 46' Elevation General Area	CB-32 CB-33	CB-34 CP-34	CPLE
Figure 3-4	Primary Auxiliary Building 11' Elevation General Area	PLAD PLDC PLEC	PLLA PLPA	PLPD PLPW
Figure 3-5	Primary Auxiliary Building 21' Elevation General Area	PLAD PLPD P21D	P21E P21H P21L	P21L
Figure 3-6	Primary Auxiliary Building 36' Elevation General Area	PU-48 PUDD PUEC PUFN	PUFN PUHV PULI PULI	PUSA PUSA PUTC PUWG
Figure 3-7	Containment Building Electrical Penetration Room	RMCC	RMCC	RMCC
Figure 3-8	Containment Building Mechanical Penetration Room Levels 4 & 5	SVH1		
Figure 3-9	Spray Building 4' Elevation	SBRP		
Figure 3-10	Spray Building 6' Elevation	SBRP		
Figure 3-11	Spray Building 11' Elevation	SBRP		
Figure 3-12	Spray Building 12' & 21" Elevation			
Figure 3-13	Spray Building 20' & 30' Elevations General Area	RMCC	SBRP	
Figure 3-14	Fuel Building 21' Elevation Decon Room	RCAD		
Figure 3-15	RCA Storage Building Waste Solidification	RCAW		
Figure 3-16	Fuel Building 21' Elevation Spent Fuel Pool Heat Exchanger Area	SFPH		
Figure 3-17	Fuel Building 21' Elevation Fuel Laydown Area	SFP03		
Figure 3-18	Fuel Building New Fuel Storage 31'-1 ½" Elevation	NFLA		
Figure 3-19	Fuel Building 44'-6" Elevation	SFP02	SFP04	SFPV

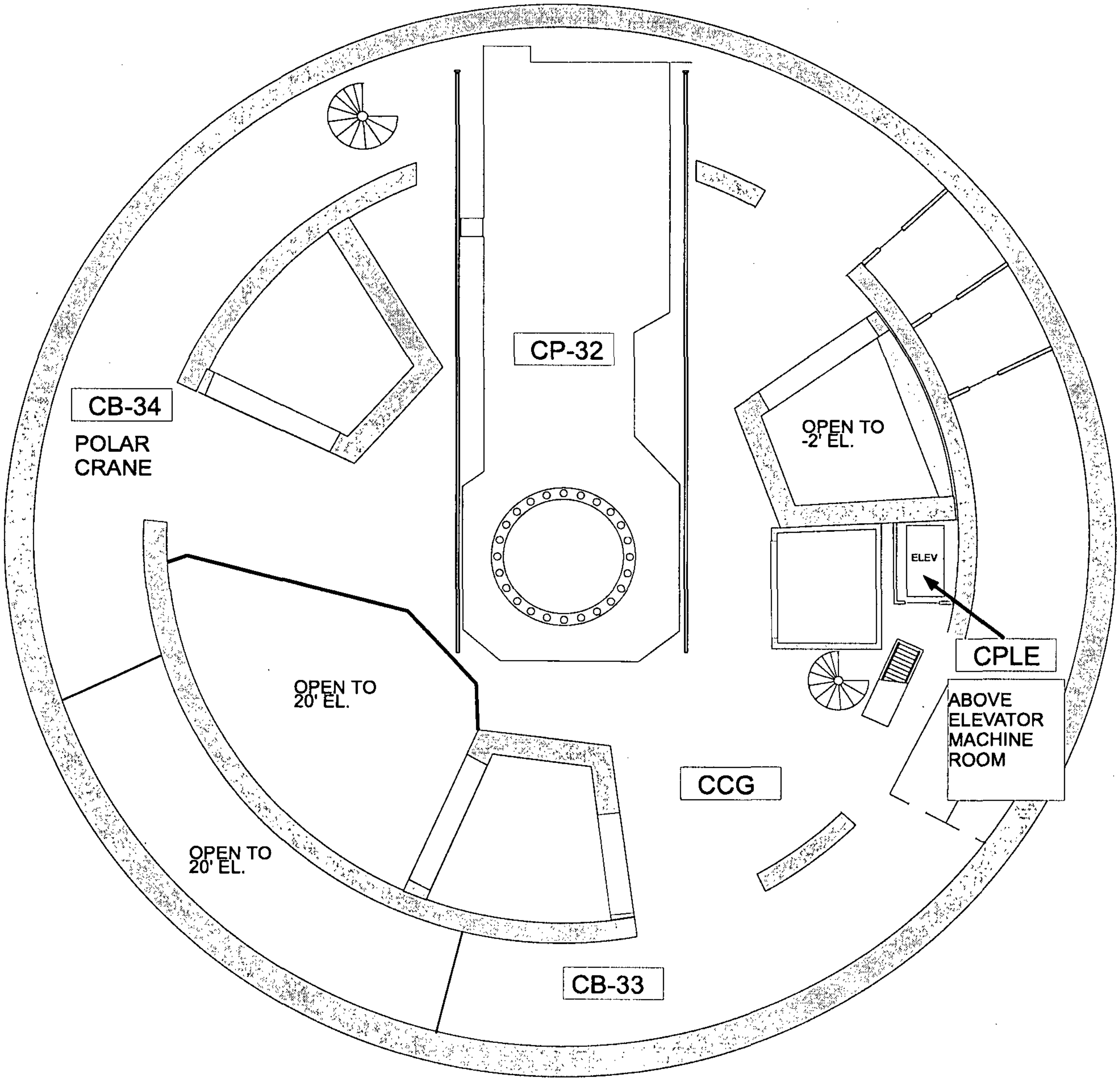
TABLE 3A-1 DECOMMISSIONING AREAS				
Figure Number	**Figure Title**	**Areas**		
Figure 3-20	Montsweag Bay			
Figure 3-21	Site Area Layout	CCWI FPHI HRB	PWST STPI X-16	X-19
Figure 3-22	Service Building 21' Elevation	SB011 SB02 SB03	SBDR SBHP SBMS	SBP SBP SBSR
Figure 3-23	Service Building 21' Elevation Control and Computer Room			
Figure 3-24	Cold Side Service Building 30'-10" Elevation 2nd Floor General Area	SBP		
Figure 3-25	Service Building 39' Elevation	SB05	SB07	SB08
Figure 3-26	Cable Vault Room 49' Elevation	SB09	SB10	
Figure 3-27	Turbine Building 21' Elevation	TBLD TBMS TBSO TCA TCBA	TCDA TCNE TCSE TFPA THDT	THEA TMC TP2C TSPP
Figure 3-28	Turbine Building 61' Elevation	MTGE TBR2 TD01	TDBV TDRW TDSN	TDSS TSRP TSRU
Figure 3-29	Turbine Building 39' Elevation	FWH TCTP TCTU TDR2 TMAE TMCH TMDV	TMDW TMEC TMFA TMFR TMFV TMGL	TMHD TMIA TMLT TMNC TMSE TMWC TMWT
Figure 3-30	Final Site Configuration			



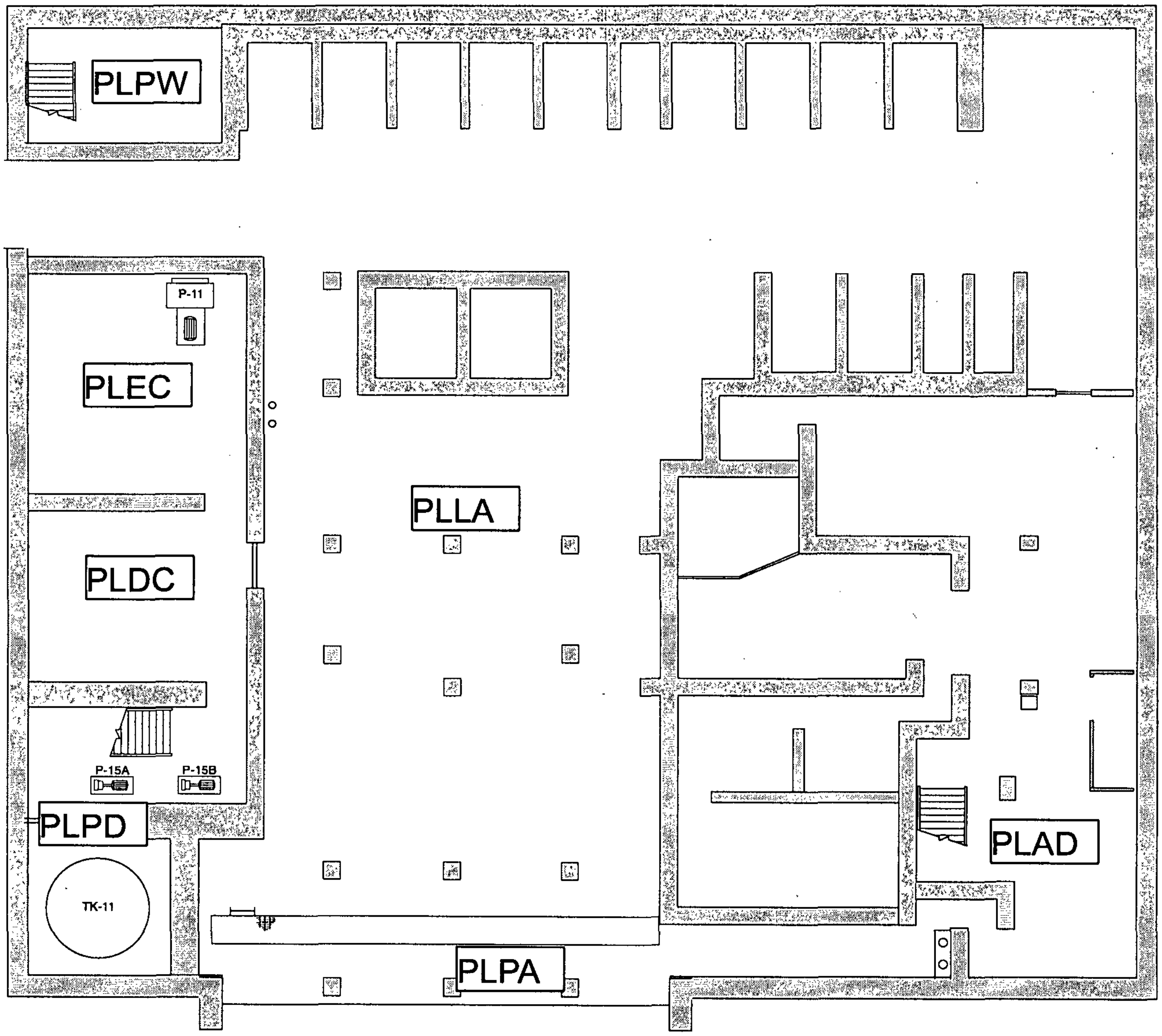
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Containment Building -2' Elevation General Area	Figure 3-1



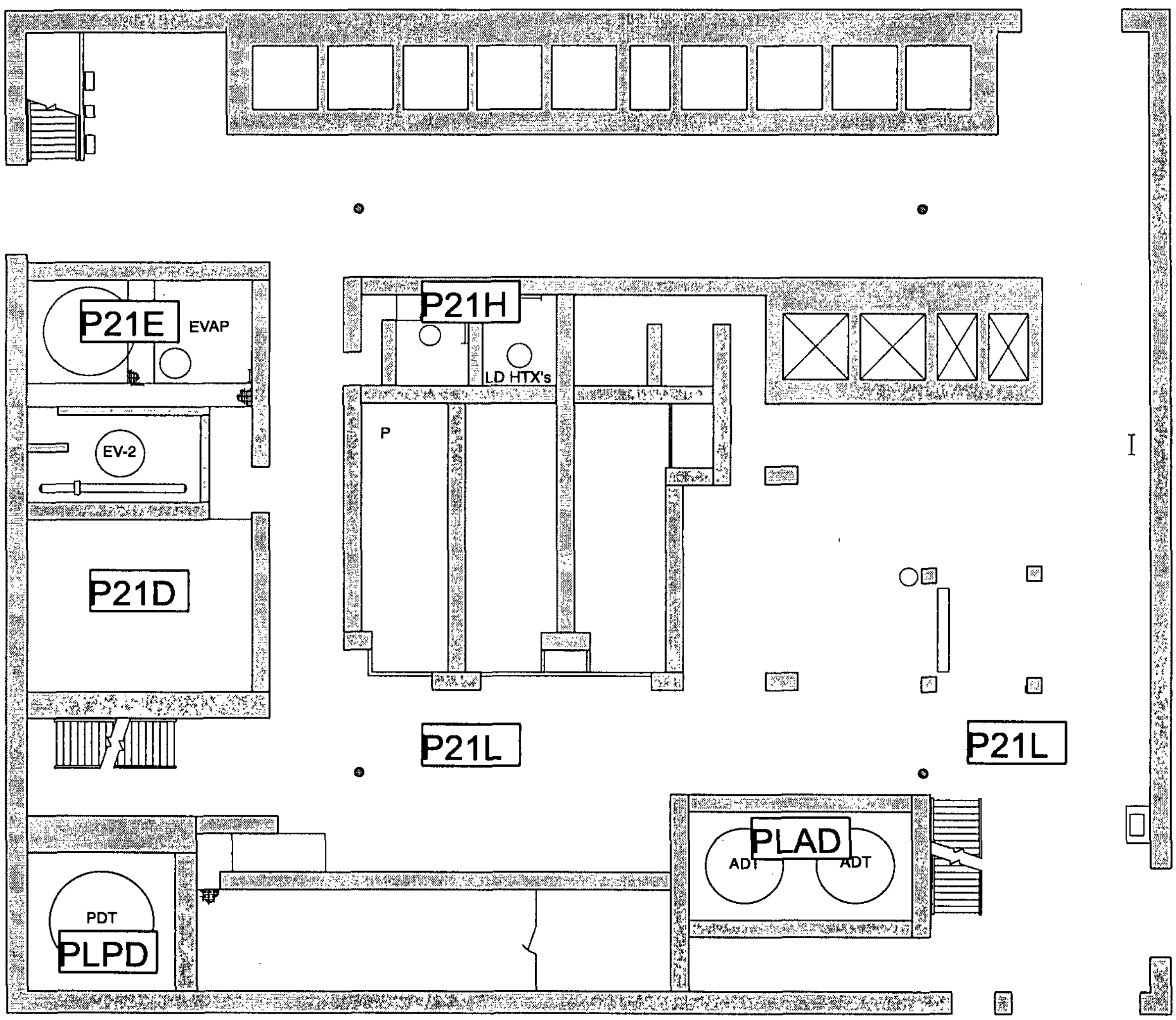
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Containment Building 20'Elevation General Area	Figure 3-2



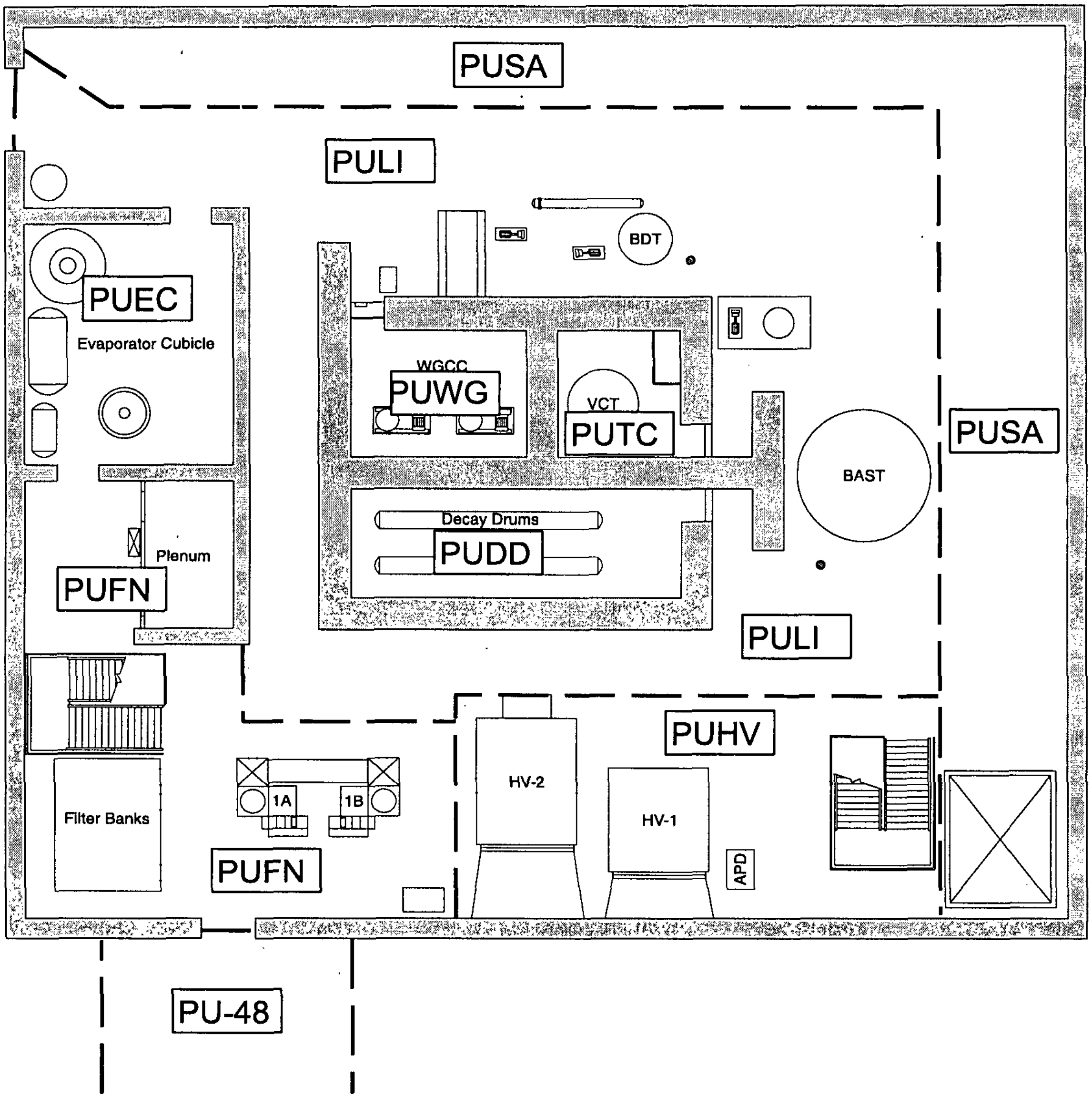
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Containment Building 46' Elevation General Area	Figure 3-3



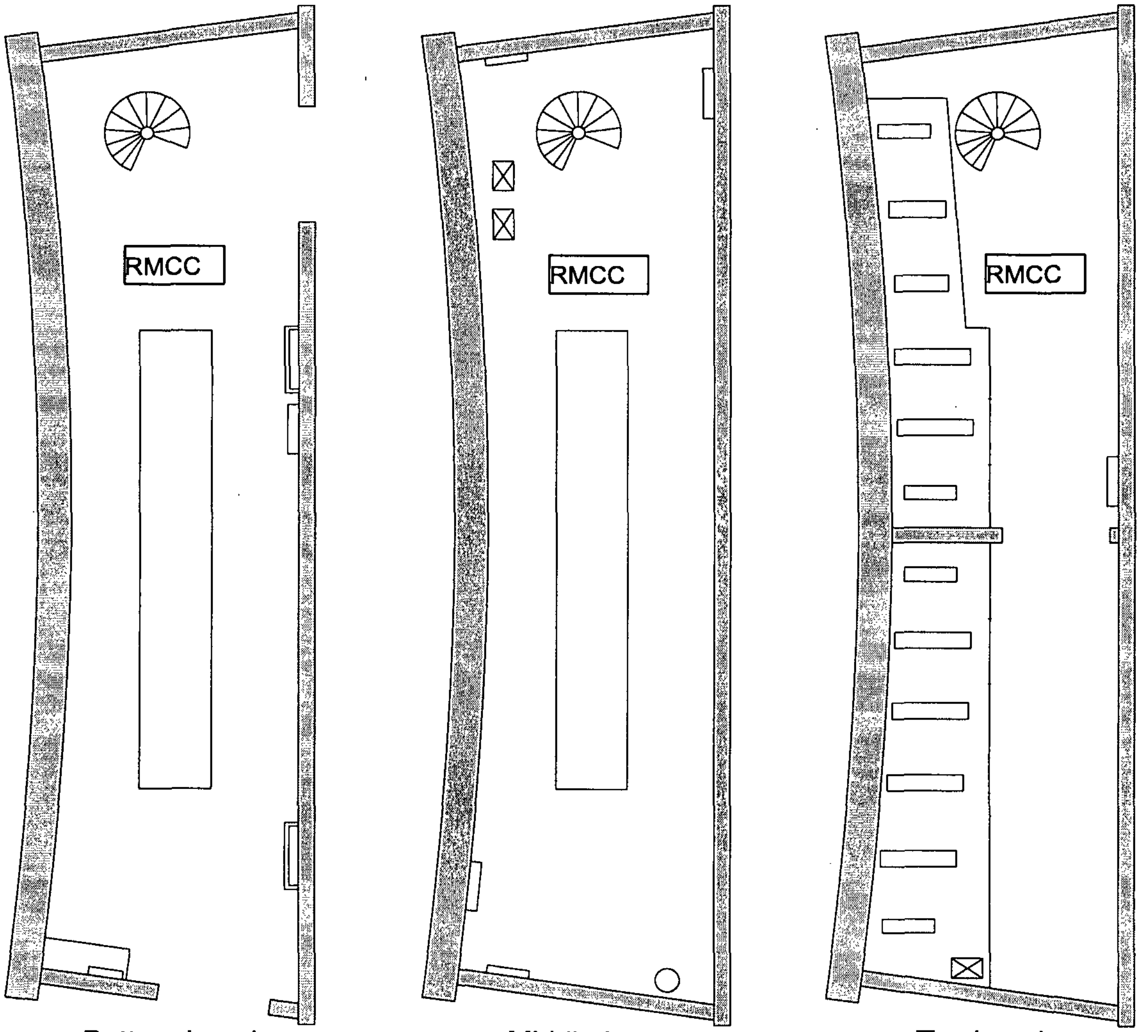
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Primary Auxiliary Building 11' Elevation General Area	Figure 3-4



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Primary Auxiliary Building 21' Elevation General Area	Figure 3-5



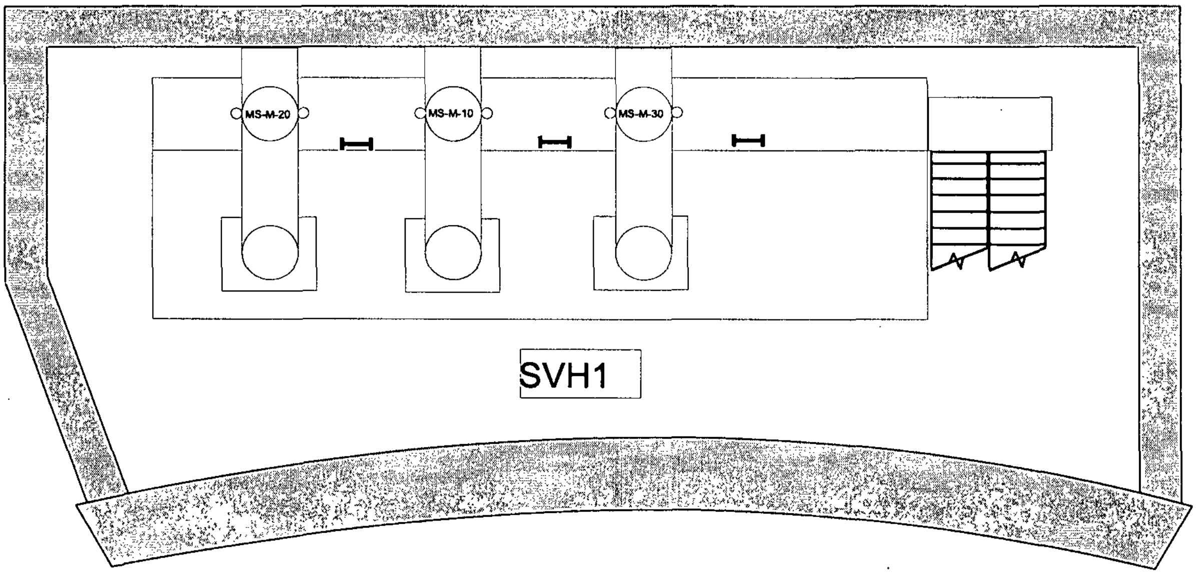
MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Primary Auxiliary Building 36' Elevation General Area	Figure 3-6



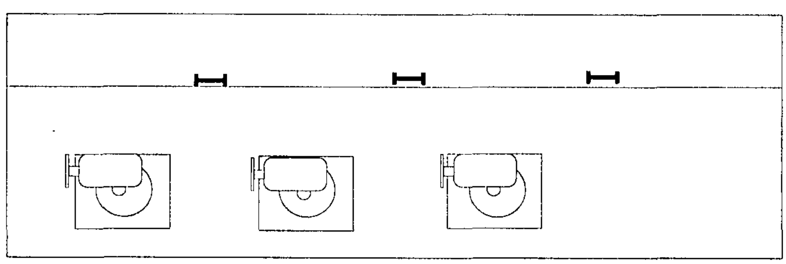
Bottom Level　Middle Level　Top Level

Containment Building
Electrical Penetration Room

Figure 3-7

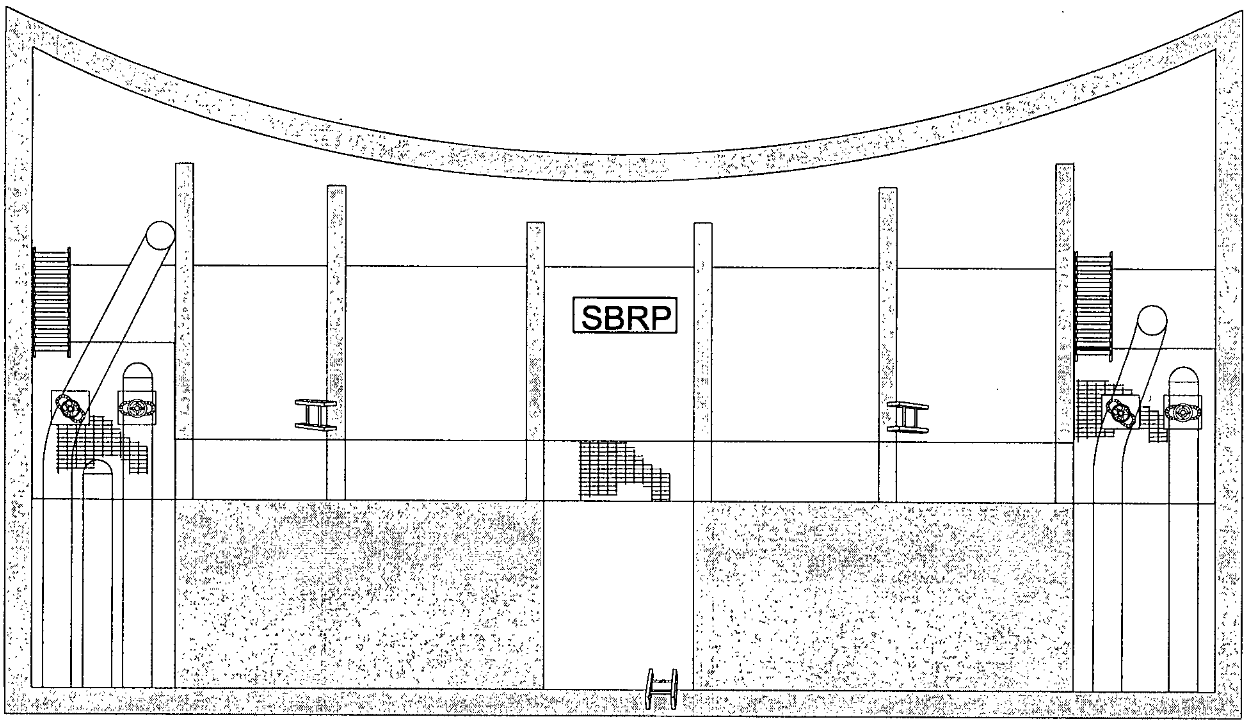


Level 4

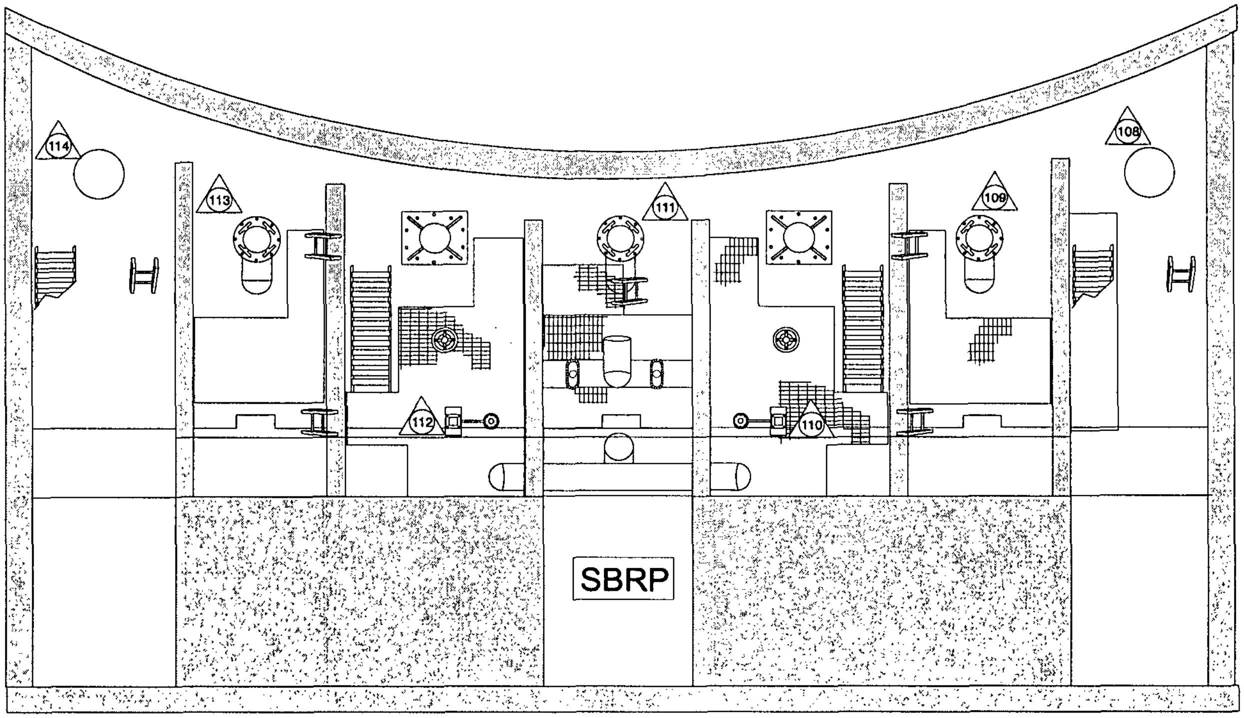


Level 5

MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Containment Building Mechanical Penetration Room Levels 4 & 5	Figure 3-8

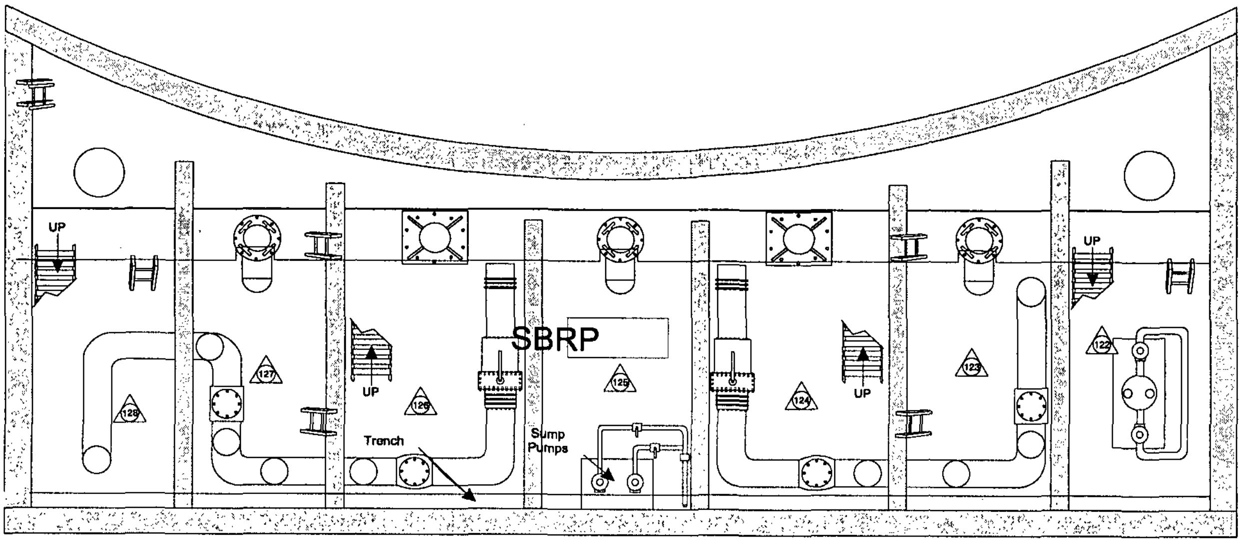


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Spray Buliding 4' Elevation	Figure 3-9



Spray Building
6' Elevation

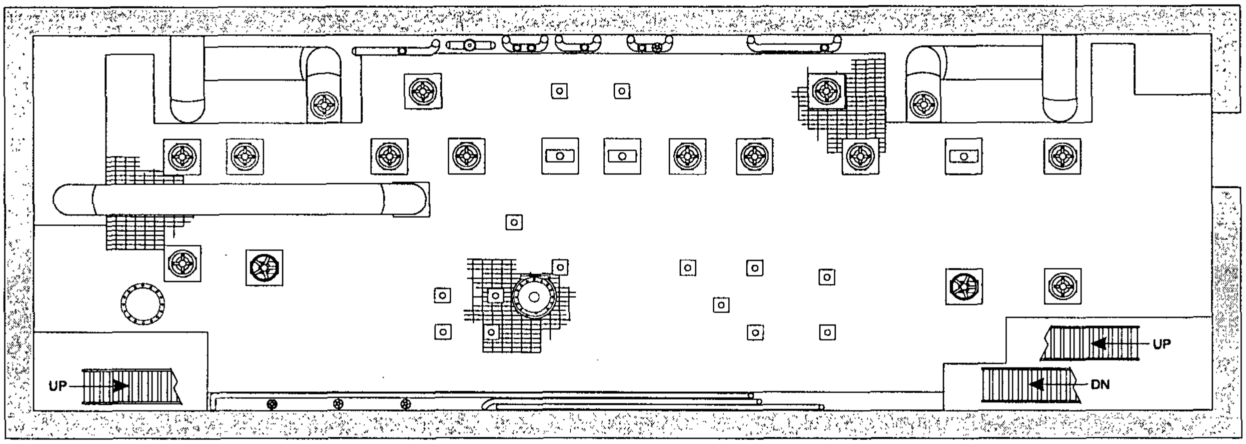
Figure 3-10



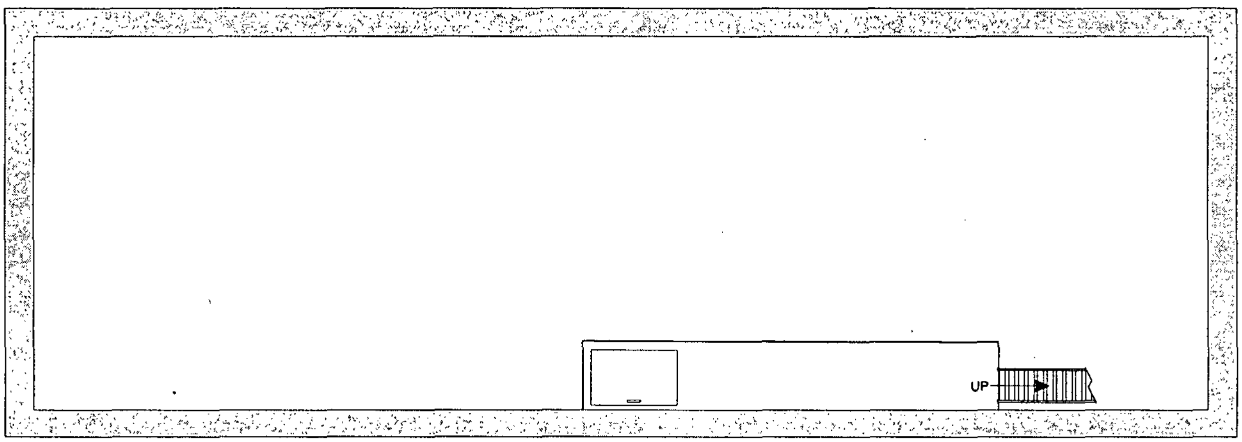
MAINE YANKEE
ATOMIC POWER CO.
LICENSE
TERMINATION PLAN

Spray Building
11' Elevation

Figure
3-11



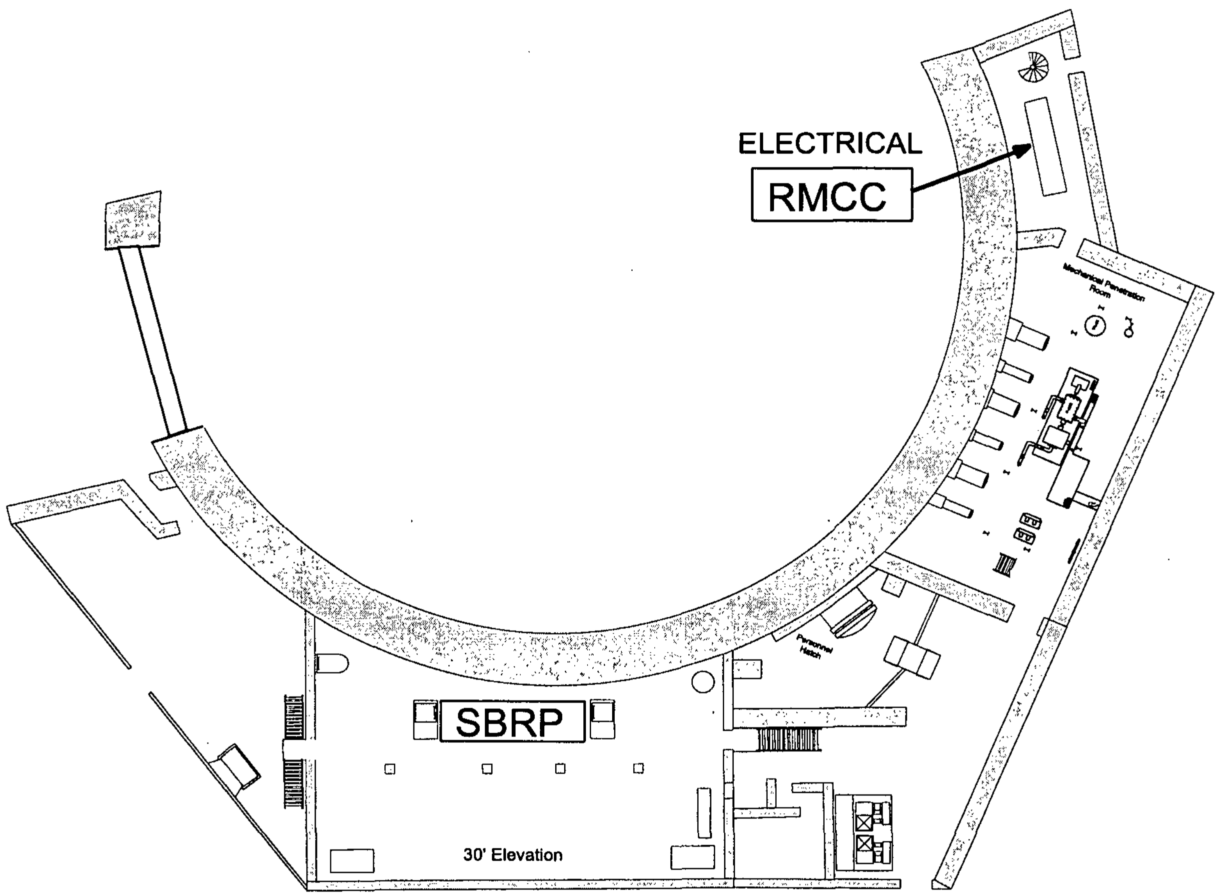
Elevation 21'-0"



Elevation 12'-0"

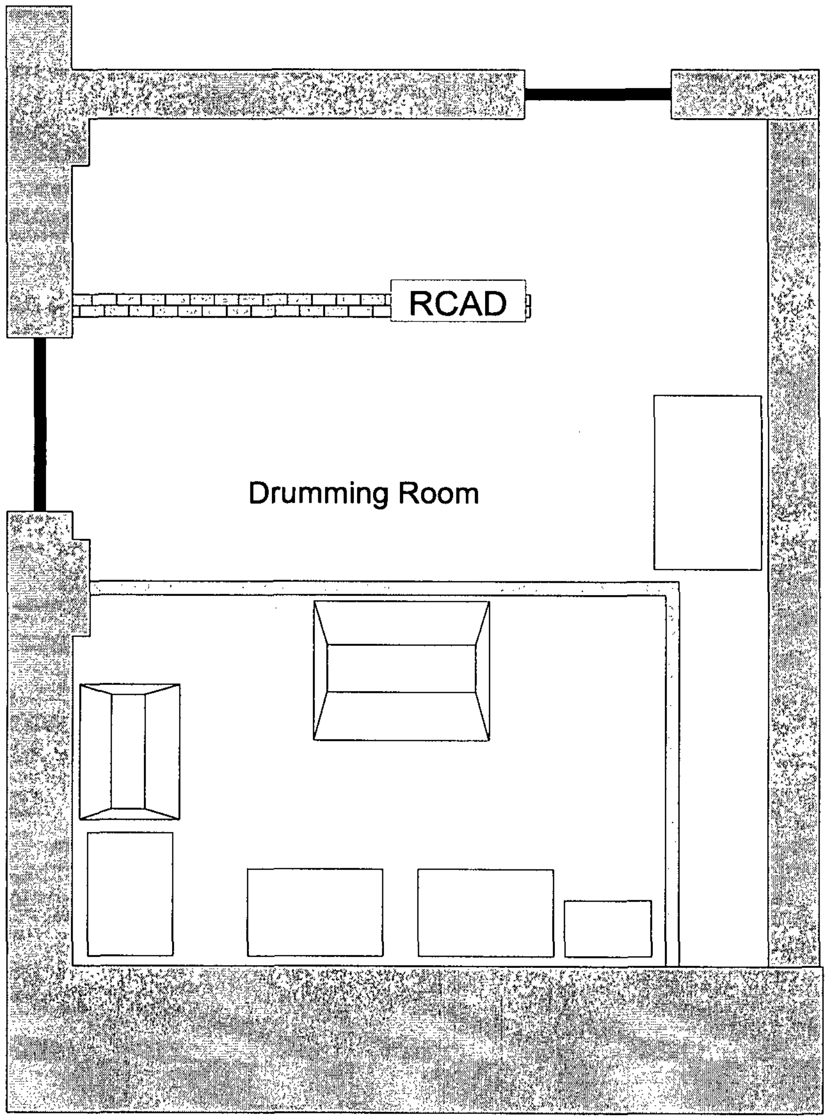
Spray Building
12' & 21' Elevation

Figure 3-12

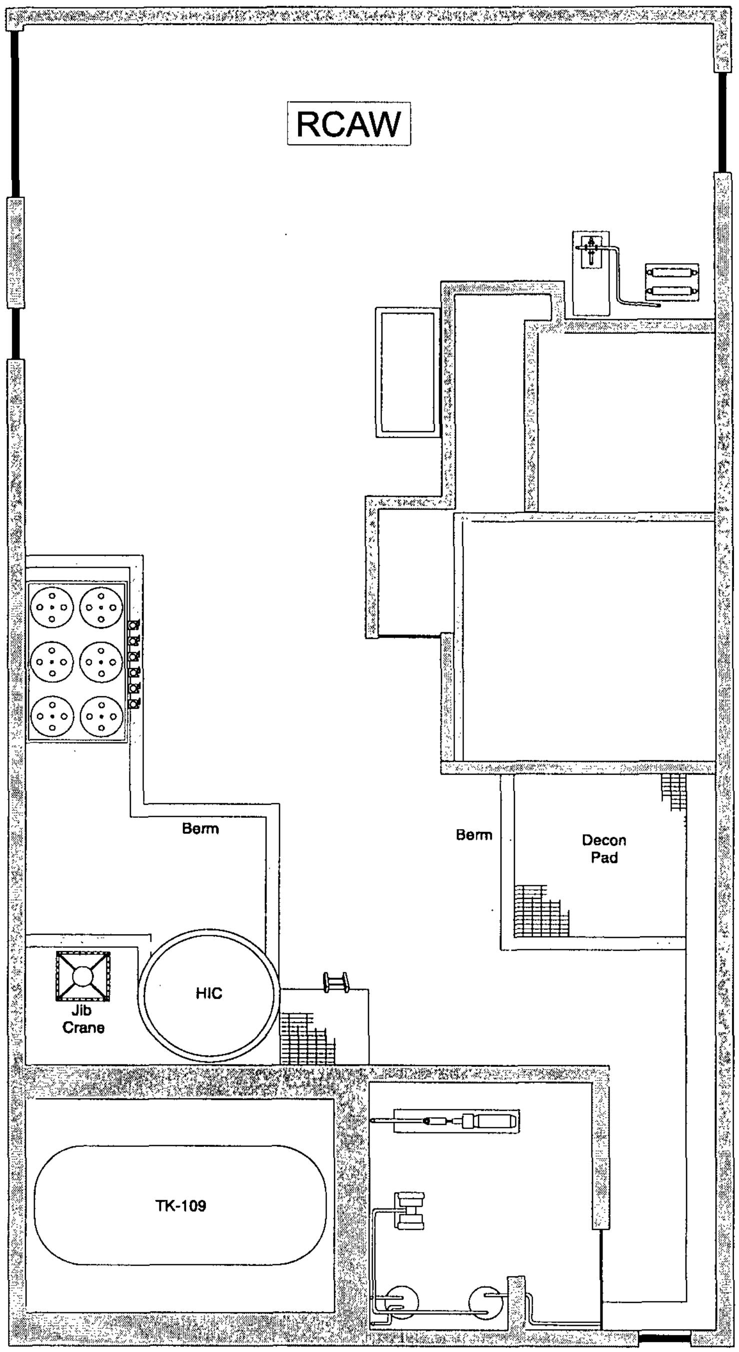


Spray Building
20' & 30' Elevations General Area

Figure 3-13

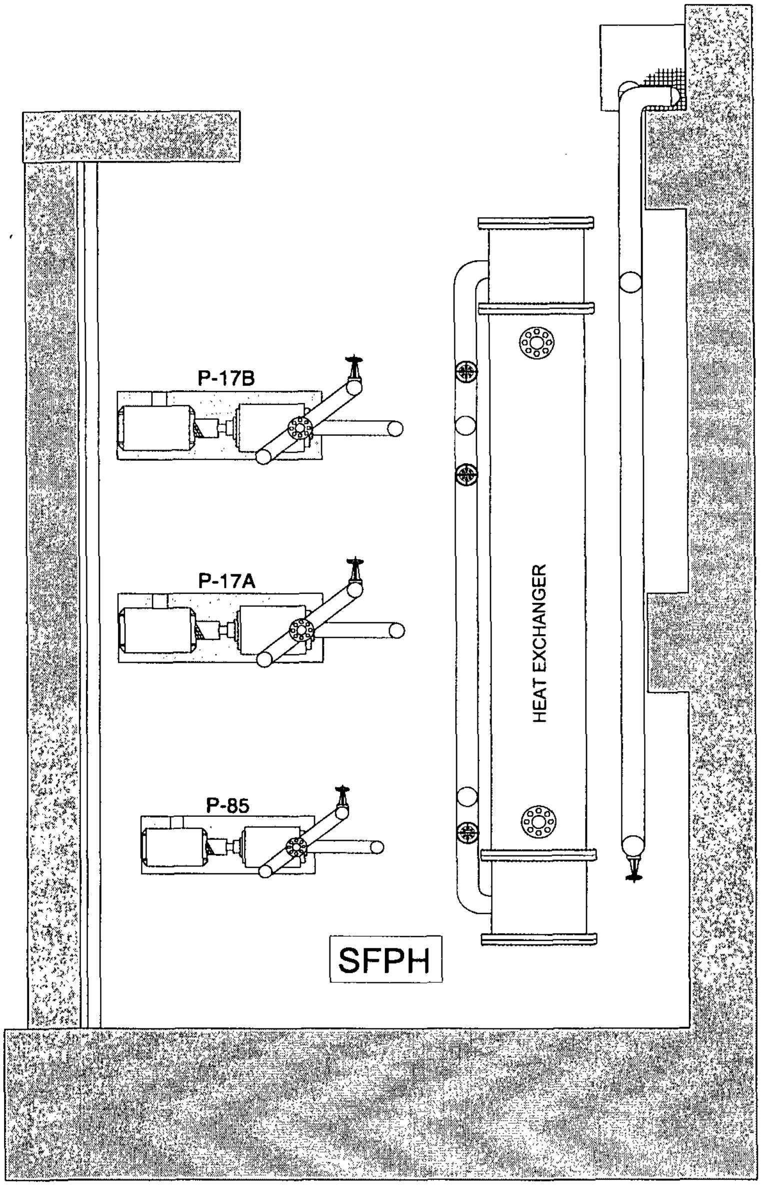


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Fuel Building 21' Elevation Decon Room	Figure 3-14



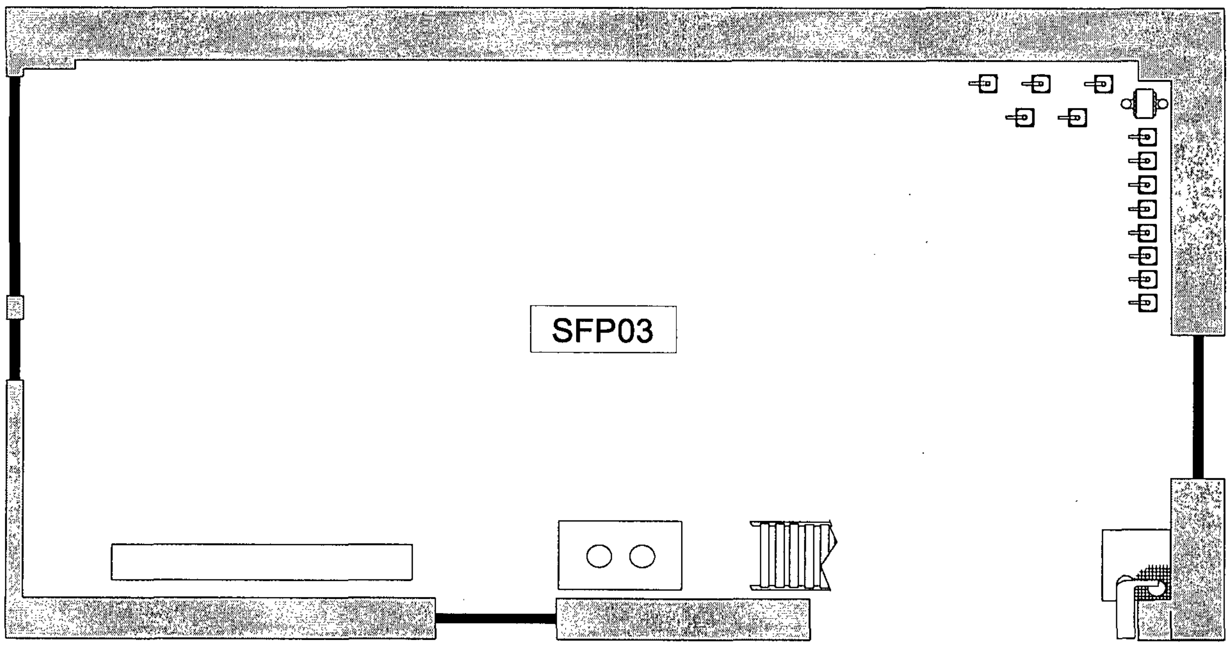
RCA Storage Building
Waste Solidification

Figure
3-15

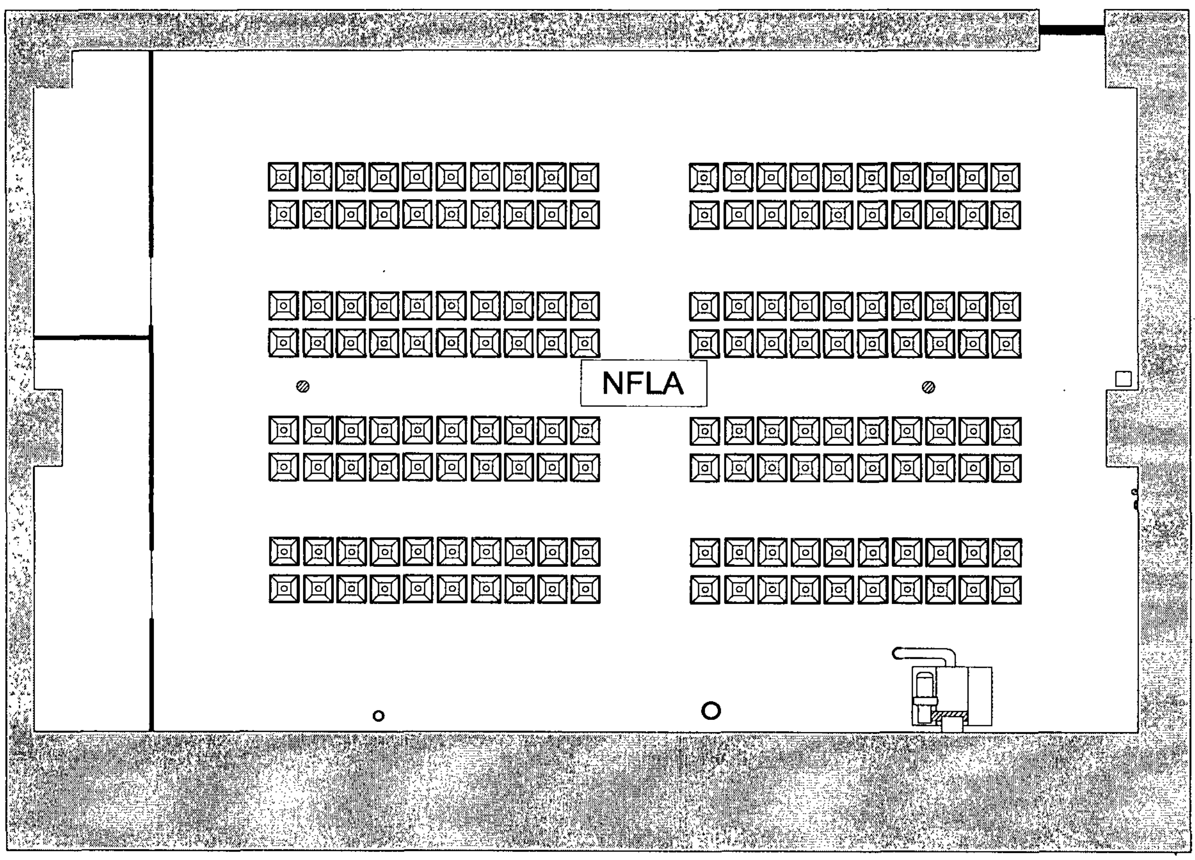


Fuel Building 21' Elevation
Spent Fuel Pool Heat Exchanger Area

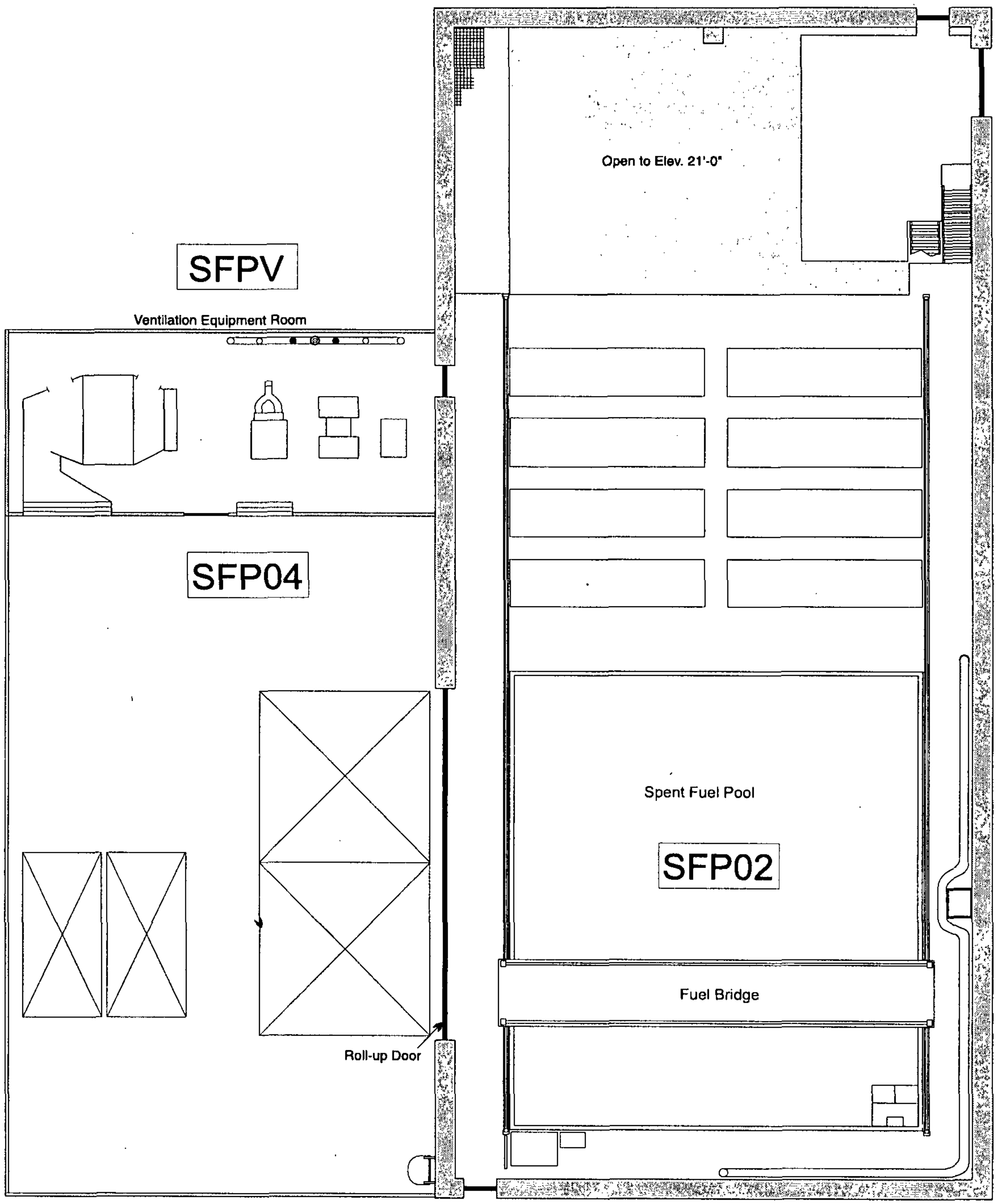
Figure 3-16



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Fuel Building 21'Elevation Fuel Laydown Area	Figure 3-17

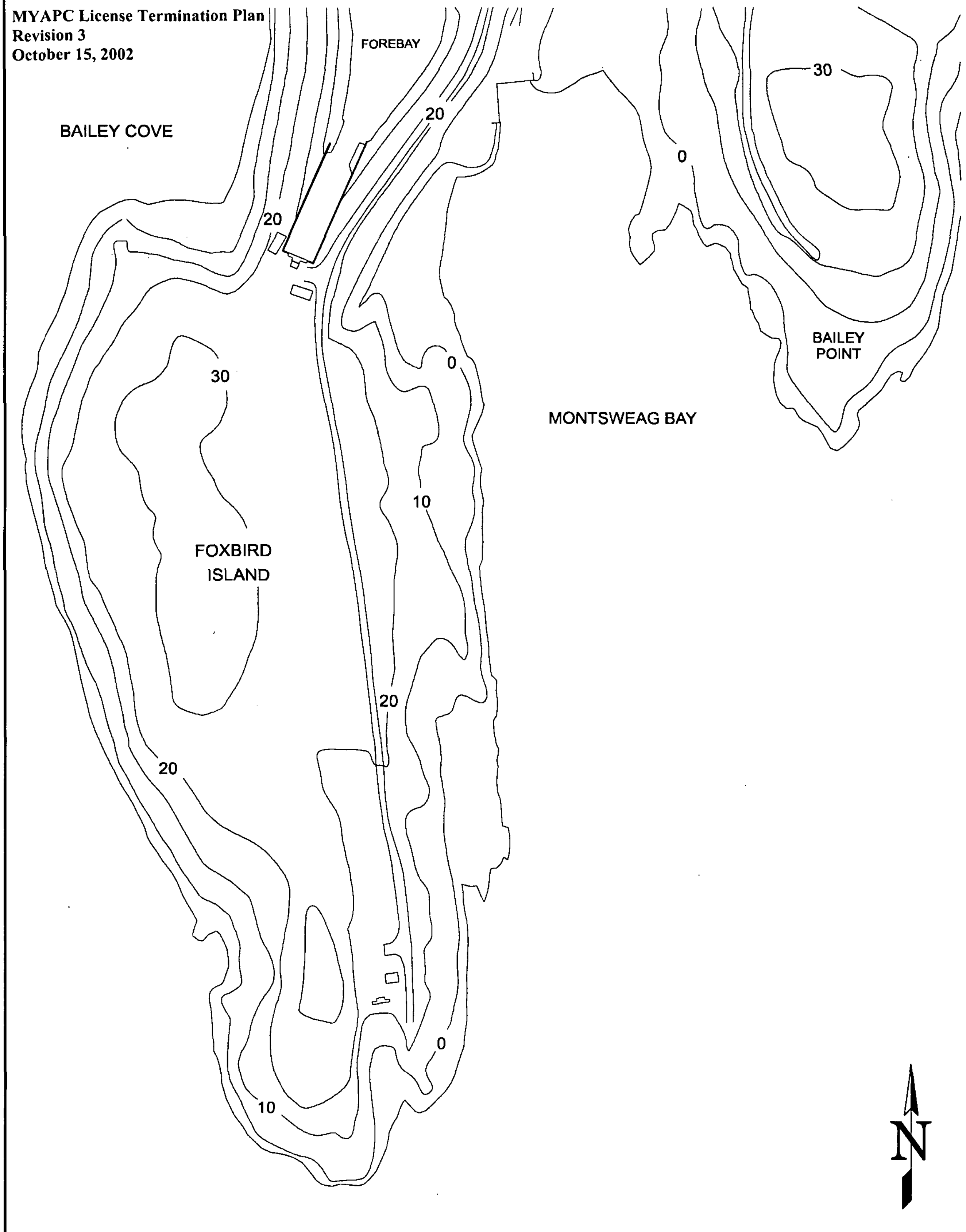


MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Fuel Building New Fuel Storage 31'-1 ½" Elevation	Figure 3-18



MAINE YANKEE ATOMIC POWER CO. LICENSE TERMINATION PLAN	Fuel Building 44'-6" Elevation	Figure 3-19

MYAPC License Termination Plan
Revision 3
October 15, 2002



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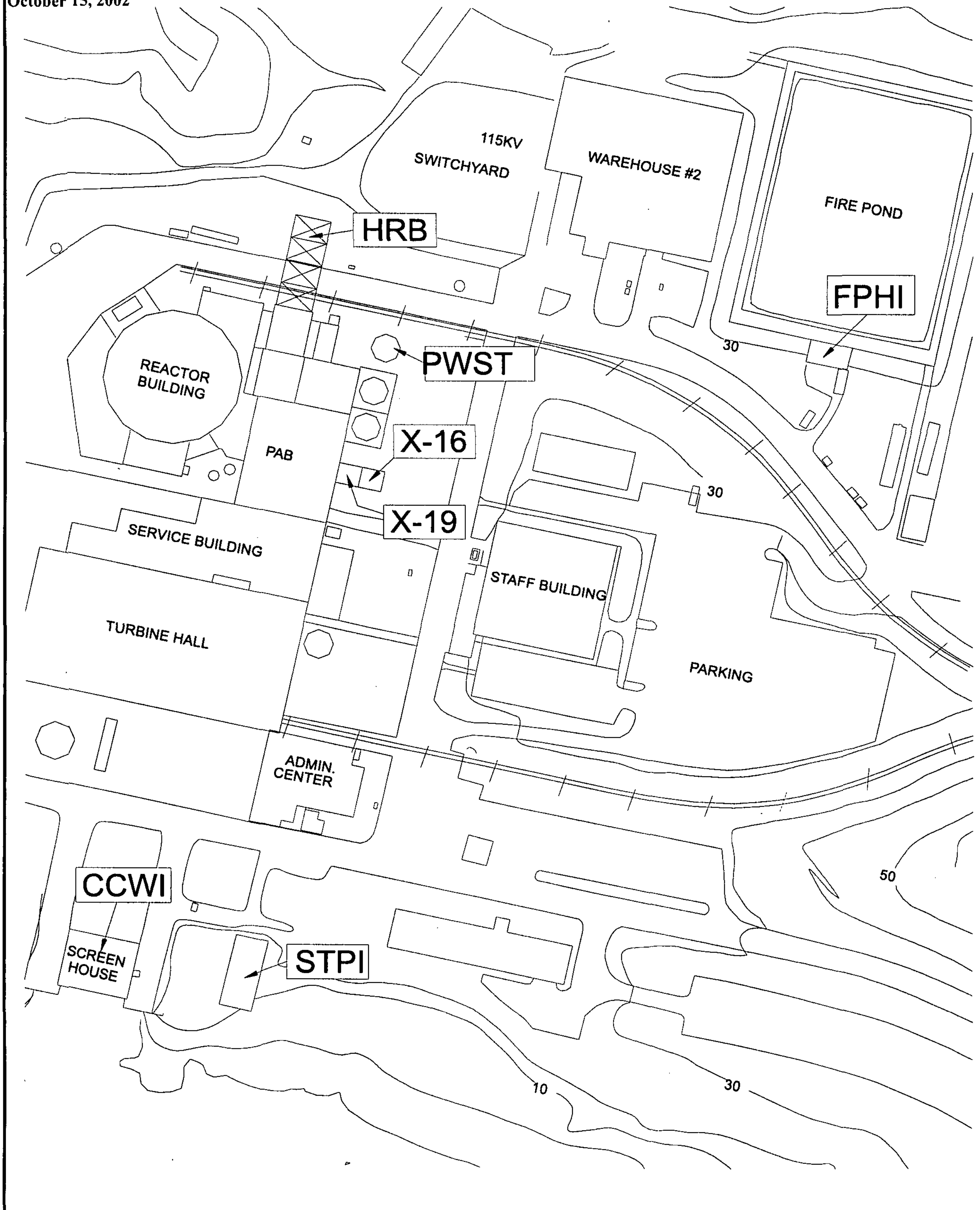

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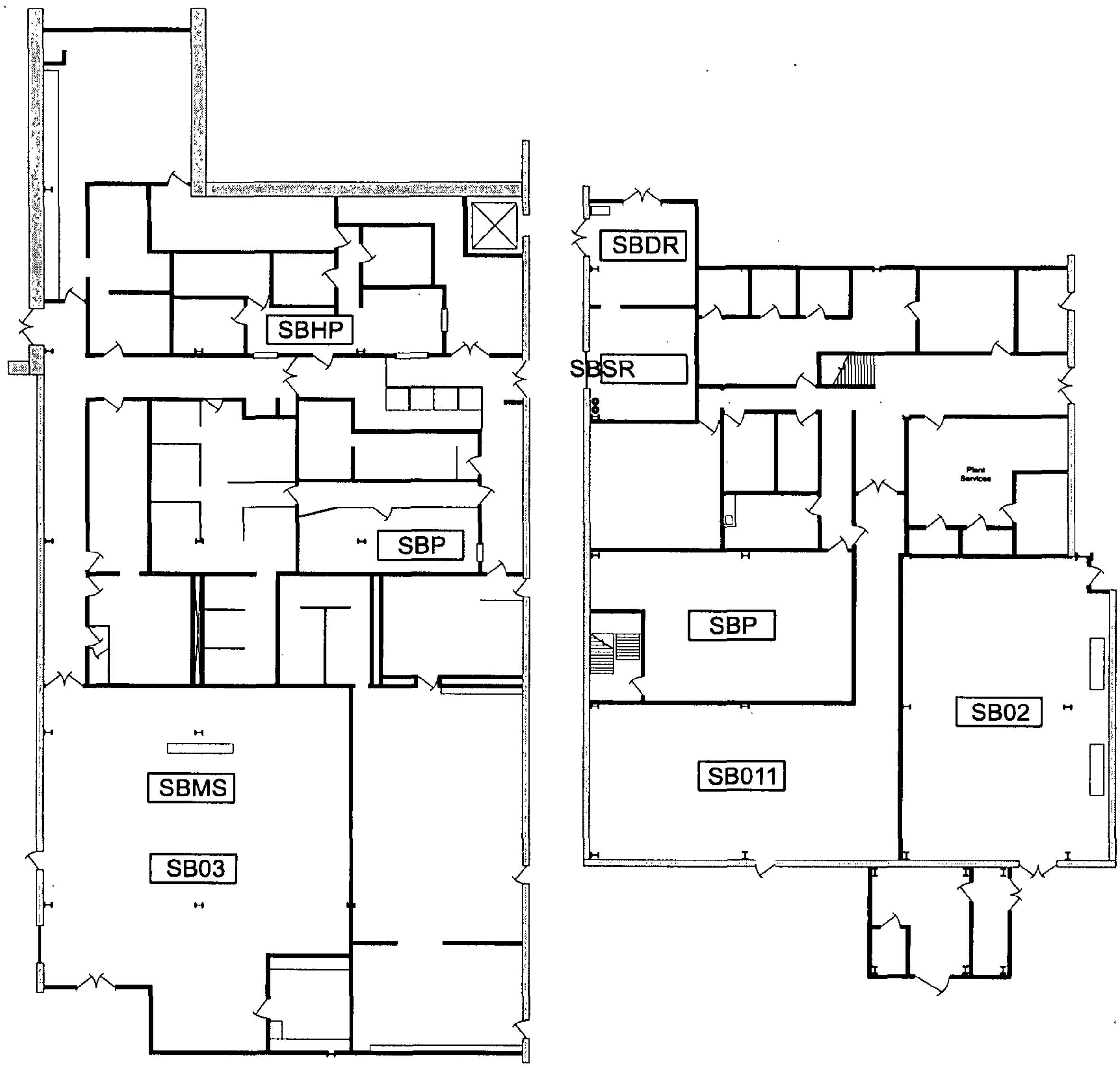

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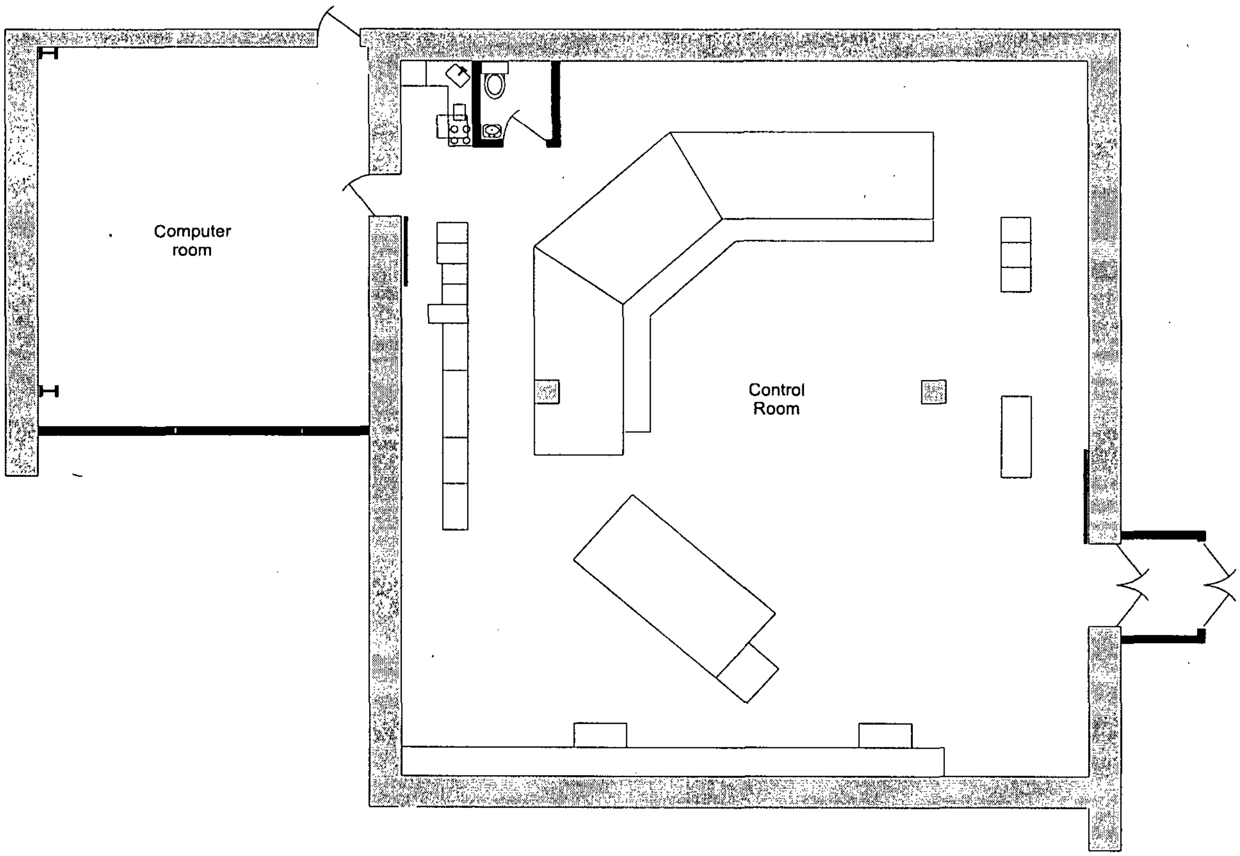

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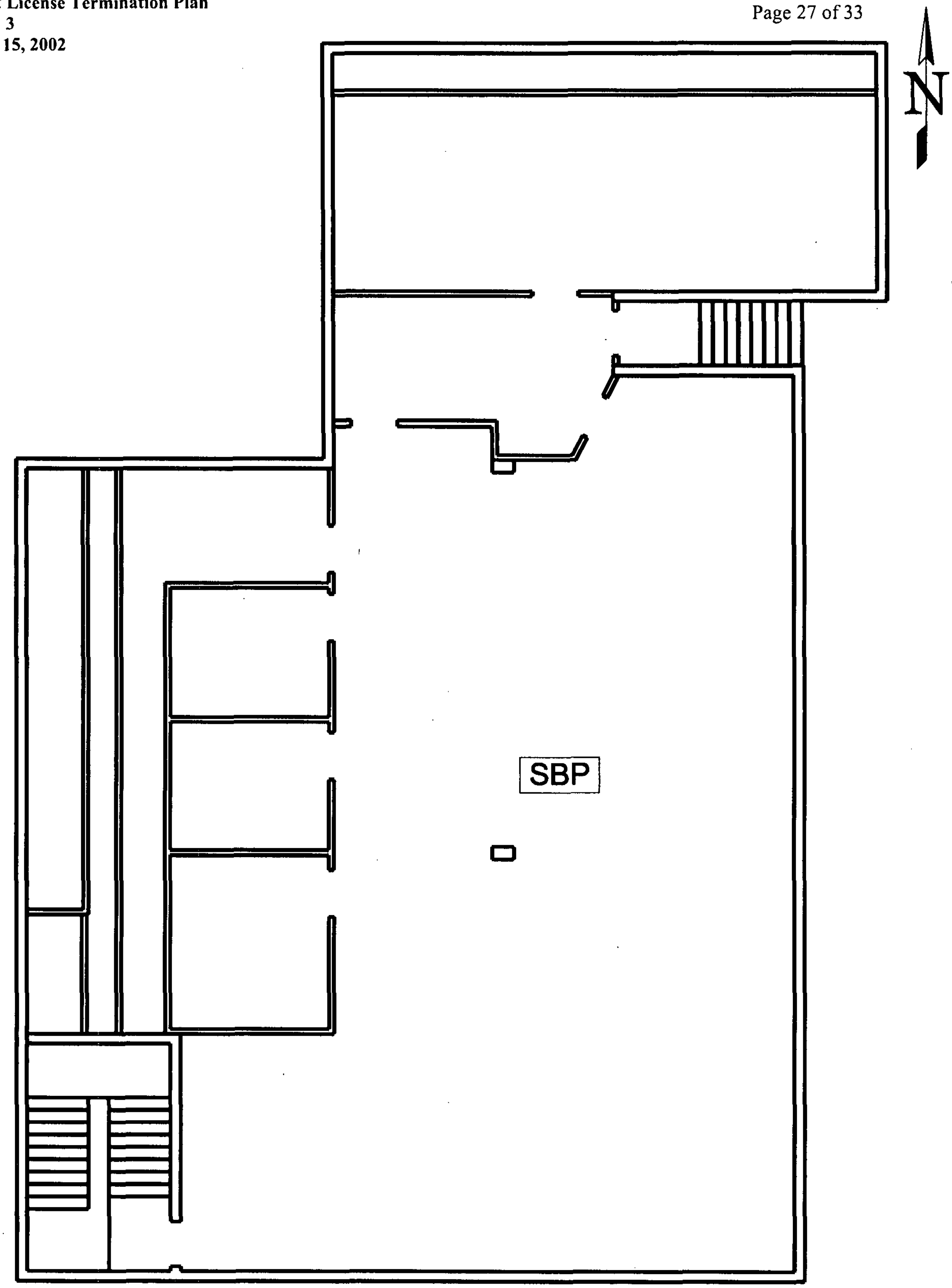

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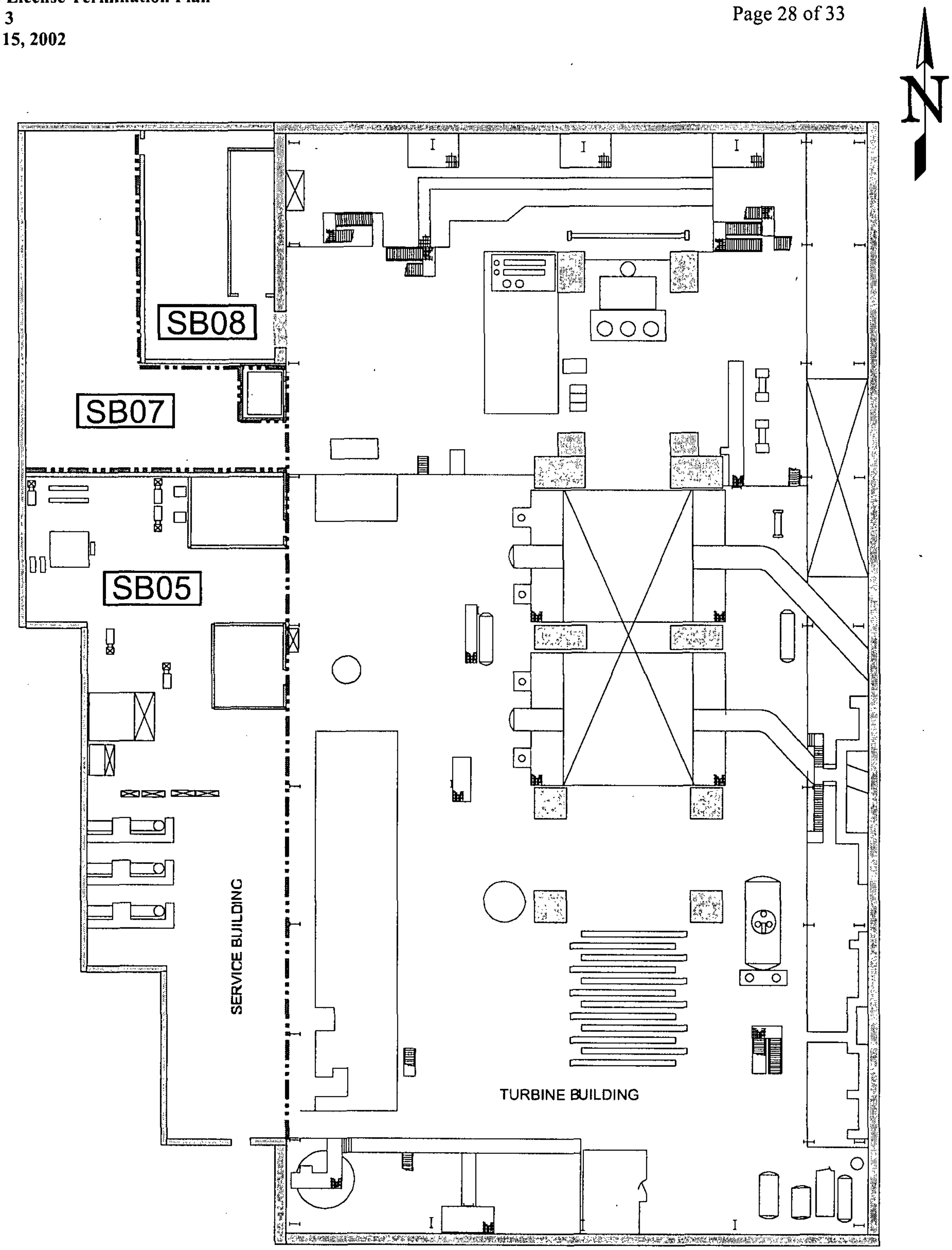

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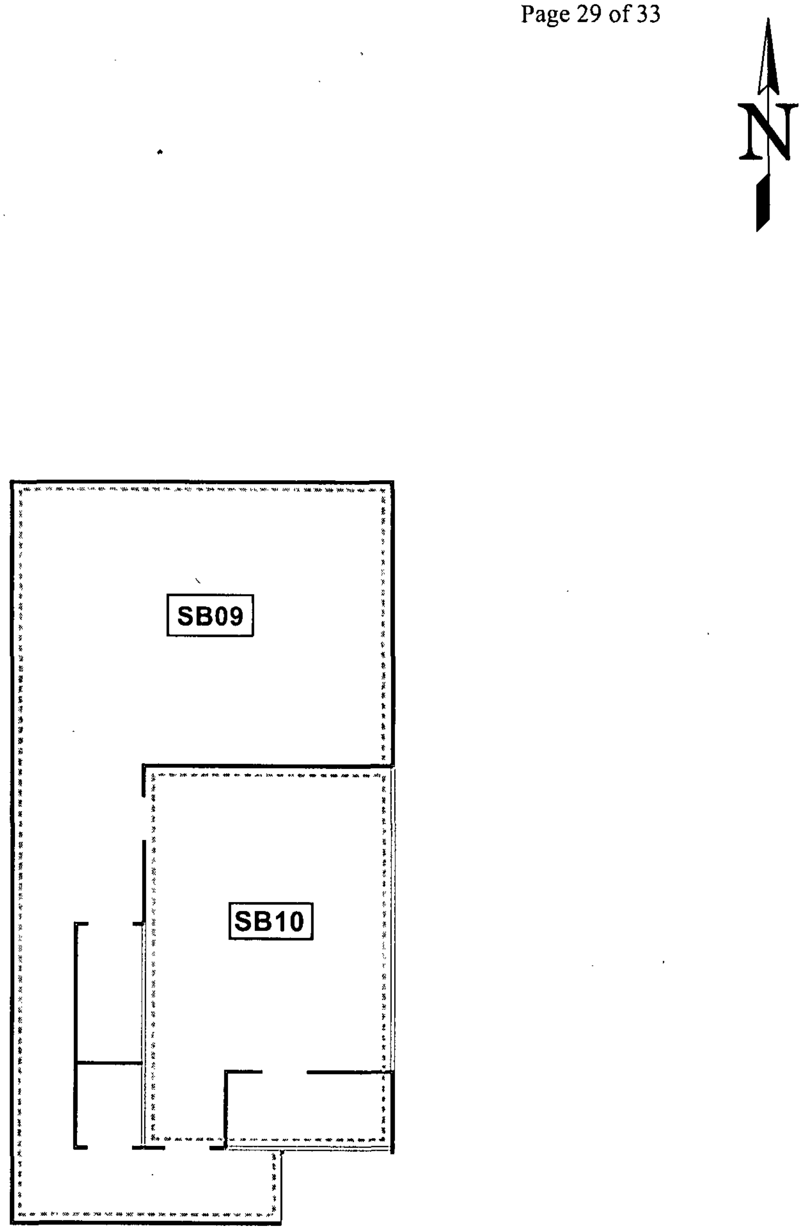

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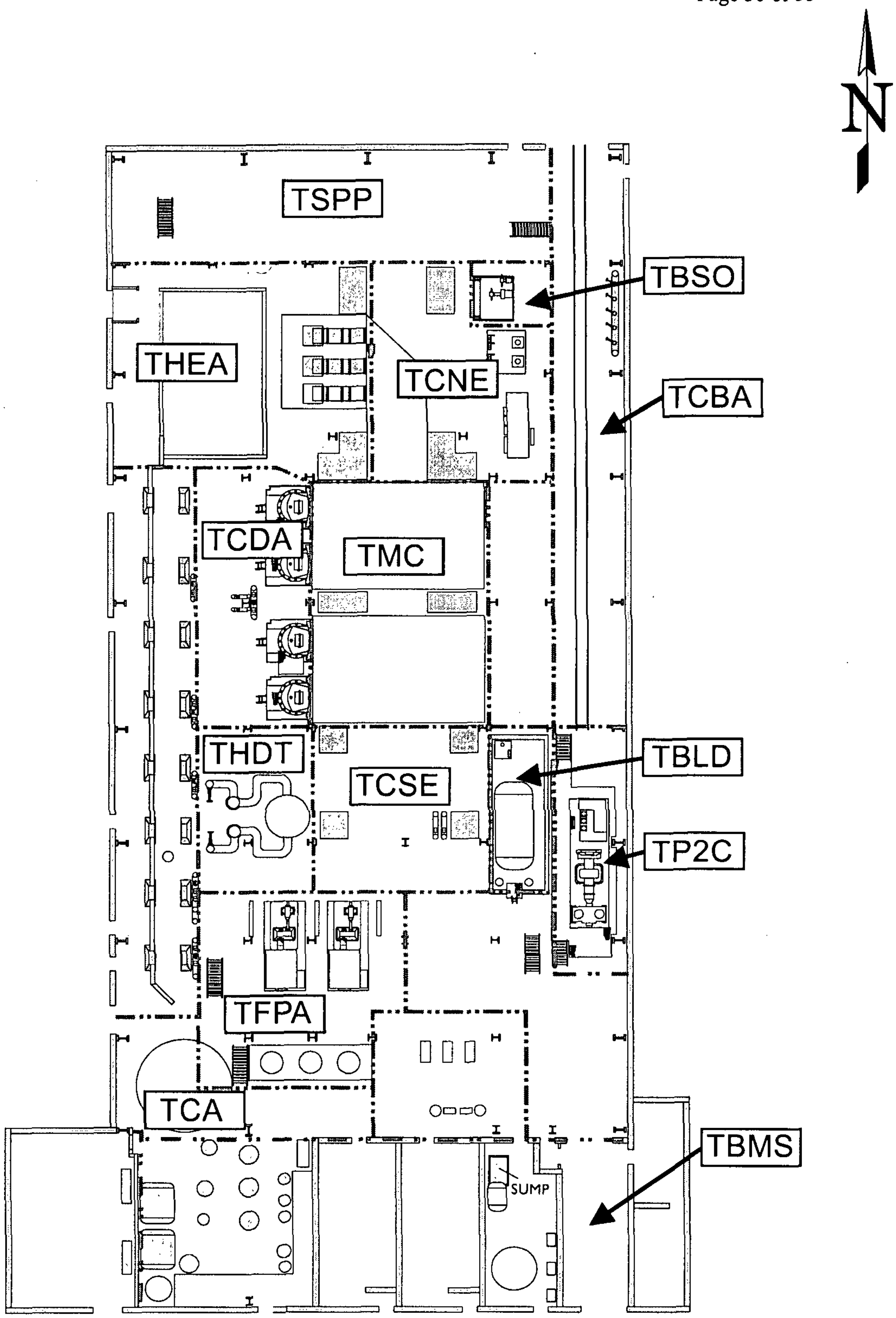

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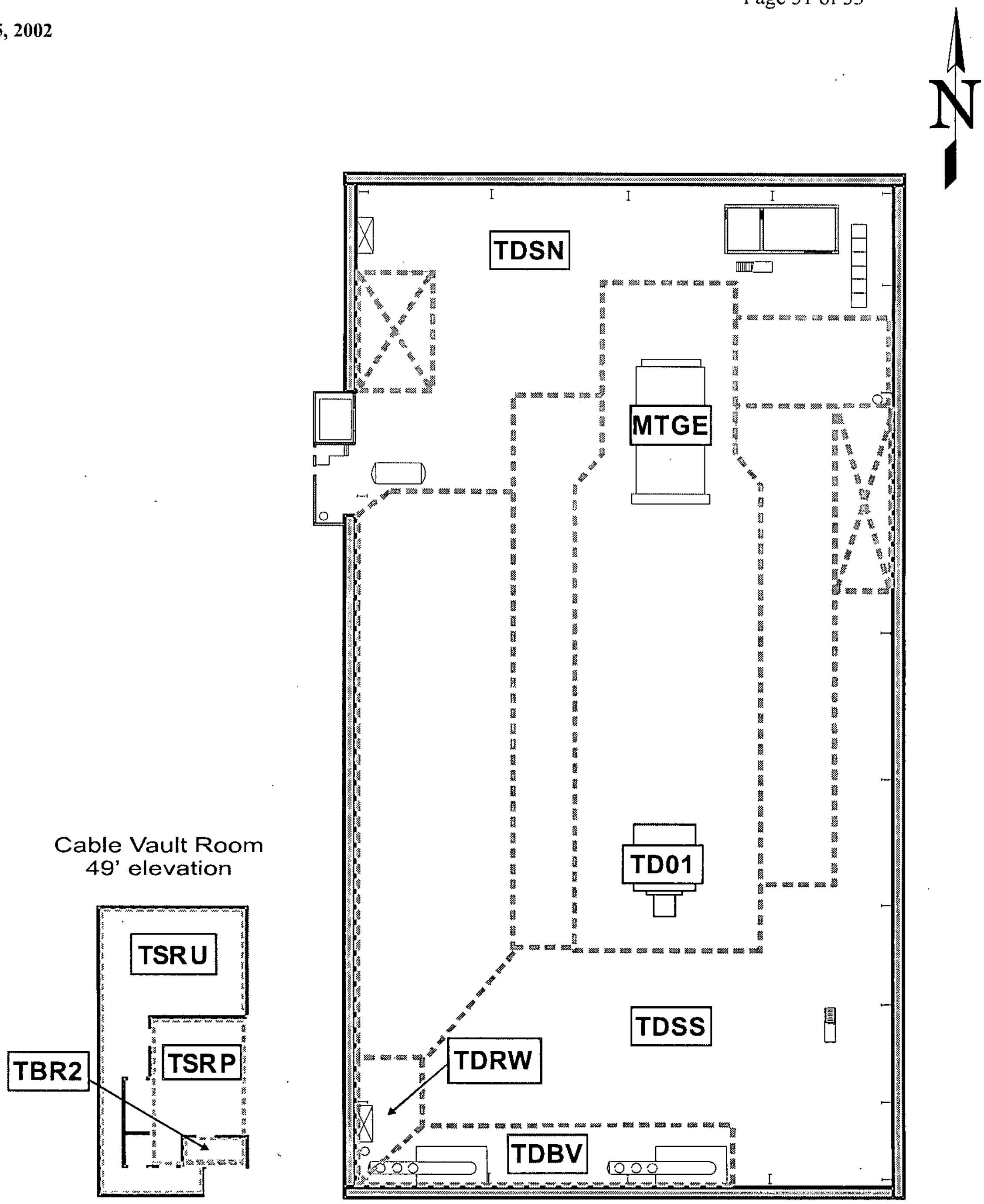

Turbine Building
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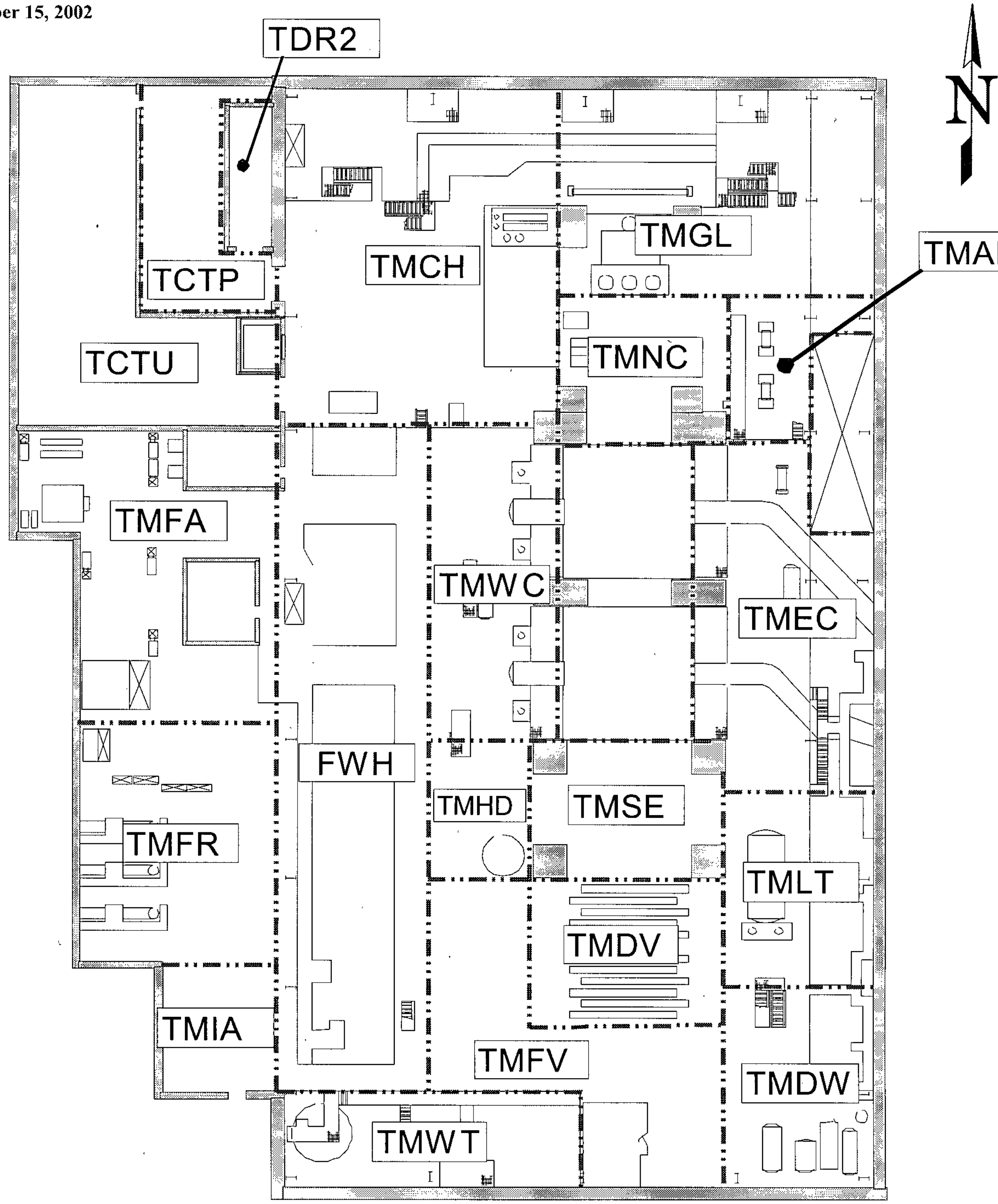

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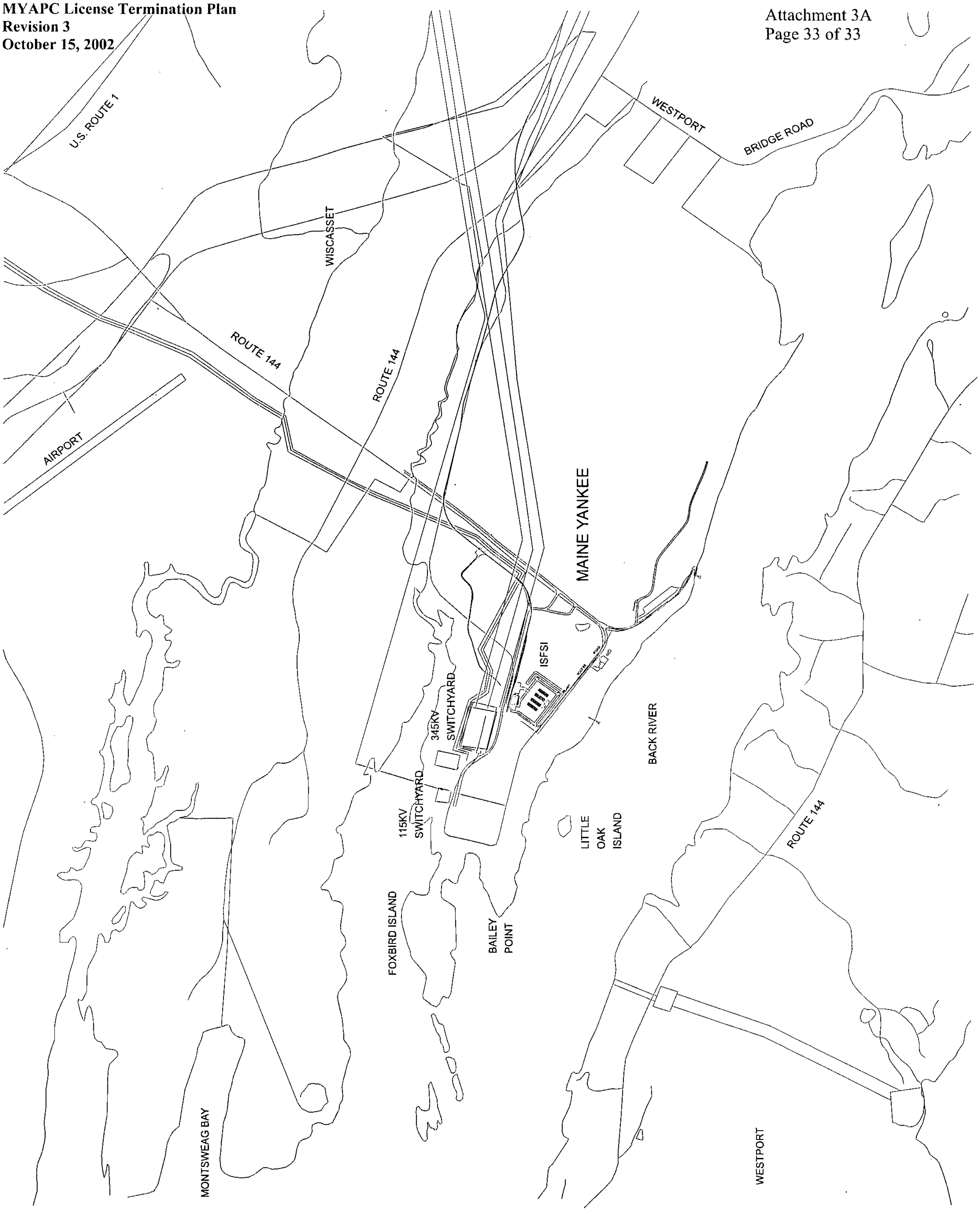

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MAINE YANKEE

LTP SECTION 4

SITE REMEDIATION PLAN

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4.0 SITE REMEDIATION PLAN

4.1 Remediation Actions and ALARA Evaluations

This section of the LTP describes various remediation actions which may be used during the decommissioning of MY. In addition, the methods used to reduce residual contamination to levels that comply with the NRC's annual dose limit of 25 mrem plus ALARA, as well as the enhanced State of Maine clean-up standard of 10 mrem/year or less for all pathways and 4 mrem/year or less for groundwater drinking sources, are described. Finally, the Radiation Protection Program requirements for the remediation are described.

4.2 Remediation Actions

Remediation actions are performed throughout the decommissioning process. The remediation action taken is dependent on the material contaminated. The principal materials that may be subjected to remediation are structure basements 3-feet below grade and soils. Attachment 4B of this section describes the equipment, personnel, and waste costs used to generate a unit cost basis for the remediation actions discussed below.

4.2.1 Structures

Following the removal of equipment and components, structures will be surveyed as necessary and contaminated materials will be remediated or removed and disposed of as radioactive waste. Contaminated structure surfaces at elevations less than 3-feet below grade will be remediated to a level that will meet the established radiological criteria provided in Section 6.0. The remediated building basements (elevations at and below - 3 foot below grade) will be backfilled.

Remediation techniques that may be used for the structure surfaces include washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting. Washing, wiping, abrasive blasting, vacuuming and pressure washing techniques may be used for both metal and concrete surfaces. Scabbling and chipping are mechanical surface removal methods that are intended for concrete surfaces. Activated concrete removal may include using machines with hydraulic-assisted, remote-operated, articulating tools. These machines have the ability to exchange scabbling, shear, chisel and other tool heads.

Scabbling

The principal remediation method expected to be used for removing contaminants from concrete surfaces is scabbling. Scabbling is a surface removal process that uses pneumatically-operated air pistons with tungsten-carbide tips that fracture the concrete surface to a nominal depth of 0.25 inches at a rate of about 20 ft^2 per hour. The scabbling pistons (feet) are contained in a close-capture enclosure that is connected by hoses to a sealed vacuum and collector system. The fractured media and dusts are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute HEPA (high efficiency particulate air filter) filtration devices. Dust and generated debris are collected and controlled during the operation.

Needle Guns

A second form of scabbling is accomplished using needle guns. The needle gun is a pneumatic air-operated tool containing a series of tungsten-carbide or hardened steel rods enclosed in a housing. The rods are connected to an air-driven piston to abrade and fracture the media surface. The media removal depth is a function of the residence time of the rods over the surface. Typically, one to two millimeters are removed per pass. Generated debris transport, collection, and dust control are accomplished in the same manner as for scabbling. Needle gun removal and chipping of media are usually reserved for areas not accessible to normal scabbling operations. These include, but are not limited to inside corners, cracks, joints and crevices. Needle gunning techniques can also be applied to painted and oxidized surfaces.

Chipping

Chipping includes the use of pneumatically operated chisels and similar tools coupled to vacuum-assisted collection devices. Chipping activities are usually reserved for cracks and crevices but may also be used in lieu of concrete saws to remove pedestal bases or similar equipment platforms. This action is also a form of scabbling.

Sponge and Abrasive Blasting

Sponge and abrasive blasting are similar techniques that use media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide, and walnut hulls. Sponge blasting is less aggressive incorporating a foam media that, upon impact and compression, absorbs contaminants. The medium is collected by vacuum and the contaminants washed from the medium for reuse.

Abrasive blasting is more aggressive than sponge blasting but less aggressive than scabbling. Both operations uses intermediate air pressures. Sponge and abrasive blasting are intended for the removal of surface films and paints. Abrasive blasting is evaluated as a remediation action and the cost is comparable to sponge blasting with an abrasive media.

Pressure Washing

Pressure washing uses a hydrolazer-type nozzle of intermediate water pressure to direct a jet of pressurized water that removes surficial materials from the suspect surface. A header may be used to minimize over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing.

Washing and Wiping

Washing and wiping techniques are actions that are normally performed during the course of remediation activities and will not always be evaluated as a separate ALARA action. When washing and wiping techniques are used as the sole means to reduce residual contamination below DCGL levels, ALARA evaluations are performed. Washing and wiping techniques used as a housekeeping or good practice measure will not be evaluated. Examples of washing and wiping activities for which ALARA evaluations would be performed include:

a. Decontamination of stairs and rails.

b. Decontamination of structural materials, metals or media for which decontamination reagents may be required.

c. Structure areas that do not provide sufficient access for utilization of other decontamination equipment such as pressure washing.

Washing and wiping is evaluated as a remediation action.

Grit Blasting

As the structures are demolished, contaminated piping will be removed and disposed of as radioactive waste. Any remaining contaminated piping in the below grade concrete may be remediated using methods such as grit blasting. Grit blasting uses grit media such as garnet or sand under intermediate air pressure directed through a nozzle that is pulled through the closed piping at a fixed rate. The grit blasting action removes the interior surface media layer of the piping. A HEPA vacuum system maintains the sections being cleaned under negative pressure and collects the media for reuse or disposal. The final system pass is performed with clean grit to remove any residual contamination.

Removal of Activated Concrete

Removal of activated concrete is intended to be accomplished using a machine-mounted, remote-operated articulating arm with exchangeable actuated hammer and bucket (sawing, impact hammering and expansion fracturing may also be employed). As concrete is fractured and rebar exposed, the metal is cut using flame cutting (oxygen-acetylene) equipment. The media are transferred into containers for later disposal. Dusts, fumes and generated debris are locally collected and as necessary, controlled using temporary enclosures coupled with close-capture HEPA filtration systems and controlled water misting. Any remaining loose media are removed by pressure washing or dry vacuuming using a HEPA filter equipped wet-dry vacuum.

The current remediation goal is to remove all activated concrete inside the containment liner. As shown in Section 6.0, the residual radioactivity due to activated concrete results in an annual dose to the critical group of less than 0.1 mrem (see Section 6.0, Table 6.9). This dose contribution to the total annual dose is a small fraction of the NRC and enhanced State dose limits and therefore ALARA evaluations are not deemed necessary. However, additional ALARA evaluations for activated concrete will be performed if the dose contribution to the critical group for activated concrete exceeds 1.0 mrem per year.

4.2.2 Soil

Soil contamination above the site specific DCGL will be removed and disposed of as radioactive waste. Operational constraints and dust control will be addressed in site excavation and soil control procedures. In addition, work package instructions for remediation of soil may include additional constraints and mitigation or control methods. The site characterization process established the location, depth and extent of soil contamination. As needed, additional investigations will be performed to ensure that any changing soil contamination profile during the remediation actions is adequately identified and addressed. A majority of site soil contamination is associated with three distinct areas (the PWST, RWST and the Shielded Radioactive Waste Storage Area) within the Radiologically Restricted Area (RRA). Sections 2.2.2 and 2.2.3 provide additional information regarding past and residual contamination associated with these areas. The information provided below generalize the anticipated activities associated with remediating these areas. For specific regions such as the area associated with the past soil contamination adjacent the RWST, remediation is expected to require removal and staging of overburden soils below the DCGL and the subsequent removal of deeper soils associated with this past contamination event. It should also be noted that soil remediation volume estimates in the LTP may vary from section to section, as appropriate, depending on their use, e.g., decommissioning cost estimates, ALARA evaluations, or dose assessment. Section 5.5.1.b discusses soil sampling and survey methods. The remediation of these areas will be performed following the removal of associated or adjacent tanks, components and pad interferences.

The contaminants within the RWST area are primarily due to past spill and heater leak incidents associated with the tank. With the exception of the area associated with clean soil overburden which was placed following remediation of the past contamination incident as stated in sections 2.2.2 and 2.2.3, soil remediation is expected to require removal of media to an average depth of approximately 1 meter immediately adjacent to the tank area. Additional remediation activities are expected to encompass a depth of 30 to 60 centimeters in the area down gradient from the tank and bounded east and west by local surface contour and the forebay berm.

Soil contamination near the PWST is due to the past storage of radioactively contaminated components and waste storage containers in the area immediately east and north of the PWST area. Local terrain features were such that associated contaminants subjected to weathering conditions would be transported toward the PWST area. The averaged soil remediation depth in this region is less than 60 centimeters.

Contaminated soil associated with the Shielded Radiological Waste Storage area originated, in part, from seasonal weathering conditions and specific tasks associated with components and stored containers. This area was evaluated in the past. A new bed of asphalt was placed over the region to mitigate the migration of any residual contaminants. The average soil contamination depth in this region is less than 60 centimeters.

Soil remediation equipment will include, but not be limited to, back and track hoe excavators. As practical, when the remediation depth approaches the soil interface region for unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique may be used. This simple methodology minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. Remediation of soils will include the use of established Excavation Safety and Environmental Control procedures which reference the required aspects of the Maine Erosion and

Sediment Control Handbook for Construction, Best Management Practices Manual. Additionally, soil handling procedures and work package instructions will augment the above guidance and procedural requirements to ensure adequate erosion, sediment, and air emission controls during soil remediation.

4.3 Remediation Activities Impact on the Radiation Protection Program

The Radiation Protection Program approved for decommissioning is similar to the Program in place during 25 years of commercial power operation. During power operations, contaminated structures, systems and components were decontaminated in order to perform maintenance or repair actions. The techniques used were the same as those being used for decommissioning. Many components were removed and replaced during operation. The techniques used for component removal were the same as those planned for use during decommissioning.

The Maine Yankee Radiation Protection Program adequately controlled radiation and radioactive contamination during decontamination and equipment removal processes. The same controls are being used during decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas. Decommissioning does not present any new challenge to the Radiation Protection Program above those encountered during normal plant operation and refueling. Decommissioning allows radiation protection personnel to focus on each area of the site and plan each activity well before execution of the remediation technique.

Low levels of surface contamination are expected to be remediated by washing and wiping. These techniques have been used over the operational history of the facility. Water washing with detergent has been the method of choice for large area decontamination. Wiping with detergent soaked or oil-impregnated media has been used on small items, overhead spaces and small hand tools to remove surface contaminants. These same techniques will be applied to remediation of lightly contaminated structure surfaces during remediation actions.

Intermediate levels of contamination and contamination on the internal surfaces of piping or components have been subjected to high-pressure washing, hydrolazing or grit blasting in the past. The refueling cavity has been decontaminated by both pressure washing and hydrolazing. Pipes, surfaces and drain lines have been cleaned and hot spots removed using hydrolazing, sponge blasting or grit blasting. Small tools, hoses and cables have been pressure washed in a self-contained glove box to remove surface contamination. These methods will be used to reduce contamination on moderately contaminated exterior surfaces as well as internal surfaces of pipes or components during decommissioning.

Scabbling or other surface removal techniques will reduce high levels of contamination, including that present on contaminated concrete. Concrete cutting or surface scabbling has been used at MY in the past during or prior to installation of new equipment or structures both outside and inside the RRA.

Abrasive water jet and mechanical cutting of components will be used to reduce the volume of reactor internals. Mechanical cutting was used at this facility during past operations. Abrasive water jet cutting uses actions similar to hydrolazing and grit blasting which have been used at the site in the past. The current radiation protection program provides adequate controls for these actions.

The decommissioning organization is experienced in and capable of applying these remediation techniques on contaminated systems, structures or components during decommissioning. The Radiation Protection Program is adequate to safely control the radiological aspects of this work and no changes to the Program are necessary in order to ensure the health and safety of the workers and the public.

4.4 ALARA Evaluation

As described in Section 6.0, dose assessment scenarios were evaluated for the residual contamination that could remain on basement surfaces and soils. The ALARA analysis is conservatively based on the resident farmer scenario. The resident farmer critical group applies to existing open land areas and all site areas where standing buildings have been removed to three feet below grade. Current decommissioning plans do not call for on site buildings to remain standing. However, consideration has been given to the potential value of the Staff Building. In view of this possibility, ALARA evaluations are also provided using the building occupancy scenario.

4.4.1 Dose Models

To calculate the cost and benefit of averted dose for the ALARA calculation, certain parameters such as size of contaminated area and population density are required. This information was developed as a part of the dose models described in Section 6 and the Final Survey Program in Section 5 and is summarized below.

a. Basement Fill Model (Resident Farmer Scenario)

As described in Section 6, after buildings and structures are removed to 3 feet below grade, the critical group is the resident farmer. Removal of residual radioactivity on basement surfaces 3 feet below grade reduces the dose associated with the resident farmer scenario. Accordingly, the ALARA evaluation for remediation actions uses the parameters for population density, evaluation time, monetary discount rate and area that are applicable to the resident farmer scenario.

b. Standing Building Occupancy Model

Although standing buildings are not planned to remain at the site, an ALARA evaluation was performed in the event plans change and a standing building will remain. In this case, the building occupancy scenario would be used. In accordance with Section 5.3 of the LTP, the building occupancy survey unit size is 180 m^2. This is based on a survey unit with a 100 m^2 floor area with contaminated walls to a height of 2 meters. ALARA cost analyses are based on an assumption that only the 100 m^2 floor area requires remediation. This is conservative since including the walls would increase remediation cost without increasing the benefit of averted dose.

4.4.2 Methods for ALARA Evaluation

NUREG-1727, "Decommissioning Standard Review Plan," Section 7.0, ALARA Analysis, states, "Licensees or responsible parties that remediate building surfaces or soil to the generic screening levels established by the NRC staff do not need to demonstrate that these levels are ALARA." The DCGLs for soil were based on generic screening levels. In addition, although no standing buildings are planned to remain, DCGLs were calculated and were also based on generic screening levels. Notwithstanding the NRC guidance, MY is conservatively providing ALARA evaluations of the remediation actions for soil and standing buildings. There are no generic screening levels for the basement fill scenario so ALARA analyses are required.

The ALARA evaluations were performed in accordance with the guidance in NUREG-1727. A spreadsheet format was used to account for the dose contribution of each radionuclide in the MY mixture. The principal equations used for the calculations are presented in Attachment 4A. The evaluation determines if the benefit of the dose averted by the remediation is greater or less than the cost of the remediaton. When the benefit is greater than the cost, additional remediation is required. Conversely when the benefit is less than the cost, additional remediation is not required.

4.4.3 Remediation Methods and Cost

For the Maine Yankee facility the remediation techniques examined are scabbling, pressure water washing, wet and dry wiping, grit blasting for embedded and buried piping and grit blasting of surfaces. The principal remediation method expected to be used is scabbling, which is intended to include needle guns and chipping. The total cost of each remediation method is provided in Attachment 4B. The cost inputs are defined in Attachment 4A, Section A.2, Calculation of Total Cost. Basement concrete is the principal surface that will require remediation.

a. Basement Concrete Surfaces

The characterization data for concrete surfaces at the Maine Yankee facility indicates that a major fraction of the contamination occurs in the top millimeter of the concrete. Scabbling actions result in the removal of the top 0.125 to 0.25 inches (0.318 to 0.635 cm) of concrete. The ALARA evaluation was performed by bounding the cost estimate for a scabbled depth of 0.125 and 0.25 inches. For each evaluation the same manpower cost is used. However, the manpower and equipment costs for the lower bounding depth do not include compressor and consumable supply costs which adds some conservatism to the cost estimate, i.e., bias the cost low. The major variables for the bounding conditions are the costs associated with manpower and waste disposal.

b. Structure Activated Concrete

Concrete activation is associated with the containment structure. Characterization of the reactor bioshield and loop area concrete has provided information regarding the identification, concentration, and distribution of the radionuclides. In addition to the observed concrete activation products, the concrete surfaces in the containment structure are radioactively contaminated by the deposition and transport of fluids and airborne distribution which occurred during plant operation. The current remediation goal is to remove all activated concrete inside the containment liner. This region comprises approximately 21 m^2 of floor surface that is hampered by accessibility and equipment staging factors.

4.4.4 Remediation Cost Basis

The cost of remediation depends on several factors such as those listed below. This section describes the attributes of each remediation method that affect cost. The detailed cost estimates for each method are provided in Attachment 4B.

- Depth of contaminants;
- Surface area(s) of contamination relative to total;
- Types of surfaces: vertical walls, overhead surfaces, media condition;
- Consumable items and equipment parts;
- Cleaning rate and efficiency (decontamination factor);
- Work crew size;
- Support activities such as, waste packaging and transfer, set up time and interfering activities for other tasks; and
- Waste volume.

a. Scabbling

It has been estimated that scabbling can be effectively performed on smooth concrete surfaces to a depth of 0.25 to 0.5 inches at a rate of 20 ft^2 per hour. The scabbling pistons (feet) are contained in a close-capture enclosure that is connected by hoses to a sealed vacuum and collector system. The waste media and dust are deposited into a sealed removable container. The exhaust air passes through both roughing and absolute HEPA filtration devices. Dust and generated debris are collected and controlled during the operation.

The operation is conservatively assumed to be performed by one equipment operator and one laborer. In addition, costs for radiation protection support activities and supervision are included.

The unit cost is presented in Table 4-1. Scabbling the room assumes that 100% of the concrete surface contains contamination at levels equal to the DCGL and that 95% of this residual activity is removed by the remediation action. The equipment is capable of scabbling 20.0 square feet per hour. The debris is vacuumed into collectors that are transferred to containers for rail shipments. For the evaluation, the rail car is assumed to carry 92 m^3 of concrete per shipment.

The assumed contamination reduction rates are very high (95%), but not unreasonable considering that the contamination is very close to the surface. Based on evaluation of concrete core samples, scabbling is expected to be the principal method used for remediation of concrete surfaces. The cost elements used to derive the unit costs for the ALARA evaluation are listed in Attachment B. The methods for calculating total cost are provided in Attachment A.

b. Pressure Water Washing

The unit costs provided in Table 4-1 for water washing were established by assuming that 100% of the site structures' surface area is pressure washed. This information was used to provide a cost per meter square factor. Attachment 4B provides the cost details. The equipment consists of a hydrolazer and when used, a header assembly. The hydrolazer type nozzle directs the jet of pressurized water that removes surficial materials from the concrete. The header minimizes over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing. The cleaning speed is approximately 9.3 square meters(100 ft^2) per hour and the process generates about 5.4 liters of liquid per square meter (NUREG-5884, V2). The contamination reduction rates are dependent on the media in which the contaminants are fixed, the composition of the contaminants, cleaning reagents used and water jet pressure. Mitigation of loose contaminants is high. Reduction of hard-to-remove surface contamination is approximately 25% for the jet pressure and cleaning speed used. The use of reagents and slower speeds can provide better contamination reduction rates but at proportionally higher costs. The operation is performed using one equipment operator and two laborers. In addition, costs for radiation protection support activities and supervision are included. The formula associated with the cost elements is provided in Attachment A and the cost elements are provided in Attachment B.

c. Wet and Dry Wiping

The unit costs provided in Table 4-1 for washing and wiping assume 100% of the site structures' surface area is washed and wiped. The information is used to develop a cost per square meter. Attachment 4B provides the detailed costs. Wet wiping consists of using a cleaning reagent and wipes on surfaces that cannot be otherwise cleaned or decontaminated. Dry wiping includes the use of oil-impregnated media to pick up and hold contaminants. The cleaning rate of these actions is estimated at 2.8 square meters per hour (~ two minutes per square foot). This action is labor intensive. The action is effective for the removal of loose contaminants and reduction of surface contaminants, especially when cleaning reagents are used. Waste generation is about 0.005 m^3 per hour (NUREG-5884, V2). Decontamination factors vary and are dependent on factors such as the reagents that are used, the level of wiping effort and the chemical and physical composition of the contaminant. The contamination reduction efficiency used for wet and dry wiping is 20 percent. Removal of loose contaminants, oil and grease is very effective (100 percent). The operation is performed using two laborers. In addition, the cost for radiation protection support activities includes an operating engineer and supervision. The formula associated with the cost elements is provided in Attachment A. Attachment B list the cost elements used for the evaluation.

d. Grit Blasting (Embedded Piping)

The cost for grit blasting was established by assuming that 6,158 linear feet of piping is decontaminated. This length of piping is the total amount of potentially contaminated buried and embedded piping identified by the Maine Yankee engineering group. For the evaluation, the entire interior surface is assumed to require decontamination and the internal diameter is assumed at 4 inches (typical drain line dimensions). The grit blasting system is comprised of a hopper assembly that delivers a grit medium (garnet or sand) at intermediate air pressures through a nozzle that is pulled at a fixed rate (~1 foot per minute) through the piping. A HEPA vacuum system maintains the piping system under a negative pressure and collects the grit for reuse (cyclone separator) or disposal. Usually several passes are required to effectively clean the piping to acceptable residual radioactivity levels. The contamination reduction efficiency used for grit blasting is 95 percent. This reduction rate can vary depending on radial bends in piping, reduction and expansion fittings, pipe material composition, physical condition and the plate-out mechanisms associated with the contaminants and effluents. The final pass is made with clean grit to mitigate the possibility of loose residual contaminants associated with previous cleaning passes. Grit decontamination factors are related to pressure, nozzle size, grit media and the number of passes made. A nominal grit usage rate of one pound per linear foot is used in the calculation. This cost unit information is provided as cost per linear foot factor and is also converted to m^2 for the spreadsheet evaluation. Attachment 4B provides the cost details used to derive unit cost. The formula associated with the cost elements is provided in Attachment A

e. Sponge and Abrasive Blasting

Sponge and abrasive blasting uses media or materials coated with abrasive compounds such silica sands, garnet, aluminum oxide and walnut hulls. The operation uses intermediate air pressures as that described for grit blasting. The operation uses a closed-capture system and air filtration system to mitigate loose and airborne radioactivity. The system includes a cyclone or similar separation system to collect the generated media. The operation is intended for removal of surficial films. The removal efficiency and depth are a function of the surface, abrasive mix, air pressure, grit media, and speed or number of passes performed over the suspect surface. Surface cleaning rates are about 30 square feet per hour. For the rate given, the removal depth using aluminum oxide grit will range from less than 1 to as much as 3 millimeters. Abrasive blasting techniques are often used for film and paint removal and are less aggressive than scabbling.

f. Soil Excavation

The unit costs provided in Table 4-1 for soil excavation were established by assuming 4.96E+04 ft^3 (1403.0 m^3) of soil is excavated from the site. This information was used to generate a cost per cubic meter for soil remediation. The equipment consists of an excavator that first moves the soil at the contaminated depth interface into a container or if necessary, a pile that is scooped into a staged shipping container. When filled, the container is moved from the excavation area with a forklift. Contamination reduction is assumed at 95%. The operation is performed using two equipment operators and two laborers. Costs for radiation protection support activities and supervision are also included. The formula associated with the cost elements is provided in Attachment A and the cost elements are provided in Attachment B.

4.5 Unit Cost Estimates

In order to effectively perform ALARA evaluations and remediation actions, unit cost values are required. These values are used to perform the NUREG-1727 cost-benefit analysis. Table 4-1 lists the unit costs of the remediation methods anticipated to be used at Maine Yankee.

The spreadsheets and information used to calculate values in Table 4-1 are summarized in Attachment 4B.

4.6 Benefit of Averted Dose

The remediation costs listed in Table 4-1 were compared to the benefit of the dose averted through the remediation action. The benefit of averted dose was calculated using Equations D1 and D2 in NUREG-1727 as modified to account for multiple radionuclides. The parameters used in the equations were taken from NUREG-1727, Table D2.

Table 4-1
Unit Cost Estimates

Remediation Technique	Unit Cost[a]	Remarks
Pressure Washing and Vacuuming	\$19.32/m^2	Unit cost factors provided in Attachment B
Wiping/Washing[a]	\$48.59/m^2	Unit cost factors provided in Attachment B
Concrete Scabbling[b] (Upper Bound)	\$106.23/m^2	Unit cost factors provided in Attachment B. Needle gun activities are included with scabbling
Concrete Scabbling (Lower Bound)	\$91.49/m^2	Unit cost factors provided in Attachment B. Needle gun activities are included with scabbling
Grit Blasting Surfaces (Upper Bound)	\$113.18/m^2	Unit cost factors provided in Attachment B
Grit Blasting Surfaces (Upper Bound)	\$87.80/m^2	Unit cost factors provided in Attachment B
Grit Blasting Embedded/Buried Piping	\$45.93/linear ft	Unit cost factors provided in Attachment B
Soil Excavation	\$1837/m^3	Unit cost factors provided in Attachment B

[a]The high cost for wiping and washing is due both to the labor intensive time (76% of the total) required and the costs of waste processing and disposal associated with the water used. Because radiation protection practices depict wiping as good practice for removing loose contamination, wiping is performed and not always as a function of an ALARA evaluation

[b]A contingency of 25% has been added to the person hour total for the activities

Combining Equations D1 and D2 results in the following. The method for adjusting this equation to account for multiple radionuclides is described in Attachment 4A, Section A.1.

$$B_{AD} = \$2000x\,P_D\,x\,A\,x0.025x\,F\left(\frac{1-e^{-(r+\lambda)N}}{r+\lambda}\right)$$

Where: B_{AD} is the benefit of averted dose

Variables are as described in NUREG-1727, Table D2 . The detailed description of the calculation of the B_{AD} is provided in Attachment 4A, Sections A.3 and A.4.

4.7 ALARA Calculation Results

The final ALARA calculations were performed by comparing the total remediation cost to the benefit of averted dose using Equation D8 from NUREG-1727. The calculations are described in detail in Attachment 4A. The results for each remediation method, for both the Basement Fill and Building Occupancy scenarios, are provided in Table 4-2. Since the Conc/DCGL$_W$ values are greater than 1 for all remediation methods, no remediation below the NRC 25 mrem/y dose limit is required. As described in Attachment 4A, the results are also valid for the enhanced State criteria since lowering the dose criteria increases the Conc/DCGL$_W$ value.

Table 4-2
ALARA Evaluation Conc/DCGL$_W$ Results

Remediation Action	Basement Fill	Building Occupancy
Pressure Washing and Vacuuming	99.4	1.9
Wiping/Washing	312.6	6.00
Concrete Scabbling(Upper Bound)	143.9	2.76
Concrete Scabbling (Lower Bound)	123.9	2.38
Grit Blasting Surfaces (Upper Bound)	153.3	2.94
Grit Blasting Surfaces (Lower Bound)	118.9	2.28
Grit Blasting Embedded/Buried Piping	91.6[a]	--
Soil Excavation	733.9[b]	--

[a]Grit blasting of embedded piping is not evaluated for Building Occupancy
[b]Soil is evaluated using the Surface Soil values from NUREG-1727 Table C2.3.

4.8 References

4.8.1 Maine Erosion and Sediment Control Handbook for Construction, Best Practices Manual

4.8.2 NUREG 1727, "Decommissioning Standard Review Plan"

4.8.3 NUREG/CR 5884, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station", Volume 2

ATTACHMENT 4A

Calculation of ALARA Residual Radioactivity Levels

This attachment provides the method for calculating residual radioactivity levels that are ALARA.

A.1 Residual Radioactivity Level ALARA Calculation

For the purposes of addressing multiple radionuclides, Equation D8 of NUREG-1727 as presented below is modified. The equation used for each spreadsheet is provided in Section A.1.1

(NUREG-1727, eq. D8).

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

Where:

$\frac{Conc}{DCGL_W}$ = Fraction of $DCGL_W$ that is ALARA

$Cost_T$ = Total monetary cost of remediation action in dollars

2000 = The dollar value of a person-rem averted ($/person-rem)

P_D = Population density for the critical group scenario (persons per m^2)

0.025* = Annual dose to an average member of the critical group from residual radioactivity at the $DCGL_W$ concentration (rem/yr)

*** NOTE:** This calculation is performed in compliance with 10 CFR 20, with regard to 25 mrem. If calculated using the 10 mrem annual dose limit an even wider divergence between cost and benefit would result.

F = Fraction of the residual radioactivity removed by remediation action.

A = Area (m^2) used to calculate the population density

r = Monetary discount rate (yr^{-1})

λ = Radiological decay constant for the radionuclide (yr^{-1})

N = Number of years over which collective averted dose is calculated (yr)

Values for the equation parameters may be found in NUREG-1727. The table below presents some of these generic values.

Table A-1
Equation Parameters

Equation Terms	NUREG-1727 Table D2 Values	
	Structure	Land
P_D	0.09	0.0004
r	0.07	0.03
N	70	1000

A.1.1 Equation D8 as used in Section 4.0 ALARA Evaluations

Equation D8, NUREG-1727 is presented below:

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

$$= \frac{Cost_T}{(\$2000)(P_D)(0.025)(F)(A)}\left[\frac{\frac{r+\lambda}{1-e^{-(r+\lambda)N}}}{1}\right]$$

The right term of the equation is multiplied by 1 as illustrated in the term below.

$$= \frac{Cost_T}{(\$2000)(P_D)(0.025)(F)(A)} \left[\frac{\frac{r+\lambda}{1-e^{-(r+\lambda)N}}}{1} \right] \left[\frac{\frac{1-e^{-(r+\lambda)N}}{r+\lambda}}{\frac{1-e^{-(r+\lambda)N}}{r+\lambda}} \right]$$

Equation D8, NUREG-1727 is then expressed as:

$$\frac{Conc}{DCGL_w} = \frac{Cost_T}{(\$2000)(P_D)(0.025)(F)(A)\left[\frac{1-e^{-(r+\lambda)N}}{r+\lambda}\right]}$$

For multiple radionuclides the denominator must be summed over all radionuclides as shown below:

$$\frac{Conc}{DCGL_w} = \frac{Cost_T}{\sum_i^n(\$2000)(P_D)(0.025)(Df_i)(F)(A)\left[\frac{1-e^{-(r+\lambda_i)N}}{r+\lambda_i}\right]}$$

Where for :

Basement Fill Scenario:

$$Df_i = Dose\,Fraction_{basement\,fill} = \frac{(nf_i)(Unitized\,Dose\,Factor_i)}{\sum_i^n(nf_i)(Unitized\,Dose\,Factor_i)}$$

or, Building Occupancy;

$$Df_i = Dose\,Fraction_{building\,occupancy} = \frac{\frac{nf_i}{Screening\,Value_i}}{\sum_i^n \frac{nf_i}{Screening\,Value_i}}$$

And,

nf_i = nuclide fraction of the mixture radionuclide

Unitized Dose Factor$_{i\ \text{(basement fill)}}$ = nuclide specific mrem/y per dpm/100 cm^2 (or pCi/g) results from the respective Unitized Dose Tables 6-2 through 6-5, and 6-7 through 6-8 of Section 6.0.

Screening Value$_{i\ \text{(building occupancy)}}$ = nuclide specific Screening Values from Table 5.19 of NUREG-5512V3 or NUREG-1727 Table C2.2.

A.2 Calculation of Total Cost

(NUREG-1727 eq. D3)

In order to evaluate the cost of remediation actions NUREG-1727 provides the elements necessary to derive the costs that are compared to the benefits. The total cost is:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + C_{TF} + C_{WDose} + C_{PDose} + C_{other}$$

The terms for "Cost" are abbreviated as "C" below (NUREG-1727 eq. D4-D7)

C_T = Total costs (all the elements below)

C_R = Monetary cost of the remediation action (may include mobilization costs).

C_{WD} = Cost for generation and disposal of the waste generated by the action:

C_{WD} = V_A x C_V

V_A Is the volume of waste produced, remediated in units of m^3 and;

C_V is the cost of waste disposal per unit volume, including transport cost, in units of \$/m^3

C_{ACC} = Cost of worker accidents during the remediation action:

C_{ACC} = \$3,000,000 x F_W x T_A

\$3,000,000 is cost of a fatality equivalent to \$2,000/person-rem;

FW is the workplace fatality rate in fatalities/hour worked (4.20E-8/h) and;

T_A is the worker time required for remediation in units of worker-hours.

C_{TF} = Cost of traffic fatalities during transport of the waste:

C_{TF} = \$3,000,000 x $_{VA}$ x $[(F_T \times D_T)/V_{ship}]$

F_T is the fatality rate per kilometer traveled in units of fatalities/km (3.80E-8), for truck shipments and 1.70E-9 for hazardous material shipped by rail (Class 1 rail = 9.8E-07). The hazardous material value is conservatively used in the calculations; however, in any case C_{TF} does not significantly impact the evaluation results.

D_T is the round trip distance from Maine Yankee to Clive, Utah (Envirocare), in km;

V_{SHIP} is volume of truck shipment in m^3(estimated at 7.93 m^3); for rail the respective volumes used for concrete and soil are 92 and 122 m^3.

C_{WDose} = \$2,000 x D_R x T:
C_{WDose} == is the cost of the remediation worker dose
\$2000 is the cost of dose received by workers performing the remediation and transporting the waste to the disposal facility.

D_R is total effective dose equivalent rate to remediation workers in units of rem/hr and,

T is time worked to remediate the area in units of person-hours

C_{PDose} = Cost of the dose to the public from excavation, transport, and disposal of the waste.

C_{other} = Other appropriate costs for the particular situation.

A.3 Calculation of Benefits

(NUREG-1727 eq. D1)

The benefit from collective averted dose is calculated by determining the present worth of the future collective averted dose and multiplying it by a factor to convert the dose to monetary value:

$$B_{AD} = (\$2000)[PW(AD_{COLLECTIVE})]$$

Where:

B_{AD} = benefit from averted dose for a remediation action, in $

$2,000 = value in dollars of a person-rem averted

$PW(AD_{COLLECTIVE})$ = present worth of future collective averted dose

A.4 Present Worth of Future Collective Averted Dose

(NUREG-1727 eq. D2)
The present worth of the future collective averted dose is estimated by:

$$PW(AD_{Collective}) = (P_D)(A)(0.025)(F)\left[\frac{Conc}{DCGL_W}\right]\left[\frac{1-e^{-(r+\lambda)N}}{r+\lambda}\right]$$

Where:

P_D = population density for the critical group scenario in people per m^2

A = Area being evaluated in m^2 and represents the floor area only for the attached ALARA calculations.

0.025* = Annual dose to an average member of the critical group from residual radioactivity at the $DCGL_W$ concentration in rem/y

*** NOTE:** This calculation is performed in compliance with 10 CFR 20, with regard to 25 mrem. If calculated using the 10 mrem annual dose limit an even wider divergence between cost and benefit would result.

F = Fraction of the residual radioactivity removed by the remediation action. F may be considered to be the removable fraction for the remediation action being evaluated.

Conc = Average concentration of residual radioactivity being evaluated in units of activity per unit area for buildings or activity per unit volume for soil.

$DCGL_W$ = derived concentration guideline level that represents a dose of 25 mrem/yr to the average member of the critical group, in the same units as "Conc"

r = monetary discount rate in units of yr^{-1}

λ = radiological decay constant for the radionuclide in units of yr^{-1}

N = number of years over which the collective dose will be calculated.

A.5 ALARA Evaluation Spreadsheets and Development

Evaluation spreadsheets incorporate the B_{AD} results for each nuclide in the mixture relative to the remediation action. The spreadsheets, if necessary, may be modified to address changes or additional regulatory guidance. The spreadsheets provide input for fraction of activity removed, total cost and remediation surface area. Other nuclide fractions can be input to address changes in mixtures and the dose factors attributing to the respective scenario can be replaced as necessary.

The spreadsheets utilize the formula provided in Section A.1.1 and are designed to sum the B_{AD} results for each radionuclide in the mixture. To correctly do so requires that the individual dose fraction be multiplied by the annual dose (0.025 rem/y) to an average member of the critical group. The total cost for the remedial action when divided by the benefit of averted dose results in the Conc/DCGL as per NUREG-1727, Equation D2. The results determine the cost effectiveness of the remedial action. Values greater than unity are already ALARA.

For scabbling and grit blasting a reduction factor of 0.95 is used. Because a majority of contamination is near the surface of the media the abrasive or scabbling actions are expected to be very efficient. Pressure washing and washing and wiping activities are designed primarily for removal of loose contaminants - grimes and adhered oils and greases. These remediation actions are intended to remove all the loose contamination and the layers of grease and oils adhered to surfaces. These actions are expected to remove a minimum of 10.0 percent of the contaminants. The characterization results in Section 2.0 show that the average loose contamination fraction is less than 10.0 percent. NUREG-1727 uses a reduction factor of 20.0 percent for washing a building. The use of decontamination agents with liquid is anticipated to increase the reduction factor for the pressure washing and washing and wiping. Conservative values of 20.0 percent for washing and wiping and 25.0 percent for pressure washing are used in the evaluations.

The Basement Fill and Building Occupancy dose models were evaluated for each applicable remediation method. For the basement fill model the occupancy area is 10,000 m^2 since the resident farmer is the critical group. The area remediated is the assumed model area of 4182 m^2. Note that reducing this area size would reduce dose proportionally. For the Building Occupancy model the occupancy area is a 100 m^2 floor in a standing building; the remediation area is also assumed to be 100 m^2.

A.5.1 ALARA Spreadsheet Evaluations:

Pressure Washing (Basement Fill Model)

A removal fraction for pressure washing utilizing standard commercial pressure washing techniques is about 0.25. This reduction fraction is associated with removal of loose contamination as well as greases and oils adhered to surfaces. The ALARA Evaluation results show that the Conc/$DCGL_W$ result is 99.4 and ALARA.

Pressure Washing (Building Occupancy Model)

The results indicate that for a removal fraction of 0.25 the action is ALARA without remediation actions. As previously stated, the use of a removal fraction of 0.25 assumes that the operation will, at a minimum, remove all loose contamination and adhering grease and oil from suspect surfaces (NUREG-5884, M.27). The ALARA Evaluation shows that the Conc/$DCGL_W$ result is 1.9 and ALARA.

Washing and Wiping (Basement Fill Model)

The removal fraction used for washing and wiping is 0.20 and shows residual radioactivity being ALARA without taking any remediation actions. The ALARA Evaluation shows the Conc/$DCGL_W$ result is 312.6.

Washing and Wiping (Building Occupancy Model)

The building occupancy model as stated is based on a 100 m^2 area. The removal fraction is 0.20. The ALARA Evaluation results shows the Conc/$DCGL_W$ result is 6.0. Residual radioactivity is ALARA without taking any remediation actions.

Scabbling (Basement Fill Model)

The Scabbling evaluation is performed using the maximum expected scabble depth and the manpower and equipment cost using a standard contingency of 1.25. The associated total cost when compared to the benefit of averted dose is determined to be ALARA without taking remediation actions. The second evaluation for scabbling evaluates the activity using one half of the maximum expected depth using the same manpower and equipment hours associated with the remediation rate. The cost for compressor and consumables at 10% of the equipment cost is not used (a cost reduction of ~14%). The results of the evaluation again show that the action is still ALARA without remediation actions.

Costs are based on assuming the entire surface area of the three foot below grade structure is scabbled (this area size assumption is used for all surface remediation activities). This is a conservative assumption since maximizing remediated area results in the lowest unit cost. The ALARA Evaluation shows the Conc/$DCGL_W$ results are 143.9 and 123.9, respectively.

Scabbling (Building Occupancy Model)

Scabbling conditions for bounding are the same as the basement fill model. The only changes are unit costs and evaluation area are 100 m^2. The results of the evaluation show the action is still ALARA without remediation actions. The ALARA Evaluation shows the Conc/$DCGL_W$ results are 2.76 and 2.38 respectively.

Embedded Piping Grit Blasting (Basement Fill Model)

Embedded and buried piping assumes a reduction fraction of 0.95. The total linear feet of piping is used (6,158 feet). The spreadsheet utilizes the same surface area as do other evaluations for the basement fill scenario. The cost basis is per linear foot. The ALARA Evaluation result for the Conc/$DCGL_W$ is 91.6 and already ALARA.

Surface Grit Blasting (Basement Fill Model)

Evaluation for surface grit blasting utilizes the same area and removal fractions as for scabbling. The results of the evaluation show the action is ALARA without remediation actions. The ALARA Evaluation shows the Conc/$DCGL_W$ results are 153.3 and 118.9 for the upper and lower bound cost contingency evaluations, respectively.

Surface Grit Blasting (Building Occupancy Model)

Evaluation for surface grit blasting utilizes the same area and removal fractions as for scabbling. The results of the evaluation again show the action is still ALARA without remediation actions. The ALARA Evaluation results shows the Conc/$DCGL_W$ results are 2.94 and 2.28 for the upper and lower bound cost contingency evaluations, respectively.

Soil Excavation

Due to high removal and shipping costs, excavation of significant quantities of soil from the site show that the residual radioactivity is ALARA without additional actions. The reduction fraction used is 0.95. The amount of soil expected to be removed is 1,403.1 m^3 or about 94 percent of what would be removed from an area 10,000 m^2 by 0.15 m deep. The ALARA Evaluation results show the Conc/$DCGL_W$ results is 733.9.

For all actions evaluated the conditions utilize 25 mrem per year as the dose to the critical group. If the annual dose criteria is changed to 10 mrem in the evaluation equation the margin for the action being ALARA without remediation actions is significantly greater. Tables A-2 through A-15 are the ALARA Evaluation Spreadsheets for each of the above evaluations.

A.5.2 Examination of Differential Solubility for Specific Decontamination Actions

To determine if differential solubility for specific nuclides could affect the reduction of specific radionuclides in the mixture, those nuclides expected to exhibit the most preferential solubility (H-3, Sr-90, Cs-134 and Cs-137) were examined. For this sensitivity analysis both washing and wiping, and pressure washing actions were used with the building occupancy scenario. These scenarios provided the lowest Conc/DCGL values. For the specific nuclides the removal rate was doubled. The analysis showed that, while the Conc/DCGL value was reduced by approximately 46 percent the conclusion is the same as that using the initially assigned values (Conc/DCGL is >1.0).

Table A-2

Basement Fill Scenario

Pressure Washing Remediation Activity
Condition (removal fraction "F"@ 0.25)
A =10k m^2, r = 0.03, N =1000, P_D = 0.0004
PWAD4prwfill.wb3)
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.25

Enter Occupancy Area in m^2 ==========> 10,000

Enter total cost (C_T, in dollars) of Action(s) =========> $80,796

Remediation Cost and Area

Unit Cost/M²	Actual Area M²
$19.32	4182.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs⁻¹)[b]	(r + λ)	(r + λ)N	e -(r + λ)N]	[1-e^-(r+λ)N]	[1-e-(r+λ)N]/(r+λ)	nuclide	Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	8.607E-02	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3	2.410E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	2.882E-01	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55	2.566E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	9.645E-01	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57	2.023E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	1.615E-01	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60	5.698E+01	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	3.692E-02	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63	2.915E+01	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	5.405E-02	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90	8.346E+01	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	3.662E-01	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134	1.097E+00	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	5.297E-02	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137	6.177E+02	5.50E-01	2.26E-05	1.24E-05	6.54E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						**Mixture Total: Benefit of Averted Dose B_{AD} ===>**			**$812.56**	1.00E+00			
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared						**Conc/DCGLw =============>**			**99.43**	Sum Check	Sum	1.90E-05	1.00E+00

Table A-3

Building Occupancy Scenario

Pressure Washing Remediation Activity
Condition (removal fraction "F"@ 0.25)
A=100 m^2, r =0.07, N=70, P_D = 0.09
PWAD4prwbo.wb3)
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action == 0.25

Enter Occupancy Area in m^2 ==========> 100

Enter total cost (C_T, in dollars) of Action(s) ========= $1,932

Remediation Cost and Area

Unit Cost/M²	Actual Area M
$19.32	100.0

Building Occupancy Scenario

nuclide	halflife[a] (yrs)	λ (yrs⁻¹)[b]	(r + λ)	(r + λ)N	e -(r + λ)N]	[1-e^-(r+λ)N]	[1-e-(r+λ)N]/(r+λ)	nuclide	Nuclide B_{AD}	Nuclide Fraction	Screening[c] Value (SC)	nf/SC	SC/sum[nf/SC]
H-3	1.236E+01	5.607E-02	**1.261E-01**	8.825E+00	1.470E-04	9.999E-01	**7.931E+00**	H-3	6.089E-03	2.36E-02	1.200E+08	1.96E-10	6.82E-06
Fe-55	2.685E+00	2.582E-01	**3.282E-01**	2.297E+01	1.056E-10	1.000E+00	**3.047E+00**	Fe-55	1.275E-02	4.81E-03	4.50E+06	1.07E-09	3.72E-05
Co-57	7.417E-01	9.345E-01	**1.005E+00**	7.032E+01	2.893E-31	1.000E+00	**9.955E-01**	Co-57	5.683E-03	3.06E-04	2.10E+05	1.46E-09	5.07E-05
Co-60	5.270E+00	1.315E-01	**2.015E-01**	1.411E+01	7.472E-07	1.000E+00	**4.962E+00**	Co-60	1.597E+02	5.84E-02	7.100E+03	8.23E-06	2.86E-01
Ni-63	1.001E+02	6.925E-03	**7.692E-02**	5.385E+00	4.586E-03	9.954E-01	**1.294E+01**	Ni-63	9.990E+00	3.55E-01	1.800E+06	1.97E-07	6.86E-03
Sr-90	2.882E+01	2.405E-02	**9.405E-02**	6.584E+00	1.383E-03	9.986E-01	**1.062E+01**	Sr-90	1.338E+01	2.80E-03	8.700E+03	3.22E-07	1.12E-02
Cs-134	2.062E+00	3.362E-01	**4.062E-01**	2.843E+01	4.494E-13	1.000E+00	**2.462E+00**	Cs-134	3.449E+00	4.55E-03	1.270E+04	3.58E-07	1.25E-02
Cs-137	3.017E+01	2.297E-02	**9.297E-02**	6.508E+00	1.491E-03	9.985E-01	**1.074E+01**	Cs-137	8.256E+02	5.50E-01	2.800E+04	1.97E-05	6.83E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						**Mixture Total: Benefit of Averted Dose B_{AD} :**			**$1,012.13**	1.00E+00			
c: From NUREG-1727 Table C2.2, dpm/100 centimeters squared						**Conc/DCGLw ============:**			**1.91**	Sum Check	Sum	2.88E-05	1.00E+00

Table A-4

Basement Fill Scenario

Washing and Wiping Remediation Activity
Condition (removal fraction "F"@ 0.25)
A =10k m^2, r = 0.03, N =1000, P_D = 0.0004
PWAD4wwfill.wb3)
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.2

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ===========> $203,203

Remediation Cost and Area	
Unit Cost/M^2	Actual Area M^2
$48.59	4182.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r+\lambda)$	$(r+\lambda)N$	e -(r + λ)N]	[1-e$^{-(r+\lambda)N}$]	[1-e-(r+λ)N]/(r+λ)	nuclide		Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		1.928E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	**2.882E-01**	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55		2.053E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	**9.645E-01**	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57		1.619E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		4.559E+01	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		2.332E+01	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	**5.405E-02**	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90		6.677E+01	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	**3.662E-01**	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134		8.775E-01	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		4.942E+02	5.50E-01	2.26E-05	1.24E-05	6.54E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						**Mixture Total: Benefit of Averted Dose B_{AD} ===>**				**$650.05**	1.00E+00			
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared						**Conc/DCGLw ==============>**				**312.60**	Sum Check		Sum 1.90E-05	1.00E+00

Table A-5

Building Occupancy Scenario

Washing and Wiping Remediation Activity
Condition (removal fraction "F"@ 0.25)
A=100 m^2, r =0.07, N=70, P_D = 0.09
PWAD4wwbo.wb3)
04/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.2

Enter Occupancy Area in m^2 ===========> 100

Enter total cost (C_T, in dollars) of Action(s) ========== $4,859

Remediation Cost and Area	
Unit Cost/M^2	Actual Area M
$48.59	100.0

Building Occupancy Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r+\lambda)$	$(r+\lambda)N$	e -(r + λ)N]	[1-e$^{-(r+\lambda)N}$]	[1-e-(r+λ)N]/(r+λ)	nuclide		Nuclide B_{AD}	Nuclide Fraction	Screening[c] Value (SC)	nf/SC	SC/sum[nf/SC]
H-3	1.236E+01	5.607E-02	**1.261E-01**	8.825E+00	1.470E-04	9.999E-01	**7.931E+00**	H-3		4.871E-03	2.36E-02	1.200E+08	1.96E-10	6.82E-06
Fe-55	2.685E+00	2.582E-01	**3.282E-01**	2.297E+01	1.056E-10	1.000E+00	**3.047E+00**	Fe-55		1.020E-02	4.81E-03	4.50E+06	1.07E-09	3.72E-05
Co-57	7.417E-01	9.345E-01	**1.005E+00**	7.032E+01	2.893E-31	1.000E+00	**9.955E-01**	Co-57		4.546E-03	3.06E-04	2.10E+05	1.46E-09	5.07E-05
Co-60	5.270E+00	1.315E-01	**2.015E-01**	1.411E+01	7.472E-07	1.000E+00	**4.962E+00**	Co-60		1.278E+02	5.84E-02	7.100E+03	8.23E-06	2.86E-01
Ni-63	1.001E+02	6.925E-03	**7.692E-02**	5.385E+00	4.586E-03	9.954E-01	**1.294E+01**	Ni-63		7.992E+00	3.55E-01	1.800E+06	1.97E-07	6.86E-03
Sr-90	2.882E+01	2.405E-02	**9.405E-02**	6.584E+00	1.383E-03	9.986E-01	**1.062E+01**	Sr-90		1.070E+01	2.80E-03	8.700E+03	3.22E-07	1.12E-02
Cs-134	2.062E+00	3.362E-01	**4.062E-01**	2.843E+01	4.494E-13	1.000E+00	**2.462E+00**	Cs-134		2.759E+00	4.55E-03	1.270E+04	3.58E-07	1.25E-02
Cs-137	3.017E+01	2.297E-02	**9.297E-02**	6.508E+00	1.491E-03	9.985E-01	**1.074E+01**	Cs-137		6.605E+02	5.50E-01	2.800E+04	1.97E-05	6.83E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						**Mixture Total: Benefit of Averted Dose B_{AD} =**				**$809.70**	1.00E+00			
c: From NUREG-1727 Table C2.2, dpm/100 centimeters squared						**Conc/DCGLw ==============**				**6.00**	Sum Check		Sum 2.88E-05	1.00E+00

Table A-6

Basement Fill Scenario

Scabbling Remediation Activity
Bounding Condition (remove 0.25 inches of concrete surface)
Using upper bound cost contingency
PWAD4scabfil.wb3)
A=10k m^2, r =0.03, N=1000, Pd = 0.0004
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ===========> $444,254

Remediation Cost and Area

Unit Cost/M^2	Actual Area M^2
$106.23	4182.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r + \lambda)$	$(r + \lambda)N$	$e^{-(r + \lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		9.158E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	**2.882E-01**	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55		9.750E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	**9.645E-01**	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57		7.689E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		2.165E+02	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		1.108E+02	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	**5.405E-02**	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90		3.171E+02	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	**3.662E-01**	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134		4.168E+00	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		2.347E+03	5.50E-01	2.26E-05	1.24E-05	6.54E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						Mixture Total: Benefit of Averted Dose B_{AD} ===>				**$3,087.72**	1.00E+00			
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared						Conc/DCGLw =============>				**143.88**	Sum Check	Sum	1.90E-05	1.00E+00

Table A-7

Basement Fill Scenario

Scabbling Remediation Activity
Bounding Condition (remove 0.125 inches of concrete surface)
Using lower bound cost (no contingency)
PWAD4scabfil.wb3)
A=10k m^2, r =0.03, N=1000, Pd = 0.0004
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ===========> $382,611

Remediation Cost and Area

Unit Cost/M^2	Actual Area M^2
$91.49	4182.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r + \lambda)$	$(r + \lambda)N$	$e^{-(r + \lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		9.158E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	**2.882E-01**	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55		9.750E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	**9.645E-01**	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57		7.689E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		2.165E+02	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		1.108E+02	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	**5.405E-02**	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90		3.171E+02	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	**3.662E-01**	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134		4.168E+00	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		2.347E+03	5.50E-01	2.26E-05	1.24E-05	6.54E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						Mixture Total: Benefit of Averted Dose B_{AD} ===>				**$3,087.72**	1.00E+00			
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared						Conc/DCGLw =============>				**123.91**	Sum Check	Sum	1.90E-05	1.00E+00

Table A-8

Building Occupancy Scenario

Scabbling Remediation Activity

ALARA EVALUATION

Bounding Condition (remove 0.25 inches of concrete surface)

A=100 m², r =0.07, N=70, P_D = 0.09

PWAD4scabo.wb3)

4/26/01

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m² ============> 100

Enter total cost (C_T, in dollars) of Action(s) ===========> $10,623

Remediation Cost and Area

Unit Cost/M²	Actual Area M²
$106.23	100.0

Building Occupancy Scenario

nuclide	halflife[a] (yrs)	λ (yrs⁻¹)[b]	(r + λ)	(r + λ)N	e -(r + λ)N]	[1-e^-(r+λ)N]	[1-e-(r+λ)N]/(r+λ)	nuclide		Nuclide B_{AD}	Nuclide Fraction	Screening[c] Value (SC)	nf/SC	SC/sum[nf/SC]
H-3	1.236E+01	5.607E-02	**1.261E-01**	8.825E+00	1.470E-04	9.999E-01	**7.931E+00**	H-3		2.314E-02	2.36E-02	1.200E+08	1.96E-10	6.82E-06
Fe-55	2.685E+00	2.582E-01	**3.282E-01**	2.297E+01	1.056E-10	1.000E+00	**3.047E+00**	Fe-55		4.846E-02	4.81E-03	4.50E+06	1.07E-09	3.72E-05
Co-57	7.417E-01	9.345E-01	**1.005E+00**	7.032E+01	2.893E-31	1.000E+00	**9.955E-01**	Co-57		2.159E-02	3.06E-04	2.10E+05	1.46E-09	5.07E-05
Co-60	5.270E+00	1.315E-01	**2.015E-01**	1.411E+01	7.472E-07	1.000E+00	**4.962E+00**	Co-60		6.069E+02	5.84E-02	7.100E+03	8.23E-06	2.86E-01
Ni-63	1.001E+02	6.925E-03	**7.692E-02**	5.385E+00	4.586E-03	9.954E-01	**1.294E+01**	Ni-63		3.796E+01	3.55E-01	1.800E+06	1.97E-07	6.86E-03
Sr-90	2.882E+01	2.405E-02	**9.405E-02**	6.584E+00	1.383E-03	9.986E-01	**1.062E+01**	Sr-90		5.084E+01	2.80E-03	8.700E+03	3.22E-07	1.12E-02
Cs-134	2.062E+00	3.362E-01	**4.062E-01**	2.843E+01	4.494E-13	1.000E+00	**2.462E+00**	Cs-134		1.311E+01	4.55E-03	1.270E+04	3.58E-07	1.25E-02
Cs-137	3.017E+01	2.297E-02	**9.297E-02**	6.508E+00	1.491E-03	9.985E-01	**1.074E+01**	Cs-137		3.137E+03	5.50E-01	2.800E+04	1.97E-05	6.83E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						Total: Benefit of Averted Dose B_{AD} =====>				**$3,846.09**	1.00E+00			
c: From NUREG-1727 Table C2.2, dpm/100 centimeters squared							Conc/DCGLw =============>			**2.76**	Sum Check	Sum	2.88E-05	1.00E+00

Table A-9

Building Occupancy Scenario

Scabbling Remediation Activity

ALARA EVALUATION

Bounding Condition (remove 0.125 inches of concrete surface)

A=100 m², r =0.07, N=70, P_D = 0.09

PWAD4scabo.wb3)

4/26/01

Enter fraction of activity removed by remedial action == 0.95

Enter Occupancy Area in m² ===========> 100

Enter total cost (C_T, in dollars) of Action(s) ========== $9,149

Remediation Cost and Area

Unit Cost/M²	Actual Area M²
$91.49	100.0

Building Occupancy Scenario

nuclide	halflife[a] (yrs)	λ (yrs⁻¹)[b]	(r + λ)	(r + λ)N	e -(r + λ)N]	[1-e^-(r+λ)N]	[1-e-(r+λ)N]/(r+λ)	nuclide		Nuclide B_{AD}	Nuclide Fraction	Screening[c] Value (SC)	nf/SC	SC/sum[nf/SC]
H-3	1.236E+01	5.607E-02	**1.261E-01**	8.825E+00	1.470E-04	9.999E-01	**7.931E+00**	H-3		2.314E-02	2.36E-02	1.200E+08	1.96E-10	6.82E-06
Fe-55	2.685E+00	2.582E-01	**3.282E-01**	2.297E+01	1.056E-10	1.000E+00	**3.047E+00**	Fe-55		4.846E-02	4.81E-03	4.50E+06	1.07E-09	3.72E-05
Co-57	7.417E-01	9.345E-01	**1.005E+00**	7.032E+01	2.893E-31	1.000E+00	**9.955E-01**	Co-57		2.159E-02	3.06E-04	2.10E+05	1.46E-09	5.07E-05
Co-60	5.270E+00	1.315E-01	**2.015E-01**	1.411E+01	7.472E-07	1.000E+00	**4.962E+00**	Co-60		6.069E+02	5.84E-02	7.100E+03	8.23E-06	2.86E-01
Ni-63	1.001E+02	6.925E-03	**7.692E-02**	5.385E+00	4.586E-03	9.954E-01	**1.294E+01**	Ni-63		3.796E+01	3.55E-01	1.800E+06	1.97E-07	6.86E-03
Sr-90	2.882E+01	2.405E-02	**9.405E-02**	6.584E+00	1.383E-03	9.986E-01	**1.062E+01**	Sr-90		5.084E+01	2.80E-03	8.700E+03	3.22E-07	1.12E-02
Cs-134	2.062E+00	3.362E-01	**4.062E-01**	2.843E+01	4.494E-13	1.000E+00	**2.462E+00**	Cs-134		1.311E+01	4.55E-03	1.270E+04	3.58E-07	1.25E-02
Cs-137	3.017E+01	2.297E-02	**9.297E-02**	6.508E+00	1.491E-03	9.985E-01	**1.074E+01**	Cs-137		3.137E+03	5.50E-01	2.800E+04	1.97E-05	6.83E-01
a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;						Mixture Total: Benefit of Averted Dose B_{AD} =				**$3,846.09**	1.00E+00			
c: From NUREG-1727 Table C2.2, dpm/100 centimeters squared							Conc/DCGLw =============>			**2.38**	Sum Check	Sum	2.88E-05	1.00E+00

Table A-10

Basement Fill Scenario

Surface Grit Blasting Remediation Activity
Using upper bound cost contingency
PWAD4surgritfil.wb3)
A=10k m^2, r =0.03, N=1000, Pd = 0.0004
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action == 0.95

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ========== $473,319

Remediation Cost and Area	
Unit Cost/M²	Actual Area M²
$113.18	4182.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r+\lambda)$	$(r+\lambda)N$	$e^{-(r+\lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		9.158E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	**2.882E-01**	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55		9.750E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	**9.645E-01**	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57		7.689E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		2.165E+02	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		1.108E+02	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	**5.405E-02**	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90		3.171E+02	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	**3.662E-01**	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134		4.168E+00	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		2.347E+03	5.50E-01	2.26E-05	1.24E-05	6.54E-01
						Mixture Total: Benefit of Averted Dose B_{AD}:				**$3,087.72**	1.00E+00			
						Conc/DCGLw ============				**153.29**	Sum Check	Sum	1.90E-05	1.00E+00

a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared

Table A-11

Basement Fill Scenario

Surface Grit Blasting Remediation Activity
Using lower bound cost contingency
PWAD4surgritfil.wb3)
A=10k m^2, r =0.03, N=1000, Pd = 0.0004
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action == 0.95

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ========== $367,180

Remediation Cost and Area	
Unit Cost/M²	Actual Area M²
$87.80	4182.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r+\lambda)$	$(r+\lambda)N$	$e^{-(r+\lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		9.158E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	**2.882E-01**	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55		9.750E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	**9.645E-01**	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57		7.689E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		2.165E+02	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		1.108E+02	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	**5.405E-02**	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90		3.171E+02	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	**3.662E-01**	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134		4.168E+00	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		2.347E+03	5.50E-01	2.26E-05	1.24E-05	6.54E-01
						Mixture Total: Benefit of Averted Dose B_{AD}:				**$3,087.72**	1.00E+00			
						Conc/DCGLw ============				**118.92**	Sum Check	Sum	1.90E-05	1.00E+00

a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared

Table A-12

Building Occupancy

Surface Grit Blasting Remediation Activity
Using upper bound cost contingency
PWAD4surgritbo.wb3)
A=100 m^2, r =0.07, N=70, Pd = 0.09
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m^2 ===========> 100

Enter total cost (C_T, in dollars) of Action(s) ===========> $11,318

Remediation Cost and Area

Unit Cost/M²	Actual Area M²
$113.18	100.0

Building Occupancy

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r + \lambda)$	$(r + \lambda)N$	$e^{-(r + \lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Screening[c] Value (SC)	nf/SC	SC/sum[nf/SC]
H-3	1.236E+01	5.607E-02	**1.261E-01**	8.825E+00	1.470E-04	9.999E-01	**7.931E+00**	H-3		2.314E-02	2.36E-02	1.200E+08	1.96E-10	6.82E-06
Fe-55	2.685E+00	2.582E-01	**3.282E-01**	2.297E+01	1.056E-10	1.000E+00	**3.047E+00**	Fe-55		4.846E-02	4.81E-03	4.50E+06	1.07E-09	3.72E-05
Co-57	7.417E-01	9.345E-01	**1.005E+00**	7.032E+01	2.893E-31	1.000E+00	**9.955E-01**	Co-57		2.159E-02	3.06E-04	2.10E+05	1.46E-09	5.07E-05
Co-60	5.270E+00	1.315E-01	**2.015E-01**	1.411E+01	7.472E-07	1.000E+00	**4.962E+00**	Co-60		6.069E+02	5.84E-02	7.100E+03	8.23E-06	2.86E-01
Ni-63	1.001E+02	6.925E-03	**7.692E-02**	5.385E+00	4.586E-03	9.954E-01	**1.294E+01**	Ni-63		3.796E+01	3.55E-01	1.800E+06	1.97E-07	6.86E-03
Sr-90	2.882E+01	2.405E-02	**9.405E-02**	6.584E+00	1.383E-03	9.986E-01	**1.062E+01**	Sr-90		5.084E+01	2.80E-03	8.700E+03	3.22E-07	1.12E-02
Cs-134	2.062E+00	3.362E-01	**4.062E-01**	2.843E+01	4.494E-13	1.000E+00	**2.462E+00**	Cs-134		1.311E+01	4.55E-03	1.270E+04	3.58E-07	1.25E-02
Cs-137	3.017E+01	2.297E-02	**9.297E-02**	6.508E+00	1.491E-03	9.985E-01	**1.074E+01**	Cs-137		3.137E+03	5.50E-01	2.800E+04	1.97E-05	6.83E-01
						Mixture Total: Benefit of Averted Dose B_{AD} ===>				**$3,846.09**	1.00E+00			
						Conc/DCGLw ==============>				**2.94**	Sum Check	Sum	2.88E-05	1.00E+00

a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;
c: From NUREG-1727 Table C2.2, dpm/100 centimeters squared

Table A-13

Building Occupancy

Surface Grit Blasting Remediation Activity
Using lower bound cost contingency
PWAD4surgritbo.wb3)
A=100 m^2, r =0.07, N=70, Pd = 0.09
4/26/01

ALARA EVALUATION

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m^2 ===========> 100

Enter total cost (C_T, in dollars) of Action(s) ===========> $8,780

Remediation Cost and Area

Unit Cost/M²	Actual Area M²
$87.80	100.0

Building Occupancy

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r + \lambda)$	$(r + \lambda)N$	$e^{-(r + \lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Screening[c] Value (SC)	nf/SC	SC/sum[nf/SC]
H-3	1.236E+01	5.607E-02	**1.261E-01**	8.825E+00	1.470E-04	9.999E-01	**7.931E+00**	H-3		2.314E-02	2.36E-02	1.200E+08	1.96E-10	6.82E-06
Fe-55	2.685E+00	2.582E-01	**3.282E-01**	2.297E+01	1.056E-10	1.000E+00	**3.047E+00**	Fe-55		4.846E-02	4.81E-03	4.50E+06	1.07E-09	3.72E-05
Co-57	7.417E-01	9.345E-01	**1.005E+00**	7.032E+01	2.893E-31	1.000E+00	**9.955E-01**	Co-57		2.159E-02	3.06E-04	2.10E+05	1.46E-09	5.07E-05
Co-60	5.270E+00	1.315E-01	**2.015E-01**	1.411E+01	7.472E-07	1.000E+00	**4.962E+00**	Co-60		6.069E+02	5.84E-02	7.100E+03	8.23E-06	2.86E-01
Ni-63	1.001E+02	6.925E-03	**7.692E-02**	5.385E+00	4.586E-03	9.954E-01	**1.294E+01**	Ni-63		3.796E+01	3.55E-01	1.800E+06	1.97E-07	6.86E-03
Sr-90	2.882E+01	2.405E-02	**9.405E-02**	6.584E+00	1.383E-03	9.986E-01	**1.062E+01**	Sr-90		5.084E+01	2.80E-03	8.700E+03	3.22E-07	1.12E-02
Cs-134	2.062E+00	3.362E-01	**4.062E-01**	2.843E+01	4.494E-13	1.000E+00	**2.462E+00**	Cs-134		1.311E+01	4.55E-03	1.270E+04	3.58E-07	1.25E-02
Cs-137	3.017E+01	2.297E-02	**9.297E-02**	6.508E+00	1.491E-03	9.985E-01	**1.074E+01**	Cs-137		3.137E+03	5.50E-01	2.800E+04	1.97E-05	6.83E-01
						Mixture Total: Benefit of Averted Dose B_{AD} ===>				**$3,846.09**	1.00E+00			
						Conc/DCGLw ==============>				**2.28**	Sum Check	Sum	2.88E-05	1.00E+00

a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;
c: From NUREG-1727 Table C2.2, dpm/100 centimeters squared

Table A-14

Basement Fill Scenario

Embedded Piping Remediation Activity — **ALARA EVALUATION**

PWAD4embfill.wb3)
A=10k m^2, r =0.03, N=1000, Pd = 0.0004
Unit cost are in Linear Feet
4/26/01

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ===========> $282,837

Remediation Cost and Area

Unit Cost/lf	Actual Area LF
$45.93	6158.0

Basement Fill Scenario

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r + \lambda)$	$(r + \lambda)N$	$e^{-(r + \lambda)N}$	$[1-e^{-(r+\lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide B_{AD}	Nuclide Fraction	Unitized Dose[c] Factor (UDF)	nf(UDF)	UDF/ Sum (UDF)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		9.158E+01	2.36E-02	3.35E-05	7.89E-07	4.15E-02
Fe-55	2.685E+00	2.582E-01	**2.882E-01**	2.882E+02	7.166E-126	1.000E+00	**3.470E+00**	Fe-55		9.750E-02	4.81E-03	5.84E-07	2.81E-09	1.48E-04
Co-57	7.417E-01	9.345E-01	**9.645E-01**	9.645E+02	0.000E+00	1.000E+00	**1.037E+00**	Co-57		7.689E-03	3.06E-04	2.42E-06	7.43E-10	3.90E-05
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		2.165E+02	5.84E-02	5.99E-05	3.50E-06	1.84E-01
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		1.108E+02	3.55E-01	1.15E-06	4.10E-07	2.15E-02
Sr-90	2.882E+01	2.405E-02	**5.405E-02**	5.405E+01	3.357E-24	1.000E+00	**1.850E+01**	Sr-90		3.171E+02	2.80E-03	6.12E-04	1.72E-06	9.02E-02
Cs-134	2.062E+00	3.362E-01	**3.662E-01**	3.662E+02	9.577E-160	1.000E+00	**2.731E+00**	Cs-134		4.168E+00	4.55E-03	3.36E-05	1.53E-07	8.03E-03
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		2.347E+03	5.50E-01	2.26E-05	1.24E-05	6.54E-01
Mixture Total: Benefit of Averted Dose B_{AD} ===>										**$3,087.72**	1.00E+00			
Conc/DCGLw ==============>										**91.60**	Sum Check	Sum	1.90E-05	1.00E+00

a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;
c: From Table 6-2,unitized annual dose rate for contaminated concrete per dpm/100 centimeters squared

Table A-15

Soil Remediation

Soil Excavation — **ALARA EVALUATION**

where: 1403.1 m^3 ~10,000 m^2 @ 0.15 m deep (94%).
And, 1403.1 m^3 is the estimated volume for site soil removal
A = 10K, P_D =.0004, r =.03, N = 1000
PWAD4soittl.wb3
4/26/01

Enter fraction of activity removed by remedial action ==> 0.95

Enter Occupancy Area in m^2 ===========> 10,000

Enter total cost (C_T, in dollars) of Action(s) ===========> $2,576,882

Remediation Cost and Area

Unit Cost/M^3	Actual Volume M^3
$1,836.58	1403.1

Surface Soil

nuclide	halflife[a] (yrs)	λ (yrs^{-1})[b]	$(r + \lambda)$	$(r + \lambda)N$	$e^{-(r + \lambda)N}$	$[1 - e^{-(r + \lambda)N}]$	$[1-e^{-(r+\lambda)N}]/(r+\lambda)$	nuclide		Nuclide PW(ADcollective)	Enter Mix Nuclide Fraction	Screening[c] Values (SC)	nf(SC)	SC/ Sum (SC)
H-3	1.236E+01	5.607E-02	**8.607E-02**	8.607E+01	4.167E-38	1.000E+00	**1.162E+01**	H-3		1.27E+01	5.30E-02	1.10E+02	4.82E-04	5.75E-03
Co-60	5.270E+00	1.315E-01	**1.615E-01**	1.615E+02	7.071E-71	1.000E+00	**6.191E+00**	Co-60		3.33E+01	9.00E-03	3.80E+00	2.37E-03	2.83E-02
Ni-63	1.001E+02	6.925E-03	**3.692E-02**	3.692E+01	9.202E-17	1.000E+00	**2.708E+01**	Ni-63		1.40E+00	4.80E-02	2.10E+03	2.29E-05	2.73E-04
Cs-137	3.017E+01	2.297E-02	**5.297E-02**	5.297E+01	9.878E-24	1.000E+00	**1.888E+01**	Cs-137		3.46E+03	8.90E-01	1.10E+01	8.09E-02	9.66E-01
Mixture Total: Benefit of Averted Dose B_{AD} ===>										**$3,511**	1.00E+00			
Conc/DCGLw ==============>										**733.91**	Check Sum	Sum	8.38E-02	1.00E+00

a: Table of the Isotopes, Seventh Edition, Lederer et al. 1978; b: Lambda = 0.69315/t½;
c: From NUREG-1727 Table C2.3 pCi/g

ATTACHMENT 4B

Unit Cost Values

B.1 General

This Attachment provides the unit cost values used to develop the total cost C_T as defined in this section.

3 Feet Below Grade Remaining Structure Surfaces

The results of Engineering Calculation 01-00 (MY) show that the total structure and buildings surface area planned to remain at 3 feet below grade is 7704 m^2. This value is the surface area assumed to require remediation and is the area used to estimate remediation cost. This is a conservative approach because increasing the remediated area decreases the cost. For building occupancy 100 m^2 is used for determining both the cost and remediation action surface area.

Remediation Activity Rates

Remediation activity rates were provided based on previous experience, from published literature, or from groups or vendors currently performing these or similar activities. Past operational experience was also used in developing the rates.

Contingency

A contingency of 1.25 was added to the manpower hours. Scabbling (the primary activity) was bounded using cost and manpower associated with the volume of concrete (disposal cost) for remediation of 0.125 inches versus using compressor, consumable materials and the volume of concrete (disposal cost) for remediation of 0.25 inches of concrete.

Equipment

Equipment costs were developed based on the cost of buying specific equipment and whenever possible prorating the cost over the task activities. Rental rates are also included for specific equipment such as fork lifts and excavators. Consumable supplies and parts were included in the cost for equipment. Shipping containers were included with shipment costs.

Mobilization and Demobilization Costs

Costs were conservatively included for delivery and pick up of equipment. Anticipated costs to stage and move equipment from location to location were also included.

Waste Disposal Cost

Disposal costs for generated waste were based on the following rail shipment values:

Concrete Rubble:	\$10.00 (disposal) + \$6.25 (shipping) per cubic foot (\$573.87/m^3)
Concrete Scabble:	\$55.00 (disposal) + \$6.25 (shipping) per cubic foot (\$2163.04/m^3)
Soil:	\$41.00 (disposal) + \$6.56 (shipping) per cubic foot (\$1,679.58/m^3)

Round trip rail transportation:

Clive, Utah (Envirocare site) round trip by rail: 7728 km.

Waste volume per shipment:

Dependent primarily on highway hauling weight restrictions and results in the use of a volume of 7.93 m^3. For rail shipments the same conditions apply and result in a single car volume of 92 m^3 for concrete and 120 m^3 for soil. More than one car can be included in a rail shipment; however, costs estimates were based on a single car. The distance and haul volume are used for determining transport accident cost in accordance with NUREG-1727 and Attachment A, Section A2. The impact to total cost of this item is minimal.

Worker Accident Costs

To determine worker accident cost in accordance with NUREG-1727 and Attachment A, Section A2, the same hours input for labor cost were used for worker accident cost.

Worker Dose

Costs associated with worker dose are a function of the hours worked and the workers' radiation exposure for the task. General dose rates for each area from the initial facility walk down summary sheets were used to estimate worker doses. The results were summed and the average (7.3 mrem/h) used for all remediation activities. For soil excavation a value of 4.0 mrem/h was used.

The value of 7.3 mrem/hr for worker dose was based on data averaging. It is anticipated that, as commodities are removed and the area(s) prepared for final remediation actions, the dose to the worker will become less. Soil excavation assumes that stored waste remains near the excavation area. (This assumption is dependent upon which activities are conducted or completed prior to soil removal.) In the event that soil remediation follows all other activities and that waste stored for off-site shipment is removed, the dose to workers can be less than the above value.

To examine the impact of a lower worker dose, a sensitivity analysis was performed. By eliminating the cost factor associated with worker dose, the ALARA evaluation for the most sensitive (lowest) Conc/DCGL (that is, pressure washing using building occupancy scenario) results in a change in the Conc/DCGL from 1.91 to 1.76. In that the resulting Con/DCGL is still greater than 1.0, lower actual worker doses will not change the outcome of the ALARA assessment.

Labor Costs

Manpower costs assumptions were based on contracts established with the principal site contractors. The individual cost for the applicable disciplines, e.g., laborer, equipment operator, health physics technicians, were developed into an hourly crew rate for the task and based on guidance provided by NUREG 5884 Volumes 1 and 2. It is important to note that the total work hours for a normal day were used and not adjusted for personnel breaks, ALARA meeting or ingress and egress from an area.

Unit Cost

The sum of all the cost elements was divided by the applicable unit (m^2, m^3or linear feet) to provide a unit cost for the activity. Other cost units for cost per hour or linear foot were also developed in the same fashion. The tables to follow provide the crew cost per hour but do not provide the individual hourly rates for individual disciplines. These values are however included in the supporting calculation.

B.2 Pressure Water Washing And Vacuuming

Area Evaluated For Unit Cost Determination:	7704.0 m^2	
Primary Crew Size:	3.0,	Operating Engineer, 1; and Laborer, 2
Support Personnel:	3.0,	Resident, Schedule Engineers, HP Technician
Hourly Cost:	$ 99.19	
Cleaning Rate:	9.29 $m^{2/}$/h	
Hours:	829.3 (7704 m^2/9.29 $m^{2/}$/h)	

Mobilization Costs	$600
Labor Cost:	$82,256
Equipment Costs:	$8,000
Liquid Processing Costs:	$12,952 [($1.00/g)(1.35g/m^2)(7704 m^2) (1.25 liquid contingency)]
Waste Disposal Cost:	$ 33,328 Solids estimated at 0.002 m^3/m^2 = 15.4 m^3($ 2163.04)
Worker Accident Cost:	$105 Per NUREG-1727
Transportation Accident Cost:	$7 Per NUREG-1727
Worker Dose:	$11,610 Per NUREG-1727
Total Costs:	$148,858
Cost per m^2:	$19.32

B.3 Washing and Wiping Remediation Actions

Area Evaluated For Unit Cost Determination:	7704.0 m^2
Primary Crew Size: Support Personnel:	2.0, Laborers 5.0, Superintendent, Resident and Schedule Engineers, Operating Engineer and HP Technician
Hourly Cost:	$75.12
Cleaning Rate:	2.8 m^2/h
Hours:	3783.2 [(7704 m^2/2.8 m^2/h) + 4h/40h set up)(1.25 contingency)]
Mobilization Costs	$600
Labor Cost:	$284,195

Equipment Costs:	\$21,571
Waste Generation:	25.4 m^3 (3.39E-03 m^3/m^2)
Waste Disposal Cost:	\$14,550 (\$573.87/m^3)
Worker Accident Cost:	\$477 Per NUREG-1727
Transportation Accident Cost:	\$10 Per NUREG-1727
Worker Dose:	\$52,965 Per NUREG-1727
Total Costs:	\$374,368
Cost per m^2:	\$48.59

B.4 Scabbling Remediation Action (Bounding Condition 0.635 cm Concrete)*

Area Evaluated For Unit Cost Determination:	7704 m^2
Primary Crew Size:	2.0, Operating Engineer, Laborer
Support Personnel:	4.0, Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	\$82.12
Cleaning Rate:	1.86 $m^{2/}$/h
Hours:	4146.4 (7704 m^2/1.858 $m^{2/}$/h)
Mobilization Costs	\$7100
Labor Cost:	\$340,502
Equipment Costs:	\$303,682 (\$73.24/hr)*
Waste Generation:	48.9 m^3 = (7704 m^2)(6.35E-3 m)
Waste Disposal Cost:	\$105,817 (\$2,163.04/m^3)
Worker Accident Cost:	\$522 Per NUREG-1727

Transportation Accident Cost:	\$21 Per NUREG-1727
Worker Dose:	\$60,753 Per NUREG-1727
Total Costs:	\$818,397
Cost per m^2:	\$106.23*

*Bounding condition includes cost for air compressor, consumables at 10% of the base equipment costs and the waste volume of 0.25 inch (0.635 cm) concrete depth.

B.4.a Scabbling Remediation Action (Bounding Condition 0.32 cm Concrete)*

Area Evaluated For Unit Cost Determination:	7704 m^2
Primary Crew Size:	2.0, Operating Engineer, Laborer
Support Personnel:	4.0, Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	\$82.12
Cleaning Rate:	1.86 $m^{2/}$/h
Hours:	4,146.4 [(7704 m^2/1.858 $m^{2/}$/h)
Mobilization Costs	\$7100
Labor Cost:	\$340,502
Equipment Cost:	\$243,062 (\$58.62/hr)
Waste Generation:	24.5 m^3 = (7704 m^2)(3.18E-3 m)
Waste Disposal Cost:	\$52,908 (\$2163.04/m^3)
Worker Accident Cost:	\$522 Per NUREG-1727
Transportation Accident Cost:	\$10 Per NUREG-1727
Worker Dose:	\$60,753 Per NUREG-1727
Total Costs:	\$704,858

Cost per m^2: $91.49

*Bounding condition uses: (1) base equipment cost , (2) assumes an on-site air compressor, (3) no added consumables, and (4) the waste volume is relative to 0.125 inches (0.35 cm) depth of concrete, i.e., one-half of that assumed in B.4.

B.5 Grit Blasting (Embedded/Buried Piping) Remediation Action

Area Evaluated For Unit Cost Determination:	6,158 linear feet (LF)
Primary Crew Size:	3.0, Operating Engineer, 1; Laborers, 2
Support Personnel:	4.0, Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	$117.12
Cleaning Rate:	1 LF/minute
Hours:	1026.3 [(49,344 linear ft/60min per hr = (821 h)(1.25)]
Mobilization Costs	$4,000
Labor Cost:	$120,204
Equipment Costs:	$123,311
Waste Generation:	9.6 m^3 = (49,344 linear feet x1.96E-04 m^3/lf at ~ 1.0 lb. per linear foot)
Waste Disposal Cost:	$20,850 ($ 2163.04/m^3)
Worker Accident Cost:	$129 Per NUREG-1727
Transportation Accident Cost:	$4 Per NUREG-1727
Worker Dose:	$14,369 Per NUREG-1727

Total Costs:	\$282,867
Cost per linear foot:	\$45.93

B.6 Grit Blasting (Surfaces) Remediation Action (Bounding Condition 1.25 Contingency)

Area Evaluated For Unit Cost Determination:	7,704 m^2
Primary Crew Size:	3.0, Operating Engineer, 1; Laborers, 2
Support Personnel:	4.0, Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	\$122.12
Cleaning Rate:	2.79 m^2/hr
Hours:	3796.8 {[(7704/2.8 m^2/h) + ((7704/2.8 m^2/h)*(0.1 set up)]}* 1.25 contingency
Mobilization Costs	\$6,500
Labor Cost:	\$463,662
Equipment Costs:	\$196,977
Grit/Consumables	\$69,032
Waste Generation:	36.8 m^3 = (7704 x 3.0E-03 m + 13.7m^2 for grit)
Waste Disposal Cost:	\$79,626 (\$2163.04/m^3)
Worker Accident Cost:	\$478 Per NUREG-1727
Transportation Accident Cost:	\$16 Per NUREG-1727
Worker Dose:	\$55,630 Per NUREG-1727

Total Costs:	\$871,921
Cost per m^2	\$113.18

B.6a Grit Blasting (Surfaces) Remediation Action (Bounding Condition, No Contingency)

Area Evaluated For Unit Cost Determination:	7,704 m^2	
Primary Crew Size:	3.0,	Operating Engineer, 1; Laborers, 2
Support Personnel:	4.0,	Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	\$122.12	
Cleaning Rate:	2.79 m^2/hr	
Hours:	2761.3 (7704/2.79 m^2)	
Mobilization Costs	\$6,500	
Labor Cost:	\$337,209	
Equipment Costs:	\$143,256	
Grit/Consumables	\$69,032	
Waste Generation:	36.8 m^3 = (7704 x 3.0E-03 m + 13.7m^2 for grit)	
Waste Disposal Cost:	\$79,626 (\$ 2163.04/m^3)	
Worker Accident Cost:	\$348 Per NUREG-1727	
Transportation Accident Cost:	\$16 Per NUREG-1727	
Worker Dose:	\$40,458 Per NUREG-1727	
Total Costs:	\$676,445	
Cost per m^2:	\$87.80	

B.7 Soil Excavation Remediation Action

Area Evaluated For Unit Cost Determination:	1403.1 m^3 (49,550 ft^3)
Primary Crew Size:	4.0, Operating Engineers, 2; Laborers, 2
Support Personnel:	4.0, Superintendent, Resident and Schedule Engineers, and HP
Hourly Cost:	$157.12
Cleaning Rate:	3.06 m^3/h
Hours:	917.1 [(1403.1 m^3/3.06m^3/h)(2.0 contingency for restaging and articulation)]
Mobilization Costs	$700
Labor Cost:	$144,172
Equipment Costs:	$71,228 (consumables $9,291)
Waste Generation:	1403.1 m^3 (49,550 ft^3/35.315 ft^3/m^3)
Waste Disposal Cost:	$2,356,596 ($1,679.58/m^3)
Worker Accident Cost:	$58 Per NUREG-1727
Transportation Accident Cost:	$453 Per NUREG-1727
Worker Dose:	$3,670 Per NUREG-1727
Total Costs:	$2,576,878
Cost per m^3:	$1,836.58

Note: Remediation of an area of 10^4 m^2 to a depth of .15 m results in a total soil volume of 1500 m^3. The above remediation activity represents 94 percent of that volume

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LTP SECTION 5

FINAL STATUS SURVEY PLAN

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5.0 FINAL STATUS SURVEY PLAN

5.1 Introduction

5.1.1 Purpose

The Final Status Survey (FSS) Plan describes the final survey process used to demonstrate that the MY facility and site comply with radiological criteria for unrestricted use (NRC's annual dose limit of 25 mrem plus ALARA and the enhanced state clean-up levels of 10 mrem/year or less for all pathways and 4 mrem/year or less for groundwater drinking sources).

5.1.2 Overview

The final status survey includes remaining structures, land, and plant systems that are identified as contaminated or potentially contaminated as a result of licensed activities. The majority of the survey effort will be required in the basements of the Containment Building, Fuel Building, Primary Auxiliary Building, Spray Building and the surrounding yard areas. A final status survey of the Independent Spent Fuel Storage Installation (ISFSI) location (land area) was initiated prior to construction of the concrete base.

There are 5 major steps in the final survey process: survey preparation, survey design, data collection, data assessment, and documentation of survey results.

a. Survey Preparation

Survey preparation is the first step in the final survey process and occurs after remediation, if necessary, is completed. In areas where remediation was required, a turnover survey may be performed to confirm that remediation was successful prior to initiating final survey activities. A turnover survey may be performed using the same process and controls as a final survey so that data from a turnover survey may be used as part of the final survey data. In order for turnover survey data to be used for final status survey, it must have been designed and collected in compliance with LTP Sections 5.4 through 5.7 and the area controlled in accordance with Section 5.11. Following the turnover surveys, the final status survey is performed.

The area to be surveyed is isolated and/or controlled to ensure that radioactive material is not reintroduced into the area from ongoing demolition or remediation activities nearby and to maintain the final configuration of the area. Tools, equipment, and materials not needed to support survey activities are removed, unless authorized by the FSS Superintendent. Routine access, material storage, and worker transit through the area are not allowed, unless authorized by the FSS Superintendent. However, survey areas may, with proper approval, be used for staging of materials and equipment providing; 1) the staging does not interfere with performance of surveys, and 2) the external surfaces of the material or equipment are free of loose surface contamination and there is no likelihood that internal or fixed radioactive materials could escape and contaminate the surrounding area or create background concerns, and 3) the safety of survey personnel is not jeopardized.

An inspection of the area is conducted by FSS personnel to ensure that work is complete and the area is ready for final status survey. Control of activities is transferred from the Maine Yankee engineering/construction group to the FSS/RP organizations. Approved procedures provide isolation and control measures until the area is released for unrestricted use.

b. Survey Design

The survey design process establishes the methods and performance criteria used to conduct the survey. Survey design assumptions are documented in "Survey Packages" in accordance with approved procedures. The site land, structures, and systems (embedded and buried piping/conduit are the principal potentially contaminated systems that will remain after decommissioning) are organized into survey areas and classified by contamination potential as Class 1, Class 2, Class 3, or non-impacted in accordance with LTP Section 5.2 and Tables 5-1A, 5-1B, 5-1C, 5-1D, and 5-1E.

Survey unit size is based on the assumptions in the dose assessment models in accordance with the guidance provided in NUREG-1727. The percent coverage for scan surveys is determined in accordance with LTP Section 5.4.1 and Table 5-3. The number and location of structure surface measurements (and structure volumetric samples) and soil samples are established in accordance with LTP Sections 5.4.2 through 5.4.4. Investigation levels are also established in accordance with Section 5.6 and Table 5-7.

Replicate measurements are performed as part of the quality process established to identify, assess, and control errors and uncertainty associated with sampling, survey, or analytical activities. This quality control process, described in LTP Section 5.10, provides assurance that the survey data meets the accuracy and reliability requirements necessary to support the decision to release or not release a survey unit.

c. Survey Data Collection

After preparation of a survey package, the final survey data are collected. Trained and qualified personnel perform the necessary measurements using calibrated instruments in accordance with approved procedures and instructions contained in the survey package.

d. Survey Data Assessment

Survey data assessment is performed to verify that the data are sufficient to demonstrate that the survey unit meets the unrestricted use criterion (i.e., the Null Hypothesis may be rejected.). Statistical analyses are performed on the data and the data are compared to investigation levels. Depending on the results of an investigation, the survey unit may require further remediation, reclassification, and/or resurvey. Graphical representations of the data, such as posting plots or histograms, may be generated to provide qualitative information from the survey and to verify the assumptions in the statistical tests, such as spatial independence, symmetry, data variance and statistical power. The assumptions and requirements in the survey package are reviewed. Additional data needs, if required, are identified during this review.

e. Survey Results

Survey results are documented by Survey Area in "Survey Packages." Each final survey package may contain the data from the several Survey Units that are contained in a given Survey Area. The data is reviewed, analyzed, and processed and the results documented in a "Release Record." The Release Record provides the information necessary to support the decision to release the survey units for unrestricted use. A Final Survey Report is prepared that provides the necessary data and analyses from the Survey Packages and Release Records, and is submitted to the NRC.

5.1.3 Implementation

In its submittal to the NRC (MN99-26, dated 8/9/99), MY described the schedule for the phased release of site land. Two large site areas have been determined to be non-impacted (as described in Section 2 of the LTP). Details of the partial release application package are discussed in Section 1.4.2.c. The NRC granted the license amendment allowing the removal of the subject site land from the operating license by letter dated July 30, 2002. The impacted site areas are subject to a final status survey in accordance with this plan.

The final survey will be implemented in phases. The first phase was comprised of the survey of the ISFSI land and a portion of the ISFSI security operations building prior to construction of the ISFSI. The second phase includes: (a) the non-Radiological Restricted Area (RA) lands and any non-RA buildings which will remain standing within the Industrial Area; and (b) the survey of the RA land including the structural concrete which will remain three feet below grade. The third and final phase includes the ISFSI site following fuel removal, facility dismantlement and any required remediation. Survey results will be described in written reports to the NRC. The actual structures and land included in each written report may vary depending on the status of ongoing decommissioning activities.

Maine Yankee anticipates that both the NRC and the State of Maine Department of Human Services (DHS) - Division of Health Engineering (DHE) may choose to conduct confirmatory measurements in accordance with applicable laws and regulations. The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the final radiation survey and associated documentation demonstrate that the facility and site are suitable for release in accordance with the criteria for decommissioning established in 10 CFR Part 20, subpart E. Maine state law requires Maine Yankee to permit monitoring by the Maine State Nuclear Safety Inspectors (22 MRSA 664, sub-§2, as amended by PL 1999, c. 739, §1 and 38 MRSA 1451, sub-§11, as amended by PL 1999, c. 741, §1). This monitoring includes, among other things, taking radiological measurements to verify compliance with applicable state laws (including the enhanced state radiological criteria). Maine Yankee will demonstrate compliance with the 25 mrem/yr criteria of 10 CFR Part 20, Subpart E by demonstrating compliance with the enhanced state radiological criteria. Therefore, the confirmatory measurements taken by the NRC and the State of Maine will be based upon the same criteria, that is, the Derived Concentration Guideline Level (DCGL). Timely and frequent communications with these agencies will ensure that they are afforded sufficient opportunity to perform these confirmatory measurements prior to Maine Yankee implementing any irreversible decommissioning actions (e.g., backfilling basements with fill material.)

5.1.4 Regulatory Requirements and Industry Guidance

This plan has been developed using the guidance contained in the following documents:

a. Appendix E, NUREG 1727, "Demonstrating Compliance With the Radiological Criteria for License Termination" (September 2000).

b. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)", Revision 1 (June 2001).

c. NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," Revision 1 (June 1998 draft).

d. NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions" (June 1998).

e. Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (January 1999).

f. NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" (April 2000).

g. NUREG-1727, "NMSS Decommissioning Standard Review Plan" (September 2000)

Other documents used in the preparation of this plan are listed in the References Section.

5.2 Classification of Areas

Prior to beginning the final status survey, a thorough characterization of the radiological status and history of the site was completed. The methods and results from site characterization are described in Section 2 of the License Termination Plan. Based on the characterization results, the structures and open land areas were classified following the guidance in Appendix E of NUREG-1727 and Section 4.4 of NUREG 1575. There will be no above grade systems remaining following decommissioning. Contaminated systems will be disposed of as radioactive waste and non-radioactive systems will be disposed of as scrap. Area classification ensures that the number of measurements, and the scan coverage, are commensurate with the potential for residual contamination to exceed the unrestricted use criteria.

Initial classification of site areas is based on historical information and site characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance or any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the final status survey as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Once the FSS of a given survey unit begins, the basis for any reclassification will be documented, requiring a redesign of the survey unit package and the initiation of a new survey using the redesigned survey unit package. If during the conduct of a FSS survey sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit, the survey may be terminated without completing the survey unit package.

5.2.1 Non-Impacted Areas

Non-Impacted areas have no reasonable potential for residual contamination because there was no known impact from site operations. These areas are not required to be surveyed beyond what has already been completed as a part of site characterization to confirm the area's non-impacted classification. Maine Yankee will continue to implement its Radiological Environmental Monitoring Program (REMP) throughout the decommissioning phase of Maine Yankee. The REMP program is focused upon the collection of radiological data from offsite, non-impacted areas. Non-impacted areas are shown on Figure 5-1.

5.2.2 Impacted Areas

Impacted areas may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. The definitions provided below are from NUREG-1727, Pages E1 and E2.

a. Class 1 areas are impacted areas that, prior to remediation, are expected to contain residual contamination in excess of the $DCGL_w$.[1]

b. Class 2 areas are impacted areas that, prior to remediation, are not likely to contain residual radioactivity in excess of the DCGLw.

c. Class 3 areas are impacted areas that have a low probability of containing residual radioactivity.

5.2.3 Initial Classification of Basements, Land, Embedded Piping, and Buried Piping

Based on more than 19,000 measurements made during the site characterization and the information evaluated as part of the Historical Site Assessment, all land areas, basements, structures, and piping to remain after decommissioning were assigned an initial classification. The scope of the final status survey includes land and structures south of the Old Ferry Road. The areas to the north and west have been shown to meet the non-impacted criteria (LTP Section 2, Appendix A). The scope and boundaries of the FSS will be increased if survey data show significant levels of radioactivity above background in peripheral areas. (Initial Class 1 areas south of Ferry Road are shown on Figure 5-2. Additional Class 1 areas may be added as a result of ongoing characterization, remediation or survey activities.)

The primary interfaces between the impacted and non-impacted areas are the public road (Old Ferry Rd.) and the railroad spur. Both sides of the public road will be surveyed for FSS. If residual radioactivity greater than 0.5 DCGL is detected on the road or sides of the road, an investigation will be conducted to determine the extent of contamination and to identify any possible migration into the non-impacted areas. The portion of the railroad spur within the impacted area will be included in the final survey. If residual radioactivity greater than 0.5 DCGL is detected on the last 100 meters prior to exit from the impacted area, an investigation similar to that described above will be conducted.

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[1] The "w" in $DCGL_w$ refers to the Wilcoxon Rank Sum test per MARSSIM (NUREG-1575, page 2-3) but generally represents the uniform level of residual contamination that results in the dose limit, regardless of the statistical test used. See also, LTP Section 5.4.2.

Characterization was performed and reported by survey area. The area designations used for characterization were used, for the most part, to delineate and classify areas for final survey. This allowed the characterization data to be efficiently used for final survey area classification and for estimating the sigma value for sample size determination.

Tables 5-1A through 5-1E list the survey areas for basements, structure foundation footprints, land areas possibly augmented by structure footprints, embedded piping, and buried piping. See Attachment 5A for additional detail on embedded and buried piping and related discussions on the basis for the initial MARSSIM classification of the survey units. The major land areas are designated in Figure 5-3. For operational efficiency, each of the final survey areas listed in the tables may be subdivided into multiple areas. Smaller survey areas may be necessary to enhance the efficiency of data collection, processing, and review and serve to better support the decommissioning schedule. The classification of all subdivided survey areas will be the same as indicated in Tables 5-1A through 5-1E, unless reclassified in accordance with this LTP. The sigma values are based on site characterization data. See LTP Section 5.4.2 for the use of these sigma values in sample size determination.

Some survey areas have been assigned more than one classification based on the levels of activity found. During the FSS design process, when these areas are divided into survey units, administrative controls will ensure that each survey unit will have only one classification.

Survey areas for structures that are demolished will either be applied to the remaining footprint (if the foundation is removed) or the building basement. The soil below removed foundations in the RA and Industrial areas will undergo final survey prior to backfill. The need to survey soil in excavated footprints before backfill will be evaluated on a case by case basis and documented in the Final Survey Package. The soil in the excavated footprints of several structures may be combined into a single survey area and/or survey unit if final survey is required prior to backfill. Each survey unit will be comprised of one or more structural foundation footprints, will meet the size constraints for the associated structure or structures (per Table 5-2) and will possess generally uniform characteristics, including:

- Survey unit classification
- Material type and nuclide fraction
- Sigma
- Historical radiological impact of the area

The excavated foundation areas for any building or structure outside of the IA may not be surveyed prior to backfill.

A conservative approach of classifying the excavated foundation footprints will be to classify the footprints as one class lower than would have been assigned to the foundation concrete surface. For example, if contamination below the DCGL were identified on a given foundation surface that would have resulted in the concrete surface being Class 2, the soil remaining after the foundation is removed would be given a Class 3 designation. The intent of classifying the building footprints as one classification lower (than that for the foundation concrete surface) is based on the assumption that there was no evidence of external contamination and that the only potential for soil contamination would be building demolition. If there were any evidence of soil contamination or sub-slab contamination, such information would form the basis for the footprint classification. Absent such information, the footprint would be classified at one classification below the footprint structure. Following the satisfactory performance of FSS on the excavated foundation footprint surface, if required, the excavation area would be backfilled.

The major land areas are designated in Figure 5-3.

Table 5-1A
Survey Area Classification - Building Basements

Package Number	Survey Area-Structures	Interior		Exterior		Mean Direct Beta dpm/100cm^2	Maximum Direct Beta dpm/100cm^2	Approx. Survey Area Size (Meters^2)
		Sigma [g] (dpm/100 cm^2)	Class	Sigma (dpm/100 cm^2)	Class			
A0100	Containment-El.-2ft	6,853	1	N/A	N/A	81,976	1,970,974	4800
A0400	Fuel Bldg.	3,606	1	N/A	N/A	6,815	312,939	300
A0600	PAB-El.11ft	3,811	2,1	N/A	N/A	1,106	32,328	2200
A1700	Containment Spray Bldg.	6,132	2,1	N/A	N/A	83,249	4,968,088	1700

Table 5-1B
Survey Area Classification-Structural Foundation Footprints

Package Number	Survey Area-Structures	Interior		Exterior		Mean Direct Beta dpm/100cm^2	Maximum Direct Beta dpm/100cm^2	Approx. Survey Area Size (Meters^2)
		Sigma [g] (dpm/100 cm^2)	Class	Sigma (dpm/100 cm^2)	Class			
A0500[a]	DWST (Tk-21)	760	2	N/A	N/A	438	2,659	114
A0900[a]	Service Bld. Hot Side	1,456	2,1	N/A	N/A	699	18,955	885
A1100	LLWSB	3,149	3,1	86	3	852	74,216	980
A1200[a]	RCA Bldg.	4,880	2,1	N/A	N/A	73,939	2,233,580	290
A1300[a]	Equipment Hatch	240	2,1	N/A	N/A	28	721	91
A1400[a]	Personnel Hatch	1,390	2,1	N/A	N/A	350	6,758	47
A1500[a]	Mechanical Penetration	812	3,2	N/A	N/A	215	3,678	134
A1600[a]	Electrical Penetration	319	3	N/A	N/A	-138	557	53
A1800[a]	Aux Feed Pump Rm	247	3,2	N/A	N/A	148	1,278	279
A1900[a]	HV-9 Area	510	2,1	N/A	N/A	131	2,563	186
A2100[a]	RWST (Tk-4)	5,293	1	N/A	N/A	3,602	54,719	148
A2200[a]	BWST	3,865	1	N/A	N/A	7,270	43,189	190
A2300[a]	PWST	1,262	1	N/A	N/A	668	3,258	83
A2400[a]	Test Tanks	778	1	N/A	N/A	956	4,300	180

Table 5-1B
Survey Area Classification-Structural Foundation Footprints

Package Number	Survey Area-Structures	Interior		Exterior		Mean Direct Beta dpm/100cm²	Maximum Direct Beta dpm/100cm²	Approx. Survey Area Size (Meters²)
		Sigma [g] (dpm/100 cm²)	Class	Sigma (dpm/100 cm²)	Class			
A2600[a]	LSA Bld Slab	TBD [f]	2,1	N/A	N/A			291
B0200[a]	Control Rm	317	3	N/A	N/A	216	1054	334
B0400[a]	Fire Pump House	317	3	N/A	N/A	10	840	104
B0500[a]	Turbine Building	727	2	N/A	N/A	62	8614	3723
B0700[a]	Service Bld.Cold Side	299	3,2 [d]	N/A	N/A	80	1622	3293
B0800[a]	Fuel Oil Storage Bld.	298	3	N/A	N/A	-83	451	200
B0900[a]	Diesel Generators Rooms	223	3	N/A	N/A	-177	412	Included in Turbine Bldg
B1000[a]	Aux. Boiler Rm.	354	2	N/A	N/A	183	1310	Included in Turbine Bldg
B1100[a]	Circ Water Pump House	319	3	N/A	N/A	-334	673	407
B1200[a]	Administration Bld.	432	3	N/A	N/A	293	1628	784
B1300[a]	WART Bld.	542	3	N/A	N/A	-146	1164	242
B1400[a]	Information Center	313	3	N/A	N/A	295	1929	372
B1500[a]	Warehouse 2	208	3	N/A	N/A	96	539	1900

Table 5-1B
Survey Area Classification-Structural Foundation Footprints

Package Number	Survey Area-Structures	Interior		Exterior		Mean Direct Beta dpm/100cm^2	Maximum Direct Beta dpm/100cm^2	Approx. Survey Area Size (Meters^2)
		Sigma[g] (dpm/100 cm^2)	Class	Sigma (dpm/100 cm^2)	Class			
B1600[a]	Training Annex	144	3	N/A	N/A	-13	708	375
B1700[a]	Staff Bld.	374	3	TBD[b]	3	129	952.9	1431
B1900[a]	Bailey House	327	3	TBD[b]	3	612	6,524	195
B2000[a]	Bailey Barn Slab	245	3	N/A	N/A	-97	307	332
B2400	Staff Bld.-Turbine Tunnel	381	3	N/A	N/A	19	576	116
B2500	Relay House	257	3	N/A	N/A			56
D3400	LLWSB (vent and drain)	1300	3	N/A	N/A	457	3099	N/A

Table 5-1C
Survey Area Classification-Land

Package Number	Survey Area- Land	Sigma [g] (pCi/g) Cs-137	Classification	Mean Cs-137 pCi/g	Max. Cs-137 pCi/g	Approx. Survey Area Size (Meters^2)
R0100	RCA yard West (Expanded to include portions of R0200, R0900 & R1000)	1.33	2,1 [d]	15.95	156.0	17,902
R0200	Yard East (Minus portion incorporated into R0100)	0.17	3	0.17	0.64	28,748
R0300	Roof and Yard Drains	N/A	3	0.33	0.53	Incorporated into R0100
R0400	Forebay (Expanded to include portion of R1000)	TBD [e]	2,1 3 dike surface soil	TBD [e]	TBD [e]	12,191
R0500	Bailey Point	0.28	3,2,1	0.36	1.09	16,046
R0600	Ball Field (Incorporated into R1800)	See R1800	See R1800	See R1800	See R1800	Incorporated into R1800
R0700	Construction Debris Landfill (Incorporated into R1800)	See R1800	See R1800	See R1800	See R1800	Incorporated into R1800

Table 5-1C
Survey Area Classification-Land

Package Number	Survey Area- Land	Sigma [g] (pCi/g) Cs-137	Classification	Mean Cs-137 pCi/g	Max. Cs-137 pCi/g	Approx. Survey Area Size ($Meters^2$)
R0800	Admin and Parking Area (Minus portion incorporated into R1800)	0.13	3	0.18	0.37	31,057
R0900	Balance of Plant Areas (Minus portion incorporated into R0100 and R1800)	0.48	3	0.49	1.5	35,975
R1000	Foxbird Island (Minus portion incorporated into R0100 and R0400)	0.23	3	0.26	0.86	56,822
R1100	Roof and Yard Drains	NA	3	0.07	0.09	Incorporated into FR 0200
R1200	LLWSB Yard (Incorporated in R1300)	See R1300	See R1300	See R1300	See R1300	Incorporated into R1300
R1300	ISFSI (Expanded to include R1200 and portion of R2100)	0.07	3,2,1	0.09	0.28	29,240
R1500	Ash Road Area	NA	NI [c]	0.08	0.21	NA
R1600	Area West of Bailey Cove	NA	NI [c]	0.46	1.43	NA
R1700	Area North of Ferry Road	NA	NI [c]	0.47	1.55	NA

Table 5-1C
Survey Area Classification-Land

Package Number	Survey Area- Land	Sigma [g] (pCi/g) Cs-137	Classification	Mean Cs-137 pCi/g	Max. Cs-137 pCi/g	Approx. Survey Area Size ($Meters^2$)
R1800	Bailey House Land Area	0.23	3	0.25	0.83	367,000
R2000	Diffuser	TBD [c]	3	0.10	0.13	TBD
R2100	Maintenance Yard (Incorporated into R1300 and R1800)	See R1300 & R1800	See R1300 & R1800	See R1300 & R1800	See R1300 & R1800	Incorporated into R1300 & R1800
R2300	SFPI Substation Slab Area (Incorporated into R0100)	See R0100	See R0100	See R0100	See R0100	Incorporated into R0100
R2900	Roads/Railroad Final Verification. [h]	See R1800	See R1800	See R1800	See R1800	$1000m^2$ roads $500m^2$ railroads

Notes for Tables 5-1A, 5-1B and 5-1C:

a. Structural footprint may be incorporated into land area as indicated in Table 5-1C.
b. Exterior characterization will be conducted if buildings selected to remain standing
c. "NI" refers to Non Impacted
d. Contains known sub-surface or sub-slab residual activity
e. To be determined upon opening the system or other pending characterization efforts
f. Current background radiation levels preclude accurate survey. (Radioactive waste is still being packaged and stored in this area). Area will be surveyed when background allows.
g. Sigma values listed were developed using characterization data. Sigmas may be recalculated based on post-remediation survey data.
h. If contamination of 0.5 DCGL is detected in the last 100m prior to exit, an investigation as to source and impact will be conducted.

Table 5-1D
Land Areas Possibly Augmented by Backfilled Structural Footprints

Land Area Package No.	Land Area Description	Structure Package No.	Structure Area Description
R0100	RCA Yard West	A0500	DWST
		A0900	Service Bldg. Hot Side
		A1200	RCA Bldg
		A1300	Equipment Hatch
		A1400	Personnel Hatch
		A1500	Mechanical Penetration
		A1600	Electrical Penetration
		A1800	Aux Feed Pump Rm
		A1900	HV-9 Area
		A2100	RWST (Tk-4)
		A2200	BWST
		A2300	PWST
		A2400	Test Tanks
		A2600	LSA Bld

Table 5-1D
Land Areas Possibly Augmented by Backfilled Structural Footprints

Land Area Package No.	Land Area Description	Structure Package No.	Structure Area Description
R0200	Yard East	B0200	Control Rm
		B0500	Turbine Bldg
		B0700	Service Bldg. Cold Side
		B0800	Fuel Oil Storage Bldg
		B0900	Diesel Generator Rooms
		B1000	Aux. Boiler Rm
		B1100	Circ Water Pump House
		B1200	Administrative Bld. (Front Office)
		B1300	WART Bldg
		B2100	Lube Oil Storage Rm.
		B2200	Cold Machine Shop
R0800	Admin and Parking Area	B1400	Information Center
		B1600	Training Annex
		B1700	Staff Bldg.

Table 5-1D
Land Areas Possibly Augmented by Backfilled Structural Footprints

Land Area Package No.	Land Area Description	Structure Package No.	Structure Area Description
R0900	Balance of Plant Areas	B0400	Fire Pump House
		B2600	Warehouse 5
R1800	Bailey House Land Area	B1900	Bailey House
		B2000	Bailey Barn
R2600	Duct Banks	N/A	Underground Duct Banks

Table 5-1E
Survey Area Classification-Embedded and Buried Pipe

Package Number	Description	Classification
C0300	Containment Spray	Class 1
C2000	Containment Foundation Drains	Class 2
D0400	Sanitary Waste (2)	Class 3
D0500	Circulating Water	Class 3
D0700	Fire Protection (Water)	Class 3
D3500	Storm Drains	Class 1/3
D3600	Roof Drains (1)	Class 1/3
D3700	Containment Building Penetrations	Class 1
D0600	Service Water	Class 1/3

Note 1: Roof Drains will be surveyed as part of D3500 Storm Drains
Note 2: D0400 may require additional characterization surveys (see 5.2.4).

5.2.4 Discussion of Initial Classification

During the initial site characterization of Survey Area D2600, "Bailey House" (Environmental Services Laboratory Systems) some elevated direct readings were identified in sink drains and traps. When these pipes were checked by gamma spectrum analysis, there were no plant derived radionuclides detected. Survey Area R0400 "Forebay," received limited survey during initial characterization. This area has been subjected to further characterization to support dose modeling, remediation and FSS efforts. (See Section 2.5.3e and Attachment 2H.) Survey Area D0400 "Sewage Treatment Plant" is currently classified as a Class 3. There were some contamination events recorded for this system in the historical site assessment; however, the systems and components affected by these events have since been replaced. Additional characterization may be required to confirm classification.

The classification tables do not show any previously (Rev. 0) classified above grade structural elevations such as A0200 "Containment El. 20 ft.," A0300 "Containment El. 46 ft.," A0700 "PAB El. 21 ft.," A0800 "PAB El. 36 ft.," B0100 "Turbine Bld El. 61 ft.," B0300 "Motor Control Center," B0600 "Turbine Bld El. 39 ft.," or B2300 "Cable Vault." These area classifications have been removed since they are associated with upper level elevations of buildings which will be demolished and the resulting debris disposed of offsite.

A detailed discussion of the basis for the classification of the embedded piping and buried piping listed in Table 5-1E is provided in Attachment 5-A.

5.2.5 Changes in Classification

Initial classification of site areas is based on historical information and site characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance and any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the final status survey as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Once the FSS of a given survey unit begins, the basis for any reclassification will be documented. If during the conduct of a FSS survey sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit in accordance with LTP Section 5.6, the survey may be terminated without completing the survey unit package.

5.2.6 Selected Survey Area Boundaries Redefined

During the review of initial and continuing characterization, it was noted that there were some survey areas that contained areas of elevated activity that were adjacent to one another. The boundaries of these survey area have been redrawn for FSS to consolidate the elevated areas into one survey area, where practical. Other survey areas have been combined for efficiency because they have the same classification and characteristics. Table 5-1C and Figure 5-3 reflects the redefinition of these boundaries which are described in further detail below:

R0100 "RCA Yard West" - Portions of land areas formerly belonging to Survey Areas R0200 "Yard East", R0900 "Balance of Plant Areas", R1000 "Foxbird Island" and R2300 "Spent Fuel Pool Island (SFPI) Substation Slab Area" have been incorporated into R0100. These areas are adjacent to the previous border line and had some indications of elevated activity. Portions of land areas formerly associated with R0200 and R2300 showed elevated activity adjacent to the Borated Water Storage Tanks (BWST's), Primary Water Storage Tank (PWST) and the SFPI Pagoda. Portions of land areas formerly associated with R0900 and R1000 showed elevated activity above the north end of the Forebay. This adjustment to R0100 consolidated those adjacent areas of elevated activity into areas with a similar classification.

R0400 "Forebay" - Land areas formerly belonging to Survey Area R1000 "Foxbird Island," including the entire west bank of the forebay, have been added to R0400. These areas have been added R0400 to consolidate areas associated with the forebay.

R1800 "Bailey House Land Area" - Land areas formerly belonging to Survey Areas R0600 "Ball Field"(entire area), R0700 "Construction Debris Landfill"(entire area), and portions of land areas formerly belonging to Survey Areas R0800 "Admin and Parking Area," R0900 "Balance of Plant Areas," and R2100 "Maintenance Yard" have been added to R1800. These areas have been combined for efficiency because they have the same classification and characteristics. Any unique historical or site characterization information associated with these areas is being maintained to support final status surveys, judgmental scanning decisions and any followup investigations.

R1300 "Independent Spent Fuel Storage Installation" (ISFSI) - Land areas formerly belonging to Survey Areas R1200 "Low Level Waste Storage Building Yard" (entire area) and portions of land areas formerly belonging to Survey 2100

"Maintenance Yard" have been added to R1300. These areas were combined for efficiency because they were associated with the ISFSI construction project.

5.3 Establishing Survey Units

5.3.1 Survey Unit

Each survey area listed in Tables 5-1A - 5-1E may be divided into discrete survey units. Survey units are areas that have similar characteristics and contamination levels. Survey units are assigned only one classification. The site and facility are surveyed, evaluated, and released on a survey unit basis.

a. Survey Unit Size

NUREG-1727, Appendix E, provides suggested sizes for survey units. However, as stated in NUREG-1727, page E3, the suggested survey unit sizes were based on a finding of reasonable sample density and consistency with commonly used dose modeling codes. The Basement Fill model described in Section 6 is, by necessity, not generally consistent with the "commonly used codes" because the basic conditions are different, i.e., filled basement versus standing buildings or soil contamination.

For standing buildings, the MARSSIM recommends a survey unit size of 100 m^2 floor area in a Class 1 area based on the dose model assumption that a 100 m^2 office would be occupied. The source term in this case is essentially the 100 m^2 floor surface; 180 m^2 if the lower walls are included. For soil, the recommended survey unit size for a Class 1 area was conservatively based on the dose model assumption of a 2,000 m^2 resident farm. The source term area in this case is 2,000 m^2. For basement surfaces, the non-containment basement fill model assumes an area of 4182 m^2. Therefore, the source term, and survey unit size, for basements should be based on an area of 4182 m^2. For containment, the model assumes an area of 1130 m^2, so the survey unit size would be limited to 1130 m^2.

However, using a 4182 m^2 Class 1 survey unit size may not result in a "reasonable sample density" per MARRSIM. This is somewhat difficult to evaluate since MARSSIM provides no explanation for the statement and the statement is somewhat inconsistent with the MARSSIM premise that sample size is determined using DQO's and a statistically based method. To provide a rationale for a "reasonable sample density" finding, the recommended sample densities for standing building and soil surveys were evaluated.

Using the recommended survey unit sizes for standing buildings and soil, and assuming a sample size of 14 per survey unit (for the sign test with an α and $\beta = 0.05$ and relative shift = 3 as presented in Section 5.2), sample densities of 1/13 m^2 for standing buildings and 1/143 m^2 for soil would be required. The primary reason for the difference in sample densities for standing buildings and soil is the source term assumptions in the dose model as described previously. Both sample densities are considered reasonable in MARSSIM. In accordance with the same logic, a sample density of 1/298 m^2 would be called for in a 4182 m^2 survey unit (14/4182).

However Maine Yankee proposes to use a much higher sample density 1/50 m^2 for the Class 1 basement surfaces. There is no sample density limitation for Class 2 or Class 3 basement surfaces. This value satisfies the MARSSIM "reasonable sample density" criteria since it is at the low end of the range of the recommended sample densities for standing building and soil and is consistent with the dose model assumptions. The number of samples in a survey unit will, in all cases, meet or exceed the minimum number required per survey unit in MARSSIM. For example, if a survey unit size is 280 m^2, the sample density will be 1/20 m^2 to maintain the minimum 14 samples per survey unit. On the other hand, if a survey unit size is 1000 m^2, 20 samples will be collected as opposed to the 14 that are statistically required, to maintain the minimum 1/50 m^2density. In addition, if sample size adjustments are required because of scan survey MDA, the required higher sample number will be used, regardless of the sample density. The non-containment Basement surface survey unit size will be limited to 2000 m^2. The containment Basement surface survey unit size will be limited to 1130 m^2.

It is important to recognize that 100% scan survey of accessible areas is required in a Class 1 area. This provides a high level of confidence that no significant contamination will be missed. The fixed point measurements or samples are used in the statistical analysis, assuming a random distribution. For the statistical analysis, a sample density of 1/50 m^2 that meets or exceeds the required MARSSIM minimum number is considered sufficient.

The actual survey unit areas and location designated within a survey area, particularly in the building basements, will be based on decommissioning operations and schedule as well as the physical configuration of the areas. Basement survey units will, in most cases be on the order 1000 m^2 or less. Scale drawings of building or land areas, and walkdowns, will be used to calculate the surface area of the basement surfaces or soil within a survey area.

The survey unit sizes are related to the dose models described in Section 6. Therefore, the standing structure survey units are based on the building occupant scenario pathways and the basement structure survey units are based on the basement fill scenario pathways. The typical survey unit sizes for building basements, soil, and standing buildings are listed in Table 5-2.

Table 5-2
Survey Unit Areas

Class	**Suggested Survey Unit Area**		
	Standing Structures	**Basement Structures**	**Land**
1	180 m^2*	2000 m^2 **	2000 m^2
2	180 to 1000 m^2	2000 m^2 **	2000 to 10^4 m^2
3	No Limit	No Limit	No Limit

* includes floor and lower walls
** 1130 m^2 for containment basement structure

Table 5-2 lists the survey unit size for basement structures as 2,000 m^2 surface area. Note that for embedded piping, this size is also justified since the dose model for residual radioactivity in embedded piping is identical to that used for basement structure contamination. Therefore, the same survey unit size of 2,000 m^2 is appropriate. For buried piping, the 2,000 m^2 survey is not appropriate. In fact, all of the buried pipe could be considered as one survey unit based on dose modeling assumptions. The dose model for buried piping assumes that the entire inventory of residual radioactivity in all buried piping expected to remain is instantaneously removed from the pipe surface and mixed into a volume of soil equal to the 141 m^3, which is the volume of all the buried pipe. Under the assumption that this 141 m^3 of soil is excavated and uniformly spread over a 15 cm layer on the ground surface, it would cover an area of 940 m^2. This is less than the 2,000 m^2 that would be allowed for surface soil. Therefore, all of the buried piping could be included in one survey unit. In actuality, as listed in Table 5-1, the buried piping will be surveyed as several distinct survey units based on physical and system considerations.

b. Site Reference Coordinate System (Reference Grid)

A reference coordinate system is used for impacted areas to facilitate the identification of survey units within the survey area. The reference coordinate system is basically an X-Y plot of the site area referenced to the state of Maine mercator projections as shown in Figures 5-4 and 5-5. Once the reference point is established, grids may be overlaid parallel to lines of latitude and longitude.

5.4 Survey Design

This section describes the methods and data required to determine the number and location of measurements or samples in each survey unit, the coverage fraction for scan surveys, and requirements for measurements in background reference areas. The design activities described in this section will be documented in a survey package for each survey unit. Survey design includes the following:

a. Scan Survey Coverage

b. Sample Size Determination

c. Background Reference Areas as necessary

d. Reference Grid and Sample Location

LTP Section 5.4.5 describes the process for designing, developing and reviewing survey packages.

5.4.1 Scan Survey Coverage

The area covered by scan measurement is based on the survey unit classification as described in NUREG 1727 and as shown in Table 5-3 below. A 100% accessible area scan of Class 1 survey units will be required. The emphasis will be placed on scanning the higher risk areas of Class 2 survey units such as soils, floors and lower walls. Scanning percentage of Class 3 survey units will be performed on likely areas of contamination based on the judgement of the FSS engineer.

Table 5-3 Scan Measurements			
	Class 1	Class 2 *	Class 3
Scan Coverage	100%	10-100%	1 to 10%

* For Class 2 Survey Units, the amount of scan coverage will be proportional to the potential for finding areas of elevated activity or areas close to the release criterion in accordance with MARSSIM Section 5.5.3. Accordingly, Maine Yankee will use the results of individual measurements collected during characterization to correlate this activity potential to scan coverage levels.

5.4.2 Sample Size Determination

NUREG-1727 describes the process for determining the number of survey measurements necessary to ensure a data set sufficient for statistical analysis. Sample size is based on the relative shift, the Type I and II errors, sigma, and the specific statistical test used to evaluate the data.

Alternate processes may be used if such gain NRC and industry acceptance between the time this plan is adopted and the commencement of final survey activities. However, any new technologies must still meet the applicable requirements of this plan for calibration, detection limit, areal coverage, operator qualification, etc.

a. Determining Which Test Will Be Used

Appropriate tests will be used for the statistical evaluation of survey data. Tests such as the Sign test and Wilcoxon Rank Sum (WRS) test will be implemented using unity rules, surrogate methodologies, or combinations of unity rules and surrogate methodologies, as described in MARSSIM and NUREG-1505 chapters 11 and 12.

If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign test will be used. If background is a significant fraction of the DCGL, the Wilcoxon Rank Sum (WRS) test will be used.

b. Establish Decision Errors

The probability of making decision errors is controlled by hypothesis testing. The survey results will be used to select between one condition of the environment (the null hypothesis) and an alternate condition (the alternative hypothesis). These hypotheses, chosen for MARSSIM Scenario A, are defined as follows:

Null Hypothesis (H_0): The survey unit does not meet the release criteria.
Alternate Hypothesis (H_a): The survey unit does meet the release criteria.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criteria. It occurs when the null hypothesis is rejected when it is true. The probability of making this error is designated as "α". A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criteria. This occurs when the Null Hypothesis is accepted when it is not true. The probability of making this error is designated as "β".

Appendix E of NUREG 1727 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the NUREG 1727 guidance, α will be set at 0.05. A β of 0.05 will initially be selected based on site specific considerations. The β may be modified, as necessary, after weighing the resulting change in the number of required survey measurements against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criteria.

c. Relative Shift

The relative shift (Δ / σ) is calculated. Delta (Δ) is equal to the $DCGL_w$ minus the Lower Boundary of the Gray Region (LBGR). Calculation of sigmas has been discussed in Section 5.2.3 and values are provided in Tables 5-1A-C. The sigmas used for the relative shift calculation may be recalculated based on the most current data obtained from post-remediation or post-demolition surveys; or from background reference areas, as appropriate The LBGR is initially set at 0.5 times the $DCGL_w$, but may be adjusted to obtain an optimal value, of normally between 1 and

3 for the relative shift.

Lower Boundary of the Gray Region

The Lower Boundary of the Gray Region (LBGR) is the point at which the Type II (β) error applies. The default value of the LBGR is set initially at 0.5 times the DCGL. If the relative shift is greater than 3, then the number of data points, N, listed for the relative shift values of 3 from Table 5-5 or Table 5-3 in NUREG -1575 will normally be used as the minimum sample size. Use of a relative shift greater than 3 requires approval by an FSS Engineer. If the minimum sample size results in a sample density less than the required minimum density (see Section 5.3.1), the sample size will be increased accordingly.

Sigma

Sigma values (estimate of the standard deviation of the measured values in a survey unit, and/or reference area) were initially calculated from characterization data. These sigma values can be used in FSS design or more current post-remediation sigma values can be used. The use of the sigma values from the characterization data will be conservative for the sample size determination since the recalculated post-remediation sigmas are expected to be smaller. The sigma values for survey areas listed in Table 5-1 which contain survey units with two different classifications, will be evaluated to ensure that the sigma conservatively represents the contaminant distribution of each associated survey unit; otherwise a specific sigma value will be developed.

The sigma values for structure surfaces were calculated using the GTS characterization data measurements on concrete that were less than 20,000 dpm/100 cm^2,which was a preliminary estimate of the $DCGL_w$. This assumes that areas above 20,000 dpm/100 cm^2 will be remediated. Using a lower concentration should lower the sigma estimate. This method should be conservative since many contaminated areas that are near the $DCGL_w$ or near other remediated areas will likely also be remediated which would serve to reduce the higher values and the resulting sigma. The characterization measurements above 20,000 dpm/100 cm^2 were not truncated to 20,000 dpm/100 cm^2 and included since it is likely that any area remediated will be well below the $DCGL_w$. The sigmas for soil areas were calculated using the GTS characterization data on measurements greater than MDA, and less than 8 pCi/g Cs-137. This should provide a conservative estimate of sigma for any Cs-137 $DCGL_w$ at 8 pCi/g or less.

The number of structure surface measurements taken to support the calculation of sigmas indicated in Tables 5-1A and 5-1B ranged from 7 to 98 per survey area. The number of soil measurements taken to support the calculation of sigmas indicated in Table 5-1C ranged from 5 to 73 per survey area. The structure sigmas calculated in Tables 5-1A and 5-1B represent the total gross beta activity measured down to the beta energy of C-14. If nuclides are present that have beta energies greater than that associated with C-14 they would be included in the gross measurement. The method for determining the average energy of the beta emitters is described in a supporting engineering calculation. Table 5-3a shows that the calibration sources with average beta-particle energies of $\leq$ 0.107 MeV are conservative with respect to the energy spectrum presented in the table.

The soil sigmas calculated in Table 5-1B are based upon distributed Cs-137. Sigmas may be recalculated based upon data obtained from post-remediation or post-demolition surveys.

There are some areas in containment, RCA, Fuel and Spray buildings that presently show large sigma values. After these areas are remediated, the sigma values are expected to be significantly lower. Where areas are remediated or changed, new sigma values may be calculated by taking measurements in the survey area at about 5 to 20 locations as recommended in Section 5.5.2.2. of NUREG 1575.

Table 5-3a
Contaminated Media Beta Energy (KeV)

Nuclide	Fraction 2004	Average Beta Energy (KeV)	Average Beta Energy Contribution (KeV)
H-3	2.36E-02	5.68	0.134
Fe-55	4.81E-03	0	0
Co-57	3.06E-04	0	0

Table 5-3a Contaminated Media Beta Energy (KeV)			
Nuclide	**Fraction 2004**	**Average Beta Energy (KeV)**	**Average Beta Energy Contribution (KeV)**
Co-60	5.84E-02	95.79	5.59
Ni-63	3.55E-01	17.13	6.08
Sr-90	2.80E-03	195.80	0.55
Cs-134	4.55E-03	156.80	0.71
Cs-137	5.50E-01	170.80	93.94
Total			107.01

d. Wilcoxon Rank Sum (WRS) Test Sample Size

The number of data points, N, to be obtained from each reference area or survey unit are determined using Table 5-3 in NUREG-1575. The table includes the recommended 20% adjustment to ensure an adequate sample size.

e. Sign Test Sample Size

The number of data points is determined from Table 5-5 in NUREG-1575 for application of the Sign Test. This table includes the recommended 20% adjustment to ensure an adequate sample size.

f. Elevated Measurement Comparison (EMC) Sample Size Adjustment

If the scan MDC is greater than the $DCGL_W$, the sample size will be calculated using the equation provided below. If N_{EMC} exceeds the statistically determined sample size (N), N_{EMC} will replace N.

$$N_{EMC} = A/A_{EMC}$$

Where: N_{EMC} is the elevated measurement comparison sample size
A is the survey unit area
A_{EMC} is the area corresponding to the area factor calculated using the MDC_{scan} concentration.

5.4.3 Background Reference Areas

Background reference area measurements are required when the WRS test is used, and background subtraction may be used with the Sign test, under certain conditions such as those described in Chapter 12 of NUREG 1505. The reference area measurements will be collected using the methods and procedures required for Class 3 final survey units. For soil, reference areas will have a soil type as similar to the soil type in the survey unit as possible. When there is a reasonable choice of possible soil reference areas with similar soil types, consideration will be given to selecting reference areas that are most similar in terms of other physical, chemical, geological, and biological characteristics. For structure survey units that contain a variety of materials with markedly different backgrounds, a reference area will be selected that has similar materials. If one material is predominant or if there is not too great a variation in background among materials, a background from a reference area containing only a single material is appropriate when it is demonstrated that the selected reference area will not result in underestimating the residual radioactivity in the survey unit.

It is understood that background reference areas should have physical characteristics (including soil type and rock formation) similar to the site and shall not be contaminated by site activities. In general, Maine Yankee commits to using background reference areas, when possible, that are offsite. If non-contaminated onsite areas are to be used, then Maine Yankee will verify and justify its use by appropriate comparison with samples from appropriate off-site locations. A White Paper (technical basis document) was developed for dealing with background (Reference 5.12.35). Information from the White Paper has been included in the appropriate FSS procedure (Reference 5.12.27).

Should significant variations in background reference areas be encountered, appropriate evaluations will be performed to define the background concentration. As noted in NUREG 1727, Appendix E, Section 3.4, the Kruskal-Wallis test can be conducted in such circumstances to determine that there are no significant differences in the mean background concentrations among potential reference areas. Maine Yankee will consider this and other statistical guidance in the evaluation of apparent significant variations in background reference areas.

If material background subtraction is performed , the sigma value used will take into account the variability of material background.

5.4.4 Sample Grid and Sample Location

Sample location is a function of the number of measurements required, the survey unit classification, and the contaminant variability.

a. Sample Grid

The reference grid is primarily used for reference purposes and is illustrated on sample maps. Physical marking of the reference grid lines in the survey unit will only be performed when necessary. For the sample grid in Class 1 and 2 survey units, a randomly selected sample start point will be identified and sample locations will be laid out in a square grid pattern[2] at distance, L, from the start point in both the horizontal and vertical directions. The sample and reference grids are illustrated on sample maps and may be physically marked in the field. For Class 3 survey units, all sample locations are randomly selected, based on the reference grid. An example is shown in Figures 5-4 and 5-5. Global Positioning System (GPS) instruments may be used in open land areas to determine reference or sample grid locations within the survey area. The manufacturer's specifications indicate a horizontal accuracy of 21 feet to 45 feet for the GPS system. Digital cameras may be employed to provide a lasting record of survey location within the survey unit. When used, these photographic records will be linked to landmark and directional information to ensure reproducibility.

Note that GPS is only one method that could be used to locate land survey points. Maine Yankee is currently using a site reference grid based on the Maine mercator system and distances and angles from fixed reference points to locate survey points. If GPS is to be the sole method used to locate survey points, a more accurate system will be obtained.

b. Measurement Locations

Measurement locations within the survey unit are clearly identified and documented for purposes of reproducibility. Actual measurement locations are identified by tags, labels, flags, stakes, paint marks, geopositioning units or photographic record. An identification code matches a survey location to a particular survey unit.

Sample points for Class 1 and Class 2 survey units are positioned in a systematic pattern or grid throughout the survey unit by first randomly selecting a start point coordinate. A random number generator is used to

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2 Note that both NUREG 1575 and 1505 recognize both the rectangular and the triangular grid pattern grid method as acceptable.

determine the start point of the square grid pattern. The grid spacing, L, is a function of the area of the survey unit as shown below:

$$L = \sqrt{\frac{A}{n}} \qquad \text{for a square grid}$$

where:

A = the area of the survey unit,

n = the number of sample points in the survey unit.

Sample points are located, L, distance from the random start point in both the X and Y directions.

Random measurement patterns are used for Class 3 survey units. Sample location coordinates are randomly picked using a random number generator.

Measurement locations selected using either a random selection process or a randomly-started systematic pattern that do not fall within the survey unit or that cannot be surveyed due to site conditions are replaced with other measurement locations as determined by the FSS Specialist or FSS Engineer.

5.4.5 Survey Package Design Process

A Final Status Survey Package is produced for each survey area. The survey package is a collection of documentation detailing survey design, survey implementation and data evaluation for a Final Status Survey of a survey area.

Maine Yankee applies the 10CFR50, App. B requirements for field and laboratory counting equipment, as well as the corrective action process to address data or programmatic discrepancies. Using the existing Part 50, App. B program precludes developing redundant measures for FSS activities. (See also Section 5.10.5.)

a. Survey Package Initiation

Each survey area and package is assigned a unique identification number. To allow continuity of area identification, the protocol used for identifying survey areas during the characterization survey is used, as appropriate.

Numbers dissimilar to those used for characterization survey may be necessary if survey boundaries are modified.

b. Review of HSA, Characterization Surveys

The FSS Specialist gathers and reviews historical data applicable to the survey area. Historical information that will be used for survey design is filed in the survey package. Sources of historical data include:

1. Historical Site Assessment
2. Characterization Survey (Initial and Continuing)[3]
3. Classification basis
4. 50.75(g) files
5. Operational Survey Records

c. Survey Area Walkdown

The FSS Specialist performs a walkdown to gather information about the physical characteristics of the survey area. The walkdown provides the Specialist an opportunity to determine if any physical or safety related interferences are present that may affect survey design or survey implementation, and to determine any support activities necessary to implement surveys. The walkdown is documented and filed in the survey package.

Following the walkdown, representative maps of the survey area are prepared.

d. Survey Design

Survey Design is the process of determining the number, type and location of survey measurements or samples required for each survey unit within a survey area. The various aspects of survey design are documented and filed in the survey package. The survey unit design process is controlled by approved procedures.

The size and number of survey units for a survey area is determined based on area classification, modeling assumptions used to develop DCGL's and the layout of the survey area. The FSS Specialist will divide the area into discrete survey units as appropriate. Each survey unit is numbered

[3] For additional explanation of initial and continuing characterization surveys, see Section 2.1.

sequentially. The FSS Specialist provides a description of each survey unit including survey unit size, classification and location. The types of material (i.e. soil, concrete, etc.) found in the survey unit and survey measurement and/or sampling methods are identified.

The FSS Engineer calculates the number of measurements or samples required for each survey unit in accordance with NUREG-1575. The FSS Engineer also calculates required investigation setpoints for survey measurements.

The FSS Specialist determines measurement/sample locations based on the classification of the survey unit and in accordance with NUREG-1575. A survey map is prepared of each survey unit. A sample and/or reference grid is superimposed on the map to provide an (x,y) coordinate system. The FSS Specialist generates random numbers, between 0 and 1, which are multiplied by the maximum x and y axis values of the sample grid. This provides coordinates for each sample location, or a random start location for systematic grid, as appropriate. The measurement/sample locations are plotted on the map. Each measurement/sample location is assigned a unique identification code which identifies the measurement/sample by Survey Area, Survey Unit, Material and sequential number.

The FSS Specialist determines the appropriate instruments and detectors, instrument operating modes and survey methods to be used to collect and analyze data.

The FSS Specialist prepares written survey instructions that incorporate the requirements set forth in the survey design. Direction is provided for selection of instruments, count times, instrument modes, survey methods, required documentation, alarm/investigation setpoints, alarm actions, background requirements and other appropriate instructions. The instructions also direct the appropriate instrument set up to ensure collected survey data is saved and downloaded to the appropriate files. In conjunction with the survey instructions, survey data forms, indicating desired measurements, are prepared to assist in survey documentation.

The FSS Engineer reviews the survey design and instructions and verifies, or has a competent person verify, all calculations. The FSS Engineer ensures that appropriate instruments, survey methods and sample locations have been properly identified. Once approved, the survey design and instructions are filed in the survey package.

The Superintendent of FSS reviews the survey package and authorizes survey implementation.

e. Survey Area Turnover

Prior to performing Final Status Surveys, the FSS Superintendent coordinates with appropriate site superintendents to ensure decommissioning activities, area remediation and housekeeping are complete. The FSS Superintendent may direct Radiation Protection to perform surveys to verify that the area meets the radiological criteria for performance of the Final Status Survey. When satisfied, the FSS Superintendent will direct the area to be posted, as appropriate, to indicate that the area is controlled for the performance of Final Status Surveys. Access controls are implemented to prevent contamination of areas during and following Final Status Surveys.

f. Survey Implementation

Survey areas and/or locations are identified by gridding, markings, or flags as appropriate. The FSS Supervisor performs a pre survey briefing with the survey technicians during which the survey instructions are reviewed. The technicians gather instruments and equipment as indicated and perform surveys in accordance with the appropriate procedures. Technicians are responsible for documenting survey results and maintaining custody of samples and instrumentation. At the completion of surveys, technicians return instruments for downloading and prepare samples for analysis.

Survey instruments provided to the technicians are prepared in accordance with appropriate procedures and the survey instructions. Instruments are performance checked prior to and following surveys. Any data collected in data logging instruments is downloaded and a hard copy printed out. The download hard copies, surveyor's data sheets and sample counting reports are reviewed and forwarded for inclusion in the survey package. The FSS Supervisor is notified of any data that exceeds investigation criteria so that appropriate investigation surveys and remediation can be performed as necessary. The downloaded data file is backed up to the system server and to appropriate storage media on a routine basis.

Several quality control measures and features have been developed for the implementation phase of the final status survey program. These elements typically include:

- Pre-implementation briefings between FSS design and implementation personnel,
- Pre-implementation area walkdowns,
- Survey location verification,
- Daily survey area background measurements,
- Instrument source checks before and after survey activities and
- Conduct of surveys in the peak trap mode, thereby providing a record of the maximum scan value for any scan grid.

g. Data Evaluation

The FSS Specialist reviews survey data, data downloads and counting reports to verify completeness, legibility and compliance with survey design. As directed by the FSS Engineer, the FSS Specialist performs the following:

1. Converts data to reporting units
2. Calculates mean, median and range of the data set
3. Reviews the data for outliers
4. Calculates the standard deviation of the data set
5. Calculates MDC for each survey type performed
6. Creates posting, frequency or quantile plots for visual interpretation of data.

The FSS Engineer reviews and verifies the statistical calculations, verifies the integrity and usefulness of the data set and determines the need for further data. The FSS Engineer will direct investigation as necessary. Once satisfied that the data are valid, the FSS Engineer will perform the appropriate statistical test and make a decision on the radiological status of each survey unit.

The data evaluation process is documented and filed in the survey package.

h. Quality Control Surveys

Following completion of Final Status Survey, the need for QC surveys (replicate surveys, sample recounts, etc.) is determined. If necessary, a QC survey package is developed and modeled after the original survey. QC measurement results are compared to the original measurement results. If QC results do not agree with the original survey, an investigation is performed. Following investigation, the FSS Engineer will decide data validity.

i. Release Record

Following data evaluation, The FSS Engineer prepares a Release Record. The Release Record describes the survey area, survey design, survey units, surveys performed and instruments used. The Release Record summarizes survey results and data evaluation. The Release Record is reviewed and approved by the FSS Superintendent and the Manager of Projects - FSS.

5.5 Survey Methods and Instrumentation

5.5.1 Survey Measurement Methods

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable procedure. The techniques for performing survey measurements or collecting samples are specified in approved procedures. Final site survey measurements include surface scans, direct surface measurements, and gamma spectroscopy of volumetric materials. In situ gamma spectroscopy or other methods not specifically described may also be used for final status surveys. If so, Maine Yankee will give the NRC 30 days notice to provide an opportunity to review the associated basis document[4] as described in LTP Section 5.3.1.

On-site lab facilities are used for gamma spectroscopy, liquid scintillation and gas proportional counting in accordance with applicable procedures. Off-site facilities are used, as necessary. No matter which facilities are used, analytical methods will be administratively established to detect levels of radioactivity at 10% to 50% of the DCGL value or below the ALARA Remediation Level, if applicable.

a. Structures

Structures will receive scan surveys, direct measurements and, when necessary, volumetric sampling.

Scan Surveys

Scanning is performed in order to locate small areas of residual activity above the investigation level. Structures are scanned for beta-gamma radiation with appropriate instruments such as those listed in Table 5-4.

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4 A Technical Basis Document was submitted for: "Forebay FSS Survey Measurement Methods (In-Situ Gamma Spectroscopy" - References 5.12.37 and 5.12.38

The measurements will typically be performed at a distance of 1 cm or less from the surface and at a scan speed of 5 cm/sec for hand-held instruments. Adjustments to scan speed and distance may be made in accordance with approved procedures. Sodium iodide detectors may be used for scanning of concrete surfaces when surface conditions would result in increased surface to detector distance (typically within 3 inches) and when the static measurement sample size is adjusted for the corresponding MDC, if necessary. In situ gamma spectroscopy may be effectively substituted for scanning surveys if technically justified following the 30 day NRC notice and opportunity to review as described previously.

Direct Measurements

Direct measurements are performed to detect surface activity levels. Direct measurements are conducted by placing the detector on or very near the surface to be counted and acquiring data over a pre-determined count time. A count time of one minute is typically used for surface measurements and generally provides detection levels well below the DCGL. (The count time may be varied provided the required detection level is achieved).

Concrete With Activated Radionuclides

Residual radioactivity within activated building materials was conservatively estimated by performing gamma spectroscopy on core slices taken from long concrete cores located in selectively higher that average neutron fluence locations for the concrete volumes represented by the cores. This activity inventory was established as the DCGL and was evaluated for dose consequences using realistic release assumptions as described in Reference 6.10.7. Because of the low dose consequences, no other final status survey requirements were established to measure the activated concrete activity. However, measurements of total activated activity were estimated using in-situ gamma spectroscopy to provide verification within the bounds of uncertainty.

Volumetric Concrete Measurements

Volumetric sampling of contaminated concrete, as opposed to direct measurements may be necessary if the efficiency or uncertainty of the gross beta measurements are too high. Volumetric concrete samples will be analyzed by gamma spectroscopy. The results will either be evaluated by

1) calculating the derived total gross beta cpm/100 cm^2 in the sample and comparing the gross beta results directly to the gross beta DCGL or 2) by using the radionuclide specific results to derive the surface activity equivalent and determine compliance using the unity rule. Use of the unity rule will require the use of a surrogate calculation to account for the radionuclides in the mixture not identified by gamma spectroscopy. This will be accomplished using the nuclide mixture listed in Tables 2-7 or 2-8 as appropriate.

Volumetric samples analyzed by gamma spectroscopy will detect the presence of radioactivity below the surface. Such sampling is typically performed following removal of paint and other surface coatings during remediation. After analysis, the data may be converted to equivalent surface activity for crack analysis.

Removable Contamination Surveys

Based on current decommissioning planning, there will be no standing buildings remaining with in the Restricted Area and only one building remaining outside the Restricted Area, namely, the switchyard relay house (per LTP Sections 3.2.4 and 6.9.1). Removable contamination surveys will be collected at discreet locations in the switchyard relay house.

b. Soil

Soil will receive scan surveys at the coverage level described in Table 5-3 and volumetric samples will be taken at designated locations. Surface soil samples will normally be taken at a depth of 0 to15 cm. Areas of sub-surface soil contamination may require sampling at a depth exceeding 15 cm. The possibility of sub-surface contamination will be considered during the survey design process and the survey design package will contain requirements for sampling soil below 15 cm. Samples will be collected and prepared in accordance with approved procedures.

Scans

Open land areas are scanned for gamma emitting nuclides. The gamma emitters are used as surrogates for the HTD radionuclides. Sodium iodide detectors are typically used for scanning. For detectors such as the SPA-3, the detector is held within a few centimeters of the ground surface and is moved at a speed of 0.25 m/sec, traversing each square meter 5 times. The area covered by scan measurements is based on the survey unit classification as described in Section 5.4.1.

Volumetric Samples

Soil materials are analyzed by gamma spectroscopy. Soil samples of approximately 1500 grams are normally collected from the surface layer (top 15 cm). If contamination below 15 cm is suspected, split spoon sampling or other methods, will be used for the final survey unless the area has already been excavated and remediated to the deep soil DCGL. If an area containing subsurface contamination has been remediated, the excavated area will be treated as a surface soil.

The areas around the RWST and Fuel Building are two of the areas that will require remediation and possibly sub-surface sampling. Subsurface sampling will be performed in accordance with the guidance in NUREG-1727, page E18, Section 11.1. The sample size for subsurface samples will be determined using the same methods described for surface soil. Per NUREG-1727, scanning is not applicable. Samples will be composited over each 1 m of depth and collected to depths at which there is high confidence that deeper samples will not result in higher concentrations. The area factors derived for surface soil will be applied to subsurface soil in Class 1 areas.

Sample preparation includes removing extraneous material and homogenizing and drying the soil for analysis. Separate containers are used for each sample and each container is tracked through the analysis process using a chain-of-custody record. Samples are split when required by the applicable FSS Quality Control procedure.

Sub-Slab Soils

Grade level foundation slabs will be removed during demolition which will afford the opportunity to sample the soil underneath the slab. The floor slabs or foundations remaining in place after demolition (at elevations less than 3 feet below grade) may be evaluated by taking samples immediately adjacent to the slab using a split spoon or core sampler depending on the contamination potential. Factors that will be evaluated to determine the need for split spoon sampling include: (1) existence of soil under the slab; (2) acceptability of alternate means of identifying the potential for sub-slab contamination, e.g., groundwater sampling; and (3) operational history.

Stored Excavated Soil

Several piles of soil have been stored on-site that were excavated from Class 3 areas. Prior to placing any soil into a pile for storage and possible future use, survey measurements are made. Scan surveys are conducted over approximately 10% of the area to be excavated using methods equivalent to FSS. Soil samples are also collected and analyzed to ensure that there is no indication of previously undetected soil contamination. Once these measurements are completed, the soil is excavated and placed into storage. The Maine Yankee soil control procedure is used to track the origin, storage location, and final disposition location of the soil. Prior to any stored soil being placed in any location on site, the sampling techniques described in Section 5.5.1.b are employed to further assure that the soil met the requirements of the area in which it was being used. This stored soil could be used for backfilling the soil excavation areas after additional volumetric sampling. Stored soil will not be used for RA basement fill. The following strategy will be followed.

Assuming the WRS test will be used, $\alpha = \beta = 0.05$, and a Δ / σ value of 3, the sample size would be 10. Based on the soil sigma data in Table 5-1C, it is likely that the Δ / σ value will be equal or greater than 3. For a Class 3 surface soil survey unit of 10,000 m^2, the equivalent volumetric sample density would be 10 samples per 1,500 m^3 (10,000 m^2 x 0.15 m depth of soil sample) or 1/150 m^3.

Using the WRS test sample size to determine a volumetric sampling frequency is consistent with the methods recommended for subsurface soil in NUREG-1727, Appendix E, Section 11.1. Regardless of the soil pile volume, a minimum of 10 samples will be collected. If the soil pile volume exceeds 1500 m^3, additional samples will be collected to ensure the 1/150 m^3 sample frequency is maintained. Soil piles from various class 3 areas may be combined prior to sampling. The origin, storage and final use of soil is controlled by an approved soil control procedure.

Soil excavated from Class 1 and 2 areas may be reused for backfill of excavated areas of the same or higher classification (eg. Class 2 stored soil may be used to backfill Class 1 or 2 excavated areas; Class 1 stored soil may be used only to backfill Class 1 excavated areas). The survey and sampling protocols will be the same or equivalent to that described above for Class 3 stored soil with the following exceptions:

1. The pre-excavation surface scan or equivalent technique will provide 100% coverage
2. The soil pile volumetric sample density will be calculated based upon a surface survey unit size of 2000 m^2 and a Δ / σ value of 0.9. Thus, the equivalent volumetric sample density would be 40 samples per 300 m^3 (2000 m^2 x 0.15 m depth of soil sample) or 1 sample per 7.5 m^3.

In-Situ Gamma Spectroscopy may be employed, as appropriate, in lieu of pre-excavation sampling and scanning. Soil routinely excavated for non-remediation purposes from areas which have been successfully FSS'ed may be used to backfill the excavation without additional survey.

c. Embedded Piping and Buried Piping

The only systems to remain after decommissioning are embedded piping and buried piping. The piping expected to remain was described in detail in the Section 2. A detailed description of the final survey methods is provided in Attachment 5A.

d. Specific Areas and Conditions

Cracks, Crevices, Wall-Floor Interfaces and Small Holes

Surface contamination on irregular structure surfaces (e.g., cracks, crevices, and holes) are difficult to survey directly. Where no remediation has occurred and residual activity has not been detected above background, these surface blemishes may be assumed to have the same level of residual activity as that found on adjacent surfaces. The accessible surfaces are surveyed in the same manner as other structural surfaces and no special corrections or adjustments have to be made.

In situations where remediation has taken place or where residual activity has been detected above background, a representative sample of the contamination within the crack or crevice may be obtained or an adjustment for instrument efficiency may be made if justifiable. If an instrument efficiency adjustment cannot be justified based on the depth of contamination or other geometry factors, volumetric samples will be collected. The total dpm/100 cm^2 contained in the volumetric sample that is attributable to the beta emitting radionuclides used to determine the DCGL will be compared directly to the concrete gross beta DCGL. As an alternative, radionuclide specific analysis, coupled with application of the unity rule may be used.

Volumetric samples analyzed by gamma spectroscopy will detect the presence of radioactivity below the surface. Such sampling is typically performed following removal of paint and other surface coatings during remediation. After analysis, the data may be converted to equivalent surface activity for crack analysis.

The accessible surfaces are surveyed in the same manner as other structure surfaces except that they are included in areas receiving judgmental scans when scanning is performed over less than 100% of the area.

Paint Covered Surfaces

Final status surveys will consider the effect of painted surfaces. Gross measurements will not be used in areas covered by thick painted surfaces that are not remediated. The surfaces will be volumetrically sampled or the coating will be removed prior to survey. No special consideration must be given to wall or ceiling areas painted before plant startup and which have not been subjected to repeated exposure to materials that would have penetrated the painted surface.

Pavement-Covered Areas

The survey design of parking lots, roads and other paved areas will be based on soil survey unit sizes since they are outdoor areas where the exposure scenario is most similar to direct radiation to surface soil. The DCGL applied to these areas will be equal to the buried piping DCGL. Scan and static gamma and beta-gamma surveys are made as determined by the survey unit design. If sub-surface contamination is possible under paved or other covered areas, sub-surface volumetric samples will be collected. Paved areas may be separate survey units or they may be incorporated into other, larger survey open land units. Surveys of paved areas will include the area within road right-of-ways to check for radioactivity relocated due to water runoff. The right-of-ways may be separate survey units.

The buried pipe model, as described in Section 6.6.8, is based on the release of surface contamination (inside piping) into soil. The potential dose from paved areas is also from the release of surface contamination into soil. The soil concentration calculated in the buried pipe model was determined assuming a surface area to soil volume ratio that was higher than would likely occur in the case of paved surfaces. This would lead to higher soil concentrations from release of contamination from the buried piping than was calculated for the paved surfaces. In addition, the buried piping DCGL was limited to ensure that the resulting hypothetical soil concentrations would be below the surface soil DCGL's. The combination of conservative assumptions included in the buried piping dose model and the similarity of the ultimate dose pathways make it suitable for application to the paved surfaces.

Forebay Sediment

The forebay is designated as a stand alone survey area. The survey area may be split into multiple survey units, i.e, the rip-rap area, the "bare rock" bottom area, and soil. The forebay area will be designated as Class 1 and sample size will be determined consistent with a Class 1 soil area. Scan survey coverage will be specific to the various media within the area because of the unique geometry considerations. The Survey Package will describe in detail the rationale for the location and percent coverage of the scan surveys.

5.5.2 Instrumentation

Radiation detection and measurement instrumentation for the final status survey is selected to provide both reliable operation and adequate sensitivity to detect the radionuclides identified at the site at levels sufficiently below the DCGL. Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field. The instrumentation will, to the extent practicable, use data logging with bar code scanning capability.

Commercially available portable and laboratory instruments and detectors are typically used to perform the three basic survey measurements: 1) surface scanning; 2) direct surface contamination measurements; and 3) spectroscopy of soil and other bulk materials, such as concrete. The Instrumentation Program Procedure controls the issuance, use, and calibration of instrumentation. Records supporting the Instrumentation Program are maintained by Document Control.

a. Selection

Radiation detection and measurement instrumentation is selected based on the type and quantity of radiation to be measured. (The instruments used for direct measurements are capable of detecting the radiation of concern to a Minimum Detectable Concentration (MDC) of between 10% and 50% of the applicable DCGL. The use of 10% to 50% of the DCGL is an administrative limit only. Any value below the DCGL is acceptable in Class 1 or 2 survey units. MDCs of less than 50% of the DCGL allow detection of residual activity in Class 3 survey units at an investigation level of 0.5 times the DCGL. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the $DCGL_{EMC}$.

Instrument MDCs are discussed in Section 5.5.2 (d) and nominal MDC values are listed in Table 5-6. Instrumentation currently proposed for used in the final status survey is listed in Table 5-4. Maine Yankee follows instrument manufacturers recommendations and/or supporting basis documents for considerations such as temperature dependency.

As the project proceeds, other measurement instruments or technologies, such as in-situ gamma spectroscopy or continuous data collection scan devices, may be found to be more efficient than the survey instruments proposed in this plan. The acceptability of such an instrument or technology for use in the final survey program would be justified in a technical basis document. The technical basis document would include among other things the following: (1) a description of the conditions under which the method would be used; (2) a description of the measurement method, instrumentation and criteria; (3) justification that the technique would provide equivalent scan coverage for the given survey unit classification and that the scan MDC is adequate when compared to the $DCGL_{EMC}$; and (4) a demonstration that the method provides data that has a Type 1 error (falsely concluding that the survey unit is acceptable) equivalent to 5% or less and provides sufficient confidence that $DCGL_{EMC}$ criteria is satisfied.

b. Calibration And Maintenance

Instruments and detectors are calibrated for the radiation types and energies of interest at the site. The calibration sources for beta survey instruments are Tc-99, Cs-137, or Co-60 because the average beta energy (100 keV) approximates the beta energy of the radionuclides found on surfaces or in piping on site (85-94 keV). The alpha calibration sources when used are Am-241or Th-230 which have an appropriate alpha energy for plant-specific alpha emitting nuclides. Gamma scintillation detectors are calibrated using Cs-137, but the energy response to Co-60 has also been determined since discrete areas of Co-60 contamination have been found by soil surface scans.

Table 5-4
Final Status Survey Instruments

Measurement Type	Detector Type	Detector Total Area/ Density	Typical Manufacturer & Model #	Units
Surface Alpha/Beta-Gamma	Gas Flow Proportional	126 cm^2 0.8 mg/ cm^2	Ludlum 43-68	cpm
Surface Alpha/Beta-Gamma	Large Area Gas Flow Proportional	584 cm^2 821 cm^2 (both 0.8 mg/cm^2)	Ludlum 43-37 43-37-1	cpm
Surface Beta -Gamma	G-M	15.5 cm^2 2mg/ cm^2	LND, TGM Eberline SHP-360	cpm
Gamma Scan	NaI(Tl)	2"x2"	Eberline SPA-3	cpm
Liquid Beta	Scintillation	N/A	Beckman	µCi
Smear Beta-Gamma	Gas Proportional	15.5 cm^2 0.8 mg/ cm^2	Tennelec	dpm
Gamma Spectroscopy	HP Ge	N/A	Canberra	pCi

Instrumentation used for final status survey will be calibrated and maintained in accordance with the Instrumentation Program procedure. Radioactive sources used for calibration are traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. If vendor services are used, these will be obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality.

c. Response Checks

Instrumentation response checks are conducted to assure proper instrument response and operation. An acceptable response for field instrumentation is an instrument reading within +/- 10% of the established check source value. Laboratory instrumentation standards will be within +/- 3 sigma as documented on a control chart. Response checks are performed daily before instrument use and again at the end of use. Check sources contain the same type of radiation as that being measured in the field and are held in fixed-geometry jigs for reproducibility. If an instrument fails a response check, it is labeled "Do Not Use" and is removed from service until the problem is corrected in accordance with applicable procedures. Measurements made between the last acceptable check and the failed check are evaluated to determine if they should remain in the data set.

d. Minimum Detectable Concentration (MDC)

The MDC is determined for the instruments and techniques used for final status surveys (Table 5-6). The MDC is the concentration of radioactivity that an instrument can be expected to detect 95 percent of the time.

Static MDC For Structure Surfaces

For static (direct) surface measurements, with conventional detectors, such as those listed in Table 5-4, the MDC is calculated as follows:

$$\text{MDCstatic} = \frac{3 + 4.65\sqrt{B}}{(K)(t)}$$

where:

MDC_{static} = minimum detectable concentration for direct counting (dpm/100 cm^2),

B = background counts during the count interval t (counts),

t = count interval (for paired observations of sample and blank, usually 1 minute),

K = calibration constant (counts/min per dpm/100 cm^2),

The value of K includes correction factors for efficiency (ε_i and ε_s). The value of ε_s is dependent on the material type.

Corrections for radionuclide absorption have been made.

Open Land Area and Structure Scan MDC Using Alarm Set Point

The MDC formulae described in NUREG-1507 rely on the audible response of the meter. Maine Yankee proposes to use the E-600 instrument, a so called "smart meter," coupled to an appropriate detector for performing scan surveys for both structures and soil. This allows data logging and a more objective evaluation of scan MDC based on an alarm set point. The probability of alarm was calculated through simulation of instrument performance and compared to the $DCGL_{EMC}$, which was calculated using the area factors established in Sections 6.8 and 6.9. The extent of scan coverage is commensurate with the radiological conditions and classification of the survey unit in accordance with Table 5-3.

The determination of the alarm set-points and the DQO Type I error rate of 0.05 are based on using a 2 X 2 NaI detector moving at 0.25 m/sec at a distance of 2 inches from the soil surface. The error rate was calculated, and determined to be acceptable, using an E-600 instrument with a weighting factor of 5. The FSS procedures require a weighting factor of 5 to be applied during FSS scan surveys.

Prior to beginning the scan survey on an area, the local area background for a given survey unit or portion of a survey unit is determined. The FSS survey designer walks down the area and determines the number of potentially different background areas or materials. The designer then determines the number of measurements that need to be taken within the area in order to establish the local background. The technician collects the required number of measurements as well as soil samples and in situ gamma spectroscopy readings to ensure that the background values are not influenced by plant-derived radioactive materials. The average background reading is used to calculate the alarm set-point. This process ensures the appropriate determination and application of background characteristics in survey units with multiple media.

Before entering the survey unit grid[5] to begin a scan, the technician takes a one minute background count to ensure the background has not changed. If the background reading meets the expectation value, the technician performs the scan survey of the grid. The technician verifies the local area background is within plus or minus 1000 cpm of the expected value. If the background exceeds +/-1000 cpm or the instrument repeatedly alarms, the technician stops the survey and requests the FSS engineer to re-evaluate background and adjust the alarm set-point as necessary. Using the conversion factor derived in Maine Yankee's technical basis document (Reference 5.12.32), 1000 cpm is equivalent to about 2.2 pCi/g. Maine Yankee will add 2.2 pCi/g to the scan MDC for open land areas to account for the possibility that the background in a scan grid could decrease by up to 1000 cpm before the alarm set-point is readjusted.

The scan MDC's for open land areas using the E-600 instrument with an alarm set point are listed in Table 5-4a. The listed MDC's were selected to ensure a Type I error rate less than 0.05. The 0.05 Type I error rate is achieved by apportioning a 0.025 error rate to the first stage scan and a 0.025 error rate to the second stage scan.

5 The scan grid size is limited to no greater than 10 square meters so that background fluctuation is not a concern (Reference 5.12.36)

The MDC calculation and results are described in a Maine Yankee technical basis document (Reference 5.12.32). Maine Yankee will multiply the MDC by a factor of 1.15 which accounts for uncertainty due to variability in scan speed and detector distance from the soil surface. The *a priori* $DCGL_{EMC}$ used for survey planning for soil survey units will be based on the scan MDC associated with a 2 m^2 land area at a 0.025 Type 1 error rate, corrected by a factor of 1.15 to account for variable scan speed and distance and increased by 2.2 pCi/g, i.e., 5.9 pCi/g Cs-137. Table 5-4a lists the DCGL for areas outside the RA. The DCGL for areas inside the RA is 2.39 pCi/g Cs-137. (See Section 6.7.2 for the determination of the DCGL and application of surrogates.)

The survey is performed in the peak trap mode and the highest value obtained in the survey grid is logged.

The beta-gamma scan MDC for structures using the E-600 instrument with an alarm set point are listed in Table 5-4b. The listed MDC's were selected to ensure a Type 1 error rate less than 0.05. The MDC calculation and results are described in a Maine Yankee technical basis document (Ref. 5.12.32). Survey planning for structure survey units will be based on the scan MDC associated with a 0.5 m^2 surface area, i.e., 1832 dpm/100 cm^2 for a 600 c/m background. Table 5-4b lists the DCGL for areas of 600 and 2000 c/m background. The DCGL for structures is 18,000 dpm/100 cm^2. (See Section 6.7.2 for the determination of the DCGL and the application of surrogates). The gamma scan MDC's for concrete structures using the E-600 instrument with an alarm set point are described using the Maine Yankee gamma scan technical basis document (Ref. 5.12.34).

The structure beta-gamma scan survey is performed using a gas flow proportional detector moving at 5 cm/sec at a distance of 1.0 cm from the structure surface. The survey is performed in the peak trap mode with the highest value obtained in the survey grid logged. The concrete structure gamma scan may be performed using sodium iodide detectors when surface conditions would result in increased surface to detector distance (typically within 3 inches) and when the static measurement sample size is adjusted for the corresponding MDC, if necessary.

Table 5-4a Land Area Scan MDC for E-600 Instrument (Outside Restricted Area - DCGL = 4.2 pCi/g)*								
	Scan Area (m^2)							
	0.5	1	2	4	6	8	16	25
Area Factor	22.3	12.0	6.8	4.1	3.2	2.8	2.0	1.7
$DCGL_{EMC}$	93.7	50.4	28.6	17.2	13.4	11.8	8.4	7.1
MDC_{5s} (pCi/g) Type 1 = 0.05	4.5	3.6	3.2	3.0	2.5	2.5	2.5	2.0

*See Section 6.7 for explanation of DCGL calculated for areas outside the Restricted Area

Table 5-4b Structure Beta-Gamma Scan MDC for E-600 Instrument						
	Scan Area (m^2)					
	0.03	0.06	0.10	0.20	0.50	1.00
Area Factor	1667	847	500	250	100	50
MDC (with 600 c/m bkg) dpm/100 cm^2	4884	3663	3053	2442	1832	1221
MDC (with 2000 c/m bkg) dpm/100 cm^2	9157	6720	5490	4270	3660	3053

e. Detection Sensitivity

The nominal detection sensitivity of some of the detectors that may be used for surface contamination surveys has been determined and is provided in Table 5-6.

Count times are instrument-specific and are selected to ensure that the measurements are sufficiently sensitive for the DCGL. For example, the count times associated with surface activity surveys (1 minute) and gamma spectroscopy of volumetric materials (17 minutes) are administratively established to achieve MDCs less than the DCGL. The MDC_{scan} values are also below the DCGL shown in Table 5-6. The MDC_{scan} values may not always be less than the $DCGL_W$, but will be less than $DCGL_{EMC}$.

A technique for performing land scans with a SPA-3 detector coupled to the E-600 has been developed which is capable of detecting discrete Co-60 particles of 1 uCi activity buried at a depth of six inches in soil. This capability has been confirmed by actual field testing using this detector with the E600, as documented in a technical basis document (Reference 5.12.32).

Cs-137 sensitivity was determined to be 3 pCi/g Cs-137 in a $2m^2$ area. This is based on modeling the SPA-3/E600 combination, as documented in Reference 5.12.32 and confirmed by field testing.

The E600 instrument will be operated in the single channel analyzer mode when used in scan surveys to optimize the instrument's energy spectrum sensitivity.

f. Total Efficiency (E_t) and Source Efficiency (E_s) for Concrete Contamination

Section 6.6 provides a detailed description of the dose assessment for contaminated basement concrete. The source term input to the groundwater calculations is the total inventory within the basement concrete. This inventory appears to be primarily located within the first mm of the concrete surface. Various fixed point measurement alternatives for determining the source term were evaluated including gross beta measurements on the surfaces, volumetric concrete sampling and in-situ gamma spectroscopy. Gross beta fixed point measurements were determined to be cost-effective and technically defensible under the assumption that the instrument efficiencies for concrete could be satisfactorily calculated using the methods recommended in NUREG-1507.

For scan surveys, gross beta measurements appear to be the only practical method. Under certain conditions, in-situ gamma spectroscopy may be a reasonable method for replacing beta scan surveys. If in-situ gamma spectroscopy is used, a technical basis document will be developed demonstrating its suitability for final survey measurements and NRC will be notified 30 days prior to its first use.

The methods for determining efficiency in NUREG-1507 were specifically developed to address situations when the source, in this case concrete, affects radiation emission rate due to self-attenuation, backscatter, thin coverings, etc. This method accounts for these source effects by separating the efficiency calculation into two components, i.e., instrument efficiency E_i

and source efficiency E_s. The total efficiency E_t, is the product of E_i and E_s as shown below.

$$E_t = (E_i)(E_s)$$

The E_i was determined by calibration to a NIST traceable, large area Tc-99 source. The E_s value was determined empirically through measurements of concrete cores collected from representative site locations. The empirically derived value of 0.35 compares reasonably with the ISO standard default values of 0.25 for betas less than 0.4 MeV and 0.5 for betas greater than 0.4MeV, considering most of the concrete activity is Cs-137 with a beta energy greater than 0.4. Forty three cores were obtained from concrete floors of the buildings known to be contaminated. Cores were collected from the Containment Building loop areas which were considered to represent reactor coolant contamination. Spray Building cores were representative of the ECCS (emergency core cooling system) contamination. Cores collected in the PAB were representative of the waste processing system contamination. The RCA Building cores represented waste systems and decontamination activities. Fuel Building cores represented the spent fuel pool contamination events. Several cores were taken from each building. The core nuclide activities were determined by gamma spectrometry, geometry corrected, then the pCi/g result was multiplied by the mass of the core sample and converted to total gross beta dpm.

The cores were moved to a low background area and counted for gross beta using final survey instrumentation. The cores were initially counted for 1 minute, corrected for background and reported as net cpm. The instrument total efficiency, E_t, was calculated as the ratio of the net count rate divided by the net activity in dpm. The initial efficiency data resulted in a mean efficiency of 0.148 with a standard deviation of 0.11. The data showed wide variability with approximately 50% of the individual efficiency values within one standard deviation of the mean. (Tchebycheff's theorem states that 68% of the values of a normally distributed population should be within one standard deviation of the mean.)

The core efficiency data have undergone a re-evaluation since the data were first obtained in order to better understand the wide variation exhibited by the initial data. New cores were collected to replace those previously destroyed during analysis. The cores still remaining were recounted. Five minute count times were used since some of the cores did not have high activity levels. Shielded and unshielded measurements were taken of each core to allow a more accurate background correction for each core. The recounted, reevaluated core data gave a mean total efficiency of 0.130 and a standard deviation of 0.06. The individual, recounted core efficiency values ranged from a high of 0.25 to a low of less than 0.01. Almost 70% of the efficiency measurements were within one sigma of the mean.

The cores were collected from many areas of the plant as described above. Upon physical examination of the cores it was noted that some cores consisted of bare concrete, some had been painted and the paint surface was well worn, some retained a thin coat of paint, and some had been painted with a thick coat of easy-to-decontaminate paint with coatings as thick as 3/32 of an inch. It appears that most of the very low efficiency values came from cores taken in areas where floors were coated with the thick, easy to decontaminate paint. Applying the paint attenuation equation given in NUREG-1507, the thick floor coating would shield the beta particles to the point of almost no detector response. These cores represent areas (RCA floor, Spray Bldg. floor, and Decon Room floor) that will not be amenable to direct measurement by gas-filled detector unless paint is removed. These areas will be surveyed by volumetric sample or in-situ gamma spectroscopy (if justified in technical basis document), or the surface will be remediated before survey. These samples have been removed from the core population in the final E_t calculation.

The cores with the high efficiencies were evaluated to determine if the presence of high levels of naturally occurring beta particles in the concrete mixture may be contributing to the high values. The background correction that was performed on these samples was for area background, not material background. Material background did not contribute significantly to the sample activity.

The use of gross beta counting is a reasonable, cost effective method for measuring concrete contamination. This technique can also be conservatively applied to activity measurements of the Containment wall liner because the liner is a smooth, nearly flat surface. The alternatives to gross counting (e.g., volumetric sampling with gamma spectrum analysis or in-situ gamma spectroscopy), while admittedly more costly and time consuming survey methods, are viable alternatives. Such measures may be applied to areas with thick floor coatings or very irregular surfaces resulting from remediation activities if an acceptable efficiency correction factor cannot be determined.

The table below lists the instrument, source and total efficiencies for the instruments proposed for material scan and direct measurements.

Table 5-5
Survey Instrument Efficiencies
(Material Scan and Direct Measurement Instruments)

Detector	Source Efficiency (E_s)	Total Efficiency (E_t)	Instrument Efficiency (E_i)
Ludlum 43-68	0.389	0.13	0.333
SHP-360	0.225	0.060	0.280

g. Pipe Survey Instrumentation

Remaining pipe will be surveyed to ensure residual remaining activity is less than the DCGL. Pipe crawlers (survey instruments) proposed for use for surveys of pipe with diameters between 1.5 and 12 inches have been shown to have 4π efficiencies ranging from 0.005 to 0.295 respectively. This equates to detection sensitivities of 2800 dpm/100cm^2 to 210 dpm/100cm^2 respectively. This level of sensitivity is adequate to detect residual activity below the BOP embedded pipe DCGL of 100,000 dpm/100cm^2 (800,000 dpm/100cm^2 for spray pipe DCGL) or the buried pipe DCGL of 9,800 dpm/100cm^2.

The Pipe Explorer^{TM} has been selected to survey the embedded Spray Building pipe. The Pipe Explorer^{TM} system has been used for alpha, beta, gamma and video surveys of over 6,000 feet of piping. The surveys have included pipes with up to 8 elbows and with vertical runs in excess of 9 m. Detectors have been successfully deployed past rocks, oil, and other debris that have obstructed up to 50 percent of the pipe's cross sectional area. The Pipe Explorer^{TM} deployment system is capable of conducting surveys in pipes with diameters ranging from 0.05 m to 1.22 m and survey lengths that vary from 30 m up to 300 m. The detectors are protected and propelled by a pneumatically-driven tubular membrane.

The MDA for the 16 inch spray pipe for example is based on Type 1 and 2 errors of 0.05 and is calculated using the Currie (1968) formula as follows:

$$\text{MDA} = \frac{2.71 + 4.65\sqrt{(BKR)(t)}}{(CF)(t)}$$ where MDA is in dpm/100 cm2, BKR is the Background Count Rate (cpm), CF is the Conversion Factor in net cpm/dpm/100 cm^2 and t is the count time in minutes. For a background count rate of 4194 counts per minute and a CF of 6.4E-2 cpm/dpm/100cm^2, an MDA for Cs-137 of 4745 dpm/100cm^2 was calculated.

Table 5-6
Measurement Detection Sensitivities**

Type of Measurement	Detector	Background*	E*** (c/d)	MDC	DCGL
Beta-Gamma Surface Scan	Pancake G-M (SHP-360)	40 cpm	0.06	10484 dpm/100 cm^2	18000 dpm/100 cm^2
Beta-Gamma Surface Scan	Ludlum 43-68 126 cm^2 Gas Proportional	600 cpm	0.13	1832 dpm/100 cm^2	18000 dpm/100 cm^2
Beta-Gamma Juncture Scan	Ludlum 43-68 126 cm^2 Gas Proportional	600 cpm	0.06	3969 dpm/100 cm^2	18000 dpm/100 cm^2
Beta-Gamma Direct	Pancake G-M (SHP-360)	40 cpm	0.06	3554 dpm/100 cm^2	18000 dpm/100 cm^2
Beta-Gamma Direct	Ludlum 43-68 126 cm^2 Gas Proportional	600 cpm	0.13	714 dpm/100 cm^2	18000 dpm/100 cm^2
Beta-Gamma Direct	Ludlum 43-37 582 cm^2 Gas Proportional	2000 cpm	0.141	257 dpm/100 cm^2	18000 dpm/100 cm^2
Beta-Gamma Surface Scan	Ludlum 43-37 582 cm^2 Gas Proportional	2000 cpm	0.141	3585 dpm/100 cm^2	18000 dpm/100 cm^2

Table 5-6 Measurement Detection Sensitivities**					
Type of Measurement	**Detector**	**Background***	**E*** (c/d)**	**MDC**	**DCGL**
Beta-Gamma Direct	Ludlum 43-94 39 cm^2 Gas Proportional	75 cpm	0.024 (for 3" pipe) 0.031 (for 2" pipe) 0.036 (for 1" pipe)	4305 dpm/100 cm^2 (for Eff. of 0.024)	100,000 dpm/100 cm^2
Alpha Direct	Ludlum 43-68 126 cm^2 Gas Proportional	1 cpm	0.20	30 dpm/100 cm^2	Beta-Gamma Direct
Gamma Scan (Soil)	NaI(Tl) (SPA-3)	10,000 cpm	0.012	5.9 pCi/g (Cs-137)	2.39 pCi/g (Inside RA) 4.2 pCi/g (Outside RA) (Cs equiv.)
Gamma Scan (Concrete)	NaI(Tl) (SPA-3)	20,000 cpm	TBD	See Ref. 5.12.34	18000 dpm/100 cm^2
Gamma Spectroscopy	HP Ge	N/A	N/A	0.01 pCi/g	2.39 pCi/g (Inside RA) 4.2 pCi/g (Outside RA) (Cs equiv.)
Liquid Beta	Beckman Liquid Scintillation	40 dpm	0.46	3.25E-6 uCi/ml	N/A
Smear Alpha / Beta-Gamma	Tennelec Gas Proportional	0.5 cpm Alpha 30 cpm Beta-gamma	0.25 Alpha 0.35 Beta	25 dpm - alpha 81 dpm - beta-gamma	N/A

*Background values are typical values. These background values are well below the MDCs and are adequate for selecting the instruments for performing surveys. **The table values are based on a one minute direct count or a surface scan rate of 2 inches per second, and a soil scan rate of 20 sec/m^2, unless otherwise noted. *** Efficiencies for concrete surfaces are E_t. E_i, adjusted for geometry effects, is used for pipe survey efficiency.

5.6 Investigation Levels and Elevated Areas Test

During survey unit measurements, levels of radioactivity may be identified by an increase in count rate, an instrument alarm or an elevated sample result that warrant investigation. Elevated measurements may result from either discrete particles, a distributed source, or a change in background activity. In either case the investigations actions would be followed. Depending on the results of the investigation, the survey unit may require no action, may require remediation, and/or may require reclassification and resurvey. Investigation levels and the investigation process are described below.

5.6.1 Investigation Levels

NUREG 1727 (Table E.2) and NUREG 1575 (Table 5.8) provide investigation levels for scan surveys. In addition to investigation levels for scan surveys, direct measurement survey investigation levels have also been developed. These additional investigation levels include a very conservative value for Class 3 survey units as shown in Table 5-7.

5.6.2 Investigation Process

Technicians will respond to all instrument alarms while surveying. Upon receiving an alarm, the technician will stop and resurvey the last square meter of area to verify the alarm. Technicians are cautioned, in training, about the importance of the alarm verification survey, instructed on expected instrument response to localized areas of elevated activity and are given specific direction in procedure as to survey extent and scan speed. If the alarm is verified, the technician will mark the area with a flag or other appropriate means. The alarm data may be evaluated by the FSSS with respect to the investigation levels specified in Table 5-7. Each area marked, which exceeds the investigation level specified in Table 5-7, will have an investigation survey instruction prepared. The instruction will require a re-scan of the area, direct measurements, field gamma spectroscopy measurement (as appropriate), and collection of a soil sample (for land surveys). Each investigation will be evaluated and reported in the survey unit Release Record.

The size and average activity level in the elevated area is determined to demonstrate compliance with the area factors. If any location in a Class 2 area exceeds the DCGL, scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s). If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area should be reclassified.

Table 5-7 Investigation Levels		
Classification	**Scan Investigation Levels**[6]	**Direct Investigation Levels**
Class 1	$>DCGL_{EMC}$	$>DCGL_{EMC}$
Class 2	$>DCGL_w$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_w$.	$>DCGL_w$
Class 3	$>DCGL_w$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_w$.	$>0.5\ DCGL_w$

Investigations should consider: (1) the assumptions made in the survey unit classification; (2) the most likely or known cause of the contamination; and (3) the possibility that other areas within the survey unit may have elevated areas of activity that may have gone undetected. Depending on the results of the investigation, a portion of the survey unit may be reclassified if there is sufficient justification. The results of the investigation process are documented in the survey area Release Record. See also Section 5.6.4 for additional discussion regarding potential reclassification of the survey unit.

5.6.3 Elevated Measurement Comparison (EMC)

The elevated measurement comparison may be used for Class 1 survey units when one or more scan or static measurements exceed the investigation level if remediation is not performed. The EMC provides assurance that unusually large measurements receive the proper attention and that any area having the potential for significant dose contribution is identified. As stated in NUREG-1575, the EMC is intended to flag potential failures in the remediation process and should not be considered the primary means to identify whether or not a survey unit meets the release criterion.

Locations identified by scan with levels of residual radioactivity which exceed the *a priori* $DCGL_{EMC}$ or static measurements with levels of residual radioactivity which exceed the *a priori* $DCGL_{EMC}$ are subject to additional surveys to determine compliance with the elevated measurement criteria. The size of the area containing the elevated residual radioactivity and the average level of residual activity within the area are determined. The average level of activity is compared to the $DCGL_W$ based on the actual area of elevated

6 Must be calculated *a priori*. The a priori $DCGL_{EMC}$ for soil was calculated to be 5.9 pCi/g in accordance with Section 5.5.6.d.

activity. (If a background reference area is being applied to the survey unit, the mean of the background reference area activity may be subtracted before conducting the EMC).

The *a priori* $DCGL_{EMC}$ is established during the survey design and is calculated as follows:

$$DCGL_{EMC} = \text{Area Factor x DCGL}$$

The area factor is the multiple of the DCGL that is permitted in the area of elevated residual radioactivity without remediation. The area factor is related to the size of the area over which the elevated activity is distributed. That area is generally bordered by levels of residual radioactivity below the DCGL and is determined by the investigation process. Area factors are calculated in Section 6 of the LTP and listed in Tables 6-12 and 6-14.

The actual area of elevated activity is determined by investigation surveys and the area factor is adjusted for the actual area of elevated activity. The product of the adjusted area factor and the DCGLw determines the actual $DCGL_{EMC}$. If the $DCGL_{EMC}$ is exceeded, the area is remediated and resurveyed.

The results of the elevated area investigations in a given survey unit that are below the $DCGL_{EMC}$ limit are evaluated using the equation below. If more than one elevated area is identified in a given survey unit, the unity rule can be used to determine compliance. If the formula value is less than unity, no further elevated area testing is required and the EMC test is satisfied.

$$\frac{\delta}{DCGL_w} + \frac{(\text{average concentration in elevated area} - \delta}{(\text{Area Factor})(DCGL_w)} < 1$$

Where: δ is the average residual activity in the survey unit. When calculating δ for use in this inequality, measurements falling within the elevated area may be excluded provided the overall average in the survey unit is less than the $DCGL_w$.[7] For contaminated concrete (basement fill model), the area factor used in the unity rule may be specified as the survey unit size divided by the elevated area size.

Compliance with the soil $DCGL_{EMC}$ will be determined using the FSS gamma spectroscopy results and a unity rule approach. These general methods will also be applied to other materials where sample gamma spectroscopy is used for FSS. The application of the unity rule to the elevated measurement comparison requires area factors and corresponding

7. MARSSIM, NUREG-1575, Revision 1, (June 2001), Section 8.5.2, per the EPA website at www.epa.gov/radiation/marssim/docs/revision1.

$DCGL_{EMC}$'s to be calculated for Cs-137, Co-60, and any other gamma emitter identified during FSS, separately.

The methods used to calculate the nuclide specific soil area factors will be the same as described in Section 6.8.2. These area factors are used to determine $DCGL_{EMC}$ for Co-60, Cs-137, and any other identified gamma emitter, for each elevated area being evaluated during FSS. The surrogate radionuclides will be conservatively accounted for through the application of the Cs-137 area factor to the surrogate Cs-137 DCGL since the HTD radionuclides have higher area factors than Cs-137. The $DCGL_{EMC}$'s are used as follows to determine compliance with the elevated measurement comparison. Background could be subtracted from each radionuclide concentration if necessary.

$$\left(\frac{Cs-137}{Cs-137_{DCGL_{EMC}}}\right)+\left(\frac{Co-60}{Co-60_{DCGL_{EMC}}}\right)+\ldots+\left(\frac{R_N}{DCGL_{EMC_N}}\right)\leq 1.0$$

Where: Cs-137 and Co-60 are the gamma spec results from FSS,

$DCGL_{EMC_N}$ is calculated for the size of the elevated area being evaluated,

R_N is any other gamma emitter identified during FSS, and

$DCGL_{EMC_N}$ is the $DCGL_{EMC}$ for radionuclide N

5.6.4 Remediation and Reclassification

As shown in Table 5-8, for any classification (1, 2 or 3), areas of elevated residual activity above the $DCGL_{EMC}$ are remediated to reduce the residual radioactivity to acceptable levels. Whenever an investigation confirms activity above an action level listed in Table 5-8, an evaluation of the HSA, operational history, design information, and sample results will be performed. The evaluation will consider: (1) the elevated area's location, dimensions, and sample results, (2) an explanation as to the potential cause and extent of the elevated area in the survey unit, (3) the recommended extent of reclassification, if considered appropriate, and (4) any other required actions. Areas that are reclassified as Class 1 are typically bounded by a Class 2 buffer zone to provide further assurance that the reclassified area completely bounds the elevated area. This evaluation process is established to avoid the unwarranted reclassification of an entire survey unit (which can be quite large) while at the same time requiring an assessment as to extent and reasons for the elevated area.

Specifically, for the reclassification (following LTP approval) of a survey unit (or portion of a survey unit) from Class 1 to Class 2, the following criteria will be followed:

1. The survey unit (or portion of a survey unit) to be reclassified as Class 2 must meet the Class 2 designation (LTP Section 5.2.2), i.e., prior to remediation, the reclassified area is not likely to contain residual

radioactivity in excess of the $DCGL_W$.

2. There is sufficient knowledge regarding the distribution of contamination within the reclassified Class 2 area to support a conclusion that subject area is not likely to contain residual radioactivity in excess of the $DCGL_W$.

3. As noted in Table 5-3, for Class 2 Survey Units, the amount of scan coverage will be proportional to the potential for finding areas of elevated activity or areas close to the release criterion in accordance with MARSSIM Section 5.5.3.

Reclassification from either Class 1 or Class 2 to Class 3 would generally observe similar criteria as listed above.

1. The reclassified survey unit (or portions thereof) would be required to meet Class 3 requirements (per Section 5.2.2).
2. There is sufficient knowledge regarding the distribution of contamination within the reclassified Class 3 area to support a conclusion that the area has a low probability of containing residual radioactivity.
3. Scan coverage for the reclassified area will meet Table 5-3 requirements

Per agreements with NRC, Maine Yankee will provide notification to the NRC prior to a reclassification (following LTP approval) of a survey unit (or portion of a survey unit) per the discussion in Section 1.4.

If an individual survey measurement (scan or direct) in a Class 2 survey unit exceeds the DCGL, the survey unit or a portion of it may be reclassified and the survey redesigned and re-performed accordingly. If an individual survey measurement in a Class 3 survey unit exceeds 0.5 DCGL, the survey unit, or portion of a survey unit, will be evaluated, and if necessary, reclassified to a Class 2 and the survey redesigned and re-performed accordingly.

Table 5-8 Investigation Actions			
Action If Investigation Results Exceed:			
Class	**$DCGL_{EMC}$**	**$DCGL_W$**	**0.5 $DCGL_W$**
1	Remediate and resurvey as necessary	Acceptable	Acceptable
2	Remediate, reclassify portions as necessary	Reclassify portions as necessary	Acceptable
3	Remediate, reclassify portions as necessary	Increase scan coverage and reclassify portions as necessary	Increase scan coverage and reclassify portions as necessary

5.6.5 Resurvey

Following an investigation, if a survey unit is reclassified or if remediation activities were performed, a resurvey is performed in accordance with procedures. If a Class 2 area had contamination greater than the $DCGL_W$ it should be reclassified. If the average value of Class 2 direct survey measurements was less than the $DCGL_W$, the $Scan_{MDC}$ was sensitive enough to detect the $DCGL_{EMC}$ and there were no areas greater than the $DCGL_{EMC}$, the survey redesign may be limited to obtaining a 100% scan without having to re-perform the direct measurements. This condition assumes that the sample density meets the requirements for a Class 1 area. If the Class 2 area had contamination greater than the $DCGL_W$, but the $Scan_{MDC}$ was not sensitive enough to detect the $DCGL_{EMC}$, the affected area is reclassified and resurveyed at the sample density determined from the EMC.

5.7 Data Collection and Processing

5.7.1 Sample Handling and Record Keeping

A sample tracking record (chain-of-custody record) accompanies each sample from the point of collection through obtaining the final results to ensure the validity of the sample data. Sample tracking records are controlled and maintained and, upon completion of the data cycle, are transferred to Document Control, in accordance with applicable procedures.

Each survey unit has a document package associated with it which covers the design and field implementation of the survey requirements. Survey unit records are quality records.

5.7.2 Data Management

Survey data are collected from several sources during the data life cycle and are evaluated.

QC replicate measurements are not used as final status survey data. See LTP Section 5.10.4(d) for design and use of QC replicate measurements.

Measurements performed during turnover and investigation surveys can be used as final status survey data if they were performed according to the same requirements as the final survey data. These requirement include: (1) the representativeness of the survey data to reflect the as-left survey unit condition untouched by further remediation; (2) the application of isolation measures to the survey unit to prevent re-contamination and to maintain final configuration; and (3) the data collection and design were in accordance with FSS methods, e.g., scan MDC, investigation levels, survey data point number and location, statistical tests, and EMC tests.

Measurement results stored as final status survey data constitute the final survey of record and are included in the data set for each survey unit used for determining compliance with the site release criteria.

Measurements are recorded in units appropriate for comparison to the DCGL. The recording units for surface contamination are dpm/100 cm^2 and pCi/g for activity concentrations. Numerical values, even negative numbers, are recorded.

Document Control procedures establish requirements for record keeping. Measurement records include, at a minimum, the surveyor's name, the location of the measurement, the instrument used, measurement results, the date and time of the measurement and any surveyor comments.

5.7.3 Data Verification and Validation

The final status survey data are reviewed before data assessment to ensure that they are complete, fully documented and technically acceptable. The review criteria for data acceptability will include at a minimum, the following items:

a. The instrumentation MDC for fixed or volumetric measurements was below the $DCGL_W$ or if no, it was below the $DCGL_{EMC}$ for Class 1, below the $DCGL_W$ for Class 2 and below 0.5 $DCGL_W$ for Class 3 survey units.

b. The instrument calibration was current and traceable to NIST standards,

c. The field instruments were source checked with satisfactory results before and after use each day data were collected or data was evaluated by the FSSE if instruments did not pass a source check in accordance with 5.5.2.c

d. The MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey,

e. The survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed,

f. "Special methods" for data collection were properly applied for the survey unit under review. These special methods are either described in this LTP section or will be the subject of an NRC notice of opportunity for review,

g. The chain-of-custody was tracked from the point of sample collection to the point of obtaining results,

h. The data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility, and

i. The data have been properly recorded.

If the data review criteria were not met, the discrepancy will be reviewed and the decision to accept or reject the data will be documented in accordance with approved procedures.

5.7.4 Graphical Data Review

Survey data may be graphed to identify patterns, relationships or possible anomalies which might not be so apparent using other methods of review. A posting plot or a frequency plot may be made. Other special graphical representations of the data will be made as the need dictates.

a. Posting Plots

Posting plots may be used to identify spatial patterns in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal patches of elevated radioactivity or local areas in which the DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Incongruities in the background data may be the result of residual, undetected activity, or they may just reflect background variability.

b. Frequency Plots

Frequency plots may be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity. Multiple peaks in the data may indicate the presence of isolated areas of residual radioactivity or background variability due to soil types or differing materials of construction. Variability may also indicate the need to more carefully match background reference areas to survey units or to subdivide the survey unit by material or soil type.

5.8 Data Assessment and Compliance

An assessment is performed on the final status survey data to ensure that they are adequate to support the determination to release the survey unit. Simple assessment methods such as comparing the survey data to the DCGL or comparing the mean value to the DCGL are first performed. The statistical tests are then applied to the final data set and conclusions are made as to whether the survey unit meets the site release criterion.

5.8.1 Data Assessment Including Statistical Analysis

The results of the survey measurements are evaluated to determine whether the survey unit meets the release criterion. In some cases, the determination can be made without performing complex, statistical analyses.

a. Interpretation of Sample Measurement Results

An assessment of the measurement results is used to quickly determine whether the survey unit passes or fails the release criterion or whether one of the statistical analyses must be performed. The evaluation matrices are presented in Tables 5-9 and 5-10.

Table 5-9 Interpretation of Sample Measurements When WRS Test Is Used	
Measurement Results	**Conclusion**
Difference between maximum survey unit concentration and minimum reference area concentration is less than DCGLw	Survey unit meets release criterion.
Difference of survey unit average concentration and reference average concentrations greater than DCGLw	Survey unit fails.
Difference between any survey unit concentration and any reference area concentration is greater than DCGLw and the difference of survey unit average concentration and reference area average concentration is less than DCGLw	Conduct WRS test and elevated measurements test.

Table 5-10 Interpretation of Sample Measurements When Sign Test is Used	
Measurement Results	**Conclusion**
All concentrations less than DCGLw	Survey unit meets release criterion
Average concentration greater than DCGLw	Survey unit fails
Any concentration greater than DCGLw and average concentration less than DCGLw	Conduct Sign Test and elevated measurements test.

When required, one of four statistical tests will be performed on the survey data:

1. WRS Test
2. Sign Test
3. WRS Test Unity Rule
4. Sign Test Unity Rule

In addition, survey data are evaluated against the EMC criteria as previously described in Section 5.6.3 and as required by NUREG 1727. The statistical test is based on the null hypothesis (Ho) that the residual radioactivity in the survey unit exceeds the DCGL. There must be sufficient survey data at or below the DCGL to reject the null hypothesis and conclude the survey unit meets the site release criterion for dose. Statistical analyses are performed using a specially designed software package or, if necessary, using hand calculations.

b. Wilcoxon Rank Sum Test

The WRS test, or WRS Unity Rule (NUREG-1505, Chapter 11), may be used when the radionuclide of concern is present in the background or measurements are used that are not radionuclide-specific. In addition, this test is valid only when "less than" measurement results do not exceed 40 percent of the data set.

The WRS test is applied as follows:

1. The background reference area measurements are adjusted by adding the $DCGL_W$ to each background reference area measurement, X_i, $Z_i = X_i + DCGL$.

2. The number of adjusted background reference area measurements, m, and the number of survey unit measurements, n, are summed to obtain N, (N = m + n).

3. The measurements are pooled and ranked in order of increasing size from 1 to N. If several measurements have the same value, they are assigned the average rank of that group of measurements.

4. The ranks of the adjusted background reference area measurements are summed to obtain W_r.

5. The value of W_r is compared with the critical value in Table I.4 of NUREG-1575. If W_r is greater than the critical value, the survey unit meets the site release dose criterion. If W_r is less than or equal to the critical value, the survey unit fails to meet the criterion.

c. Sign Test

The Sign test and Sign test Unity Rule are one-sample statistical tests used for situations in which the radionuclide of concern is not present in background, or is present at acceptable low fractions compared to the $DCGL_W$. If present in background, the gross measurement is assumed to be entirely from plant activities. This option is used when it can be reasonably expected that including the background concentration will not affect the outcome of the Sign test. The advantage of using the Sign test is that a background reference area is not needed. The Sign Test may also be used with background subtraction in accordance with Chapter 12 of NUREG-1505.

The Sign test is conducted as follows:

1. The survey unit measurements, X_i, i = 1, 2, 3, ...N; where N = the number of measurements, are listed.

2. X_i is subtracted from the $DCGL_W$ to obtain the difference $D_i = DCGL_W - X_i$, i = 1, 2, 3,..., N.

3. Differences where the value is exactly zero are discarded and N is reduced by the number of such zero measurements.

4. The number of positive differences are counted. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the site release criterion.

5. The value of S+ is compared to the critical value given in Table I.3 of NUREG-1575. The table contains critical values for given values of N and α. The value of α is set at 0.05 during survey design. If S+ is greater than the critical value given in the table, the survey unit meets the site release criterion. If S+ is less than or equal to the critical value, the survey,unit fails to meet the release criterion.

d. Unity Rule

The Cs-137 to C0-60 ratio will vary in the final survey soil samples, and this will be accounted for using a "unity rule" approach as described in NUREG-1505 Chapter 11. Unity Rule Equivalents will be calculated for each measurement result using the surrogate adjusted Cs-137 DCGL and the adjusted Co-60 DCGL, as shown in the following equation. (See Section 6.7.2 for the $Cs\text{-}137_S$ DCGL calculation.)

$$\text{Unity Rule Equivalent} \leq 1 = \frac{\text{Cs-137}}{\text{DCGL}_{(\text{Cs-137}_S)}} + \frac{\text{Co}-60}{\text{DCGL}_{(\text{Co}-60_A)}} + \ldots + \frac{R_N}{\text{DCGL}_{(N_A)}}$$

Where: Cs-137 and Co-60 are the gamma spec results,

$DCGL_{(Cs-137_S)}$ is the surrogate Cs-137s DCGL, adjusted to represent the Table 6-11 total surface dose, as applicable (inside RA)

$DCGL_{(Co-60_A)}$ is the Co-60 DCGL, adjusted to represent the Table 6-11 total surface dose, as applicable (inside RA)

R_N is any other identified gamma emitting radionuclides, and $DCGL_{(N_A)}$ is the adjusted DCGL for radionuclide N.

The unity rule equivalent results will be used to demonstrate compliance assuming the DCGL is equal to 1.0 using the criteria listed in the LTP, Tables 5-9 and 5-10. If the application of the WRS or Sign test is necessary, these tests will be applied using the unity rule equivalent results and assuming that the DCGL is equal to 1.0. An example of a WRS test using the unity rule is provided in NUREG-1505, Page 11-3, Section 11.4. If the WRS test is used, or background subtraction is used in conjunction with the Sign test, background concentrations will also be converted to Unity Rule Equivalents prior to performing test.

The Sign test will be used without background subtraction if background Cs-137 is not considered a significant fraction of the DCGL. Note that the surrogate Cs-137 DCGL will be used for both the statistical tests and comparisons with the criteria in LTP Tables 5-9 and 5-10.

The same general surrogate and unity rule methods described above for soil will be applied to other materials, such as activated concrete, where sample gamma spectroscopy is used for final survey as opposed to gross beta measurements.

5.8.2 Data Conclusions

The results of the statistical tests, including application of the EMC, allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release dose criterion. The data provide statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the release criterion. The decision to release the survey unit is made with sufficient confidence and without further analysis.

The second conclusion that can be made is that the survey unit fails to meet the release criterion. The data are not conclusive in showing that the residual radioactivity is less than the release criterion. The data are analyzed further to determine the reason for the failure.

Possible reasons are that:

1. the average residual radioactivity exceeds the DCGL, or

2. the test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

The power of the statistical test is a function of the number of measurements made and the standard deviation in measurement data. The power is determined from $1-\beta$ where β is the value for Type II errors. A retrospective power analysis may be performed using the methods described in Appendices I.9 and I.10 of NUREG-1575. If the power of the test is insufficient due to the number of measurements, additional samples may be collected as directed by procedure. A greater number of measurements increases the probability of passing if the survey unit actually meets the release criterion. If failure was due to the presence of residual radioactivity in excess of the release criterion, the survey unit must be remediated and resurveyed.

5.8.3 Compliance

The final status survey is designed to demonstrate that licensed radioactive materials have been removed from MY station facilities and property to the extent that residual levels of radioactive contamination are below the radiological criteria for unrestricted use as approved by the NRC. The site-specific radiological criteria presented in this plan demonstrate compliance with the criteria of 10CFR20.1402 and State of Maine Law LD 2688-SP1084.

If the measurement results pass the requirements of Tables 5-9 and 5-10 of Section 5.8.1, and the elevated areas evaluated per Section 5.6.3 pass the elevated measurement comparison, then the survey unit is suitable for unrestricted release.

5.9 Reporting Format

Survey results are documented in history files, survey unit release records, and in the final status survey report. Other reports may be generated as requested by the NRC.

5.9.1 History File

A history file of relevant operational and decommissioning data has been compiled. The history file consists of the HSA, GTS Characterization Report, Classification Basis, and 50.75(g) file information. The purpose of the history file is to provide a substantive basis for the survey unit classification, and hence, the level of intensity of the final status survey. The history file contains:

1. Operating history which could affect radiological status

2. Summarized scoping and site characterization data

3. Other relevant information

5.9.2 Survey Unit Release Record

A separate release record is prepared for each survey unit. The survey unit release record is a document containing sufficient information necessary to demonstrate compliance with the site release criteria. This record includes at least:

a. Description of the survey unit

b. Survey unit design information

c. Survey results

d. Survey unit investigations performed and their results

e. Survey unit data assessment results

When a survey unit release record is given final approval it becomes a quality record.

5.9.3 Final Status Survey Report

Survey results will be described in a written report to the NRC. The actual structures, land, or piping system included in each written report may vary depending on the status of ongoing decommissioning activities.

The final status survey report provides a summary of the survey results and the overall conclusions which demonstrate that the MY facility and site meet the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results are included in the report. The level of detail is sufficient to clearly describe the final status survey program and to certify the results. The format of the final report will contain the following topics:

1.0 Overview of the Results

2.0 Discussion of Changes to FSS

3.0 Final Status Survey Methodology
- Survey unit sample size
- Justification for sample size

4.0 Final Status Survey Results
- Number of measurements taken
- Survey maps
- Sample concentrations
- Statistical evaluations, including power curves
- Judgmental and miscellaneous data sets
- Investigations and results (anomalous data)

5.0 Conclusion for each survey unit

- Any Changes from initial assumptions on extent of residual activity.

- Simplified General Retrospective Dose Estimate: For illustrative purposes, relevant FSS data will be reviewed to determine a gross average of residual contamination level which will be used to calculate a retrospective dose estimate. This retrospective dose estimate, which will be provided in the final report, may be helpful in illustrating to various stakeholders Maine Yankee's compliance with the dose based release criteria.

5.9.4 Other Reports

Other reports will be prepared and submitted as requested.

5.10 FSS Quality Assurance Plan (QAP)

The Final Status Survey QAP, as described in this section, is developed and implemented by trained and qualified personnel. The FSS QAP will ensure that the site will be surveyed, evaluated and determined to be acceptable for unrestricted use if the residual activity results in an annual TEDE to the average member of the critical group of 10 mrem/year or less for all pathways and 4 mrem or less for groundwater drinking sources (enhanced state clean-up levels). Ensuring that the site meets the requirements for license termination is a complex process. Quality must be built in to each phase of the plan and measures must be taken during the execution of the plan to determine whether the expected level of quality is being achieved.

The Quality Assurance activities for decommissioning are based on the requirements of 10CFR50.82. The objective of the FSS QAP is to ensure that the survey data collected are of the type and quality needed to demonstrate with sufficient confidence that the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. The plan ensures that: 1) the elements of the final status survey plan are implemented in accordance with the approved procedures; 2) surveys are conducted by trained personnel using calibrated instrumentation; 3) the quality of the data collected is adequate; 4) all phases of package design and survey are properly reviewed, and oversight is provided; and 5) corrective actions, when identified, are implemented in a timely manner and are determined to be effective. The FSS QA Plan will be applied to the following aspects of final status survey activities.

5.10.1 Project Management and Organization

The FSS project organization has been established within the Maine Yankee radiation protection organization for planning and implementation of the final status survey. This organization, depicted in Figure 5-6 (at end of Section 5), is directed by the Manager of Projects - FSS who reports to the Radiation Protection Manager (RPM). The RPM maintains overall responsibility for the performance of the final status survey and overall integration of the FSS project with other decommissioning activities[8].

The Final Status Survey project organization consists of the following functional levels:

a. Manager of Projects (MOP) - FSS: The Manager of Projects for Final Status Survey (MOP FSS) is responsible for the administration of, and ensuring the implementation of, the FSS Plan. The MOP FSS is responsible for ensuring activities conducted as part of the FSS are performed in accordance with the FSS Quality Assurance Plan. The MOP FSS is responsible for management of personnel assigned to the FSS section. The MOP FSS is responsible for approving FSS Release Records and ensuring contractual and licensing obligations are satisfied. The MOP FSS reports to the RPM.

b. Superintendent of Radiation Remediation (SRR): The SRR has the overall responsibility for the planning, monitoring and coordination of radiological remediation in preparation for FSS activities. The SRR has responsibility for establishing, maintaining and implementing the programs, procedures and evaluations to support radiological remediation. The SRR has responsibility for the pre-demolition surveys of structures being demolished as well as the control of radioactive material resulting from demolition. The SRR, when directed, has responsibility for Turnover Surveys prior to area acceptance for FSS. The SRR reports to the MOP-FSS.

c. Superintendent of Final Status Survey (SFSS): The Superintendent of Final Status Survey (SFSS) is responsible for the preparation and implementation of the FSS program. The SFSS has overall responsibility for program

[8] See Section 5.10.1 for discussion of the relationship between the FSS project organization and the Maine Yankee Quality Assurance Program

direction, technical content, and ensuring the program complies with applicable NRC regulations and guidance. The SFSS is responsible for resolution of issues or concerns raised by NRC, the State of Maine, or other stakeholders, as well as any programmatic issues raised by Maine Yankee Management. The SFSS provides overall management and direction to FSS personnel. Interface with regulatory agencies and other outside organizations regarding the FSS Program will be conducted primarily by the SFSS. The SFSS reviews and approves the qualification and selection of FSS personnel and approves the content of training to FSS personnel and other personnel on FSS topics. The SFSS approves reports of FSS results. The SFSS reports to the MOP-FSS.

d. Radiochemist: The Radiochemist is responsible for the conduct of the day to day activities performed by Chemistry personnel and for the supervision of the counting room personnel and activities. The Radiochemist is responsible for data quality of onsite FSS sample analyses. (If samples are processed offsite, the MY Quality Assurance Program determines the quality requirements for offsite procurement.) The Radiochemist reports to the Superintendent Radiation Engineering and Technical Support.

e. FSS Engineer (FSSE): The FSS Engineer (FSSE) is responsible for the technical support, development, and implementation of FSS procedures. The FSSE is responsible for the review of survey packages and the review of all data collected in support of the FSSE. The FSSE reviews FSS procedures and reviews reports of FSS results. The FSSE reports to the SFSS.

f. FSS Specialist (FSSS): The FSSS is responsible for preparation of survey packages for individual survey areas, including history files, survey designs and instructions. In addition, the FSSS is responsible for preparation of survey maps, grid maps, layout diagrams, composite view drawings and other graphics as necessary to support FSS reporting. The FSSS reports to the Superintendent FSS.

g. FSS Supervisor: The FSS Supervisor is responsible for control and implementation of survey packages as received from the FSS Specialist. The FSS Supervisor is responsible for coordination of turnover surveys, final status surveys, and survey area preparation such as gridding and accessibility needs. The FSS Supervisor is responsible for coordination and scheduling of FSS Technicians to support the FSS schedule and ensuring all necessary instrumentation and other equipment is available to support survey activities. The FSS Supervisor is also responsible for maintaining access controls over completed FSS survey areas. The FSS Supervisor reports to the SFSS.

h. Instrumentation Technician (IT): The IT is responsible for maintaining the pedigree of instrumentation used for FSS by implementing the procedural requirements for calibration, maintenance and daily checks. The IT ensures that sufficient and properly calibrated instrumentation is available to support FSS. The IT is responsible for the calibration and maintenance of FSS instrumentation. The IT reports to the Instrumentation, Sources and Respiratory Protection Engineer (ISRPE). (The ISRPE's responsibilities include the site RP instrumentation program.)

i. FSS Technician: The FSS Technician is responsible for performance of FSS measurements and collection of FSS samples in accordance with FSS procedures and survey package instructions. The FSS Technician reports to the FSS Supervisor.

j. Site Quality and its Relationship to the Maine Yankee Quality Assurance Program.
   (1) The Maine Yankee Quality Assurance Program has been established as required by, and to assure conformance with, 10CFR50 Appendix B and other regulations relevant to the decommissioning of Maine Yankee.[9]
   (2) The MY President has overall responsibility for all aspects of the QA Program.
   (3) The Quality Programs Manager (QPM) has the overall authority and responsibility for establishing and measuring the effectiveness of the Quality Assurance Program. By provisions in the Program, the QPM has direct access to senior management positions.
   (4) The QPM reports through the Director, Nuclear Safety and Regulatory Affairs, through the Vice President and Chief Financial Officer, who in turn reports to the President).[10]
   (5) The MY Quality Assurance Program supports the FSS QAP by activities and services related to quality, such as, the establishment of requirements and assessing adequacy of implementation for procurement control, procedures and instructions, corrective actions, record retention, and audits/surveillances.

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9 Sections I.B.1 and II.C, MY Quality Assurance Program, May 1, 2002.

10 The overall MY site organization is illustrated (with QA reporting lines) in Figure 6.1-1 of the MY Defueled Safety Analysis Report (DSAR). As noted in this figure, the QPM has a "functional report" to the President on matters of quality (DSAR Section 6.1.2).

5.10.2 Project Description and Schedule

Each area of the site will be divided into survey units and classified as directed by procedure. The survey measurements for each survey unit will be determined during the survey design phase. Portions of the final status survey will be performed during deconstruction activities as areas become available for survey. The non-impacted areas may be evaluated for release prior to significant decommissioning activities taking place.

5.10.3 Quality Objectives and Measurement Criteria

Type I errors will be established at 0.05 unless authorized by the NRC. Type II errors will be set at 0.05 or greater.

a. Training and Qualification

Personnel performing final status survey measurements will be trained and qualified . Training will include the following topics:

- Procedures governing the conduct of the final status survey,
- Operation of field and laboratory instrumentation used in the final status survey, and
- Collection of final status survey measurements and samples.

The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity. Records of training will be maintained in accordance with the approved course description for Initial and Continuing Training for Decommissioning.

b. Survey Documentation

Each final status survey measurement will be identified by date, instrument, location, type of measurement, and mode of operation. Generation, handling and storage of the original final status survey design and data packages will be controlled. The FSS records have been designated as quality documents and, as such, they will be maintained as such in accordance with procedures.

5.10.4 Measurement/Data Acquisition

a. Survey Design and Sampling Methods

The site will be divided into survey areas. Each survey area package may contain one or more survey units. Each survey area package will specify the type and number of measurements required based on the classification of each survey unit.

b. Written Procedures

Sampling and survey tasks must be performed properly and consistently in order to assure the quality of the final status survey results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for the final status survey measurements.

c. Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the final survey results is established by procedure. When custody is transferred, a chain of custody form will accompany the sample for tracking purposes. Secure storage will be provided for archived samples.

d. Quality Control Surveys

Procedures establish built-in Quality Control checks in the survey process for both field and laboratory measurements, as described in LTP Section 5.4.5(f). For structures and systems, QC replicate scan measurements will consist of resurveys of a minimum of 5% of randomly selected class 1, 2, or 3 survey units typically performed by a different technician with results compared to the original measurement. The acceptance criterion shall be that the same conclusion as the original survey was reached based on the repeat scan. If the acceptance criterion is not met, an investigation will be conducted to determine the cause and corrective action.

Quality Control for direct surface contamination and/or exposure rate measurements will consist of repeat measurements of a minimum of 5% of the survey units using the same instrument type, taken by a different technician (except in cases where there is only one instrument or specialized training is required to operate the equipment) and the results compared to the original measurements using the same instrument type. The acceptance criterion for direct measurements is specified in approved procedures.

For soil, water and sediment samples, Quality Control will consist of participation in the laboratory Inter-comparison Program. However, as an additional quality measure, approximately 5% of such samples may be subjected to blind duplicate samples or third party analyses. The acceptance criterion for blank samples is that no plant-derived radionuclides are detected. The criterion for blind duplicates is that the two measurements are within the value specified by approved procedure. For third party analyses, the acceptance criterion is the same as those for blind duplicates. Some sample media, such as asphalt, will not be subjected to split or blind duplicate analyses due to the lack of homogeneity. These samples will simply be recounted to determine if the two counts are within 20% of each other, when necessary.

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be performed in accordance with approved procedures; and if necessary, the Corrective Action Process described in Section 5.10.5(c) will be implemented. The investigation will typically involve verification that the proper data sets were compared, the relevant instruments were operating properly and the survey/sample points were properly identified and located. Relevant personnel are interviewed, as appropriate, to determine if proper instructions and procedures were followed and proper measurement and handling techniques were used including chain of custody, where applicable. When deemed appropriate, additional measurements are taken. Following the investigation, a documented determination is made regarding the usability of the survey data and if the impact of the discrepancy adversely affects the decision on the radiological status of the survey unit.

e. Instrumentation Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as specified in Section 5.5.2 as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the Instrumentation Program procedure.

f. Control of Consumables

In order to ensure the quality of data obtained from FSS surveys and samples, new sample containers will be used for each sample taken. Tools used to collect samples will be cleaned to remove contamination prior to taking additional samples. Tools will be decontaminated after each sample collection and surveyed for contamination.

g. Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, will be procured from appropriate vendors in accordance with approved quality and procurement procedures.

h. Database Control

Software used for data reduction, storage or evaluation will be fully documented and certified by the vendor. The software will be tested prior to use by an appropriate test data set.

i. Data Management

Survey data control from the time of collection through evaluation is specified by procedure. Manual data entries will be second verified.

5.10.5 Assessment and Oversight

a. Assessments

FSS self-assessments will be conducted in accordance with approved procedures. The findings will be tracked and trended in accordance with these procedures.

b. Independent Review of Survey Results

Randomly selected survey packages (approximately 5%) from survey units will be independently reviewed by the Quality Programs Department to ensure that the survey measurements have been taken and documented in accordance with approved procedures.

c. Corrective Action Process

The corrective action process, already established as part of the site's 10 CFR Part 50 Appendix B Quality Assurance Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. The process will be conducted in accordance with approved procedures which describe the methods used to initiate Condition Reports (CRs) and resolve self assessment and corrective action issues related to FSS. The CR evaluation effort is commensurate with the classification of the

CR and could include root cause determination, barrier screening and extent of condition reviews.

d. Reports to Management

Reports of audits and trend data will be reported to management in accordance with approved procedure.

5.10.6 Data Validation

Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels will be made and measurements exceeding the investigation levels will be evaluated. Procedurally verified data will be subjected to the Sign test, the Wilcoxon Rank Sum (WRS) test, or WRS Unity test as appropriate. Technical evaluations or calculations used to support the development of DCGLs will be independently verified to ensure correctness of the method and the quality of data.

5.10.7 NRC and State Confirmatory Measurements

Maine Yankee anticipates that both the NRC and the State of Maine Department of Human Services (DHS) - Division of Health Engineering (DHE) may choose to conduct confirmatory measurements in accordance with applicable laws and regulations. The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the final radiation survey and associated documentation demonstrate that the facility and site are suitable for release in accordance with the criteria for decommissioning in 10 CFR Part 20, subpart E. Maine state law requires Maine Yankee to permit monitoring by the Maine State Nuclear Safety Inspectors (22 MRSA 664, sub-§2, as amended by PL 1999, c. 739, §1 and 38 MRSA 1451, sub-§11, as amended by PL 1999, c. 741, §1) This monitoring includes, among other things, taking radiological measurements for the purpose of verifying compliance with applicable state laws (including the enhanced state radiological criteria) and confirming and verifying compliance with NRC standards for unrestricted license termination. Maine Yankee will demonstrate compliance with the 25 mrem/yr criteria of 10 CFR Part 20, Subpart E by demonstrating compliance with the enhance state radiological criteria. Therefore, the confirmatory measurements taken by the NRC and the State of Maine will be based upon the same criteria, DCGL. Timely and frequent communications with these agencies will ensure that they are afforded sufficient opportunity for these confirmatory measurements prior to Maine Yankee implementing any irreversible decommissioning actions (e.g. backfilling basements with soil fill material.)

5.11 Access Control Measures

5.11.1 Turnover

Due to the large scope of the final status survey and the need for some activities to be performed in parallel with dismantlement activities, a systematic approach to turnover of areas is established. Prior to acceptance of a survey unit for final status survey, the following conditions must be satisfied, unless authorized by the FSS Superintendent in accordance with established procedures. These include:

a. Decommissioning activities having the potential to contaminate the survey unit must be complete.

b. Tools and equipment not required for the survey must be removed, and housekeeping and cleanup must be complete, except as noted in section 5.1.2.a.

c. Decontamination activities in the area must be complete.

d. Final remediation surveys, where applicable, must be complete. These surveys will consist of:

   1. Scan surveys or fixed measurements to ensure that surface contamination is within the FSS total surface contamination limits.

   2. Smear surveys to ensure that the removable surface contamination is within the FSS removable surface contamination limits (i.e., 10% of the surface contamination limit).

   3. Volumetric samples or scans to ensure soil remediation is within acceptable FSS concentration limits.

e. Access control or other measures to prevent recontamination must be implemented.

f. Turnover surveys may be performed and documented to the same standards as FSS surveys so that data can be used for FSS.

5.11.2 Walkdown

The principal objective of the walkdown is to assess the physical scope of the survey unit. For systems, it will include a review of system drawings and a physical walkdown of the system. Structures and open land areas will also be walked down. The walkdown is best completed when the final configuration of the area is known, usually near or after completion of decommissioning activities for the area.

The walkdown ensures that the area has been left in the necessary configuration for FSS or that any further work has been identified. The walkdown provides detailed physical information for survey design. Details such as floor coatings, structural interferences or sources needing special survey techniques can be determined.

Specific requirements will be identified for accessing the survey area and obtaining support functions necessary to conduct the final status surveys, such as scaffolding, interference removal, and electrical tag out. Safety concerns, such as access to confined spaces, tidal areas, and high walls and/or ceilings, will be identified.

5.11.3 Transfer of Control

Once a walkdown has been performed and the turnover requirements have been met, control of access to the area is transferred from the Construction and Radiation Protection operations groups to the FSS group. Turnover is accomplished using administrative controls. Access control and isolation methods are described below.

5.11.4 Isolation and Control Measures

Since decommissioning activities will not be completed prior to the start of the final status survey, measures will be implemented to protect survey areas from contamination during and subsequent to the final status survey. Decommissioning activities creating a potential for the spread of contamination will be completed within each survey unit prior to the final status survey. Additionally, decommissioning activities which create a potential for the spread of contamination to adjacent areas will be evaluated and controlled.

Upon commencement of the final status survey for survey areas within the RA where there is a potential for re-contamination, implementation of one or more of the following control measures will be required:

a. Personnel training

b. Installation of barriers to control access to surveyed areas

c. Installation of barriers to prevent the migration of contamination from adjacent overhead areas

d. Installation of postings requiring contamination monitoring prior to surveyed area access

e. Locking entrances to surveyed areas of the facility

f. Installation of tamper-evident labels

Routine contamination surveys will be performed in areas following FSS completion to monitor for indications of re-contamination and to verify postings and access control measures. Survey frequency will be based on the potential for re-contamination as determined by the FSS Superintendent. At a minimum, routine surveys will be performed quarterly for structures located within the RA. Routine contamination control surveys will not be required for open land areas and structures outside of the RA that are not normally occupied and are unlikely to be impacted by decommissioning activities.

Routine surveys of areas where FSS has been completed will normally include survey locations at floor level and on lower walls. Locations will be selected on a judgmental basis, based on technician experience and conditions present in the survey area at the time of the survey, but are primarily designed to detect the migration of contamination from decommissioning activities taking place in adjacent and other areas in close proximity which could cause a potential change in conditions.

5.12 References

5.12.1 10CFR20.1402, Radiological Criteria for Unrestricted Use.

5.12.2 10CFR50.82, Termination of License.

5.12.3 40CFR141.25 through 27, National Primary Drinking Water Regulations.

5.12.4 State of Maine Law - LD 2688-SP1084, "An Act to Establish Clean-up Standards for Decommissioning Nuclear Facilities," April 26, 2000

5.12.5 MY Post Shutdown Decommissioning Activities Report (PSDAR), MN-97-99, dated August 27, 1997 as supplemented by MN-98-65 dated November 3, 1998 .

5.12.6 MY Historical Site Assessment, as transmitted by MN-01-038 dated October 1, 2001.

5.12.7 GTS Duratek, "Characterization Survey Report for the Maine Yankee Atomic Power Plant," Volumes 1-9, 1998.

5.12.8 MY Quality Assurance Program.

5.12.9 MY Corrective Action Program.

5.12.10 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM), Revision 1 (June 2001)

5.12.11 NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Field Conditions," December 1997

5.12.12 NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," Rev.1, June 1998 draft.

5.12.13 NUREG-1549, "Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination," July 1998 draft.

5.12.14 Appendix E, NUREG 1727, "Demonstrating Compliance with the Radiological Criteria for License Termination," September 15, 2000.

5.12.15 NUREG-1727, "NMSS Decommissioning Standard Review Plan," September 15, 2000.

5.12.16 Initial and Continuing Training For Decommissioning course descriptions.

5.12.17 Radiation Protection Performance Assessment Program (PMP 6.0.8).

5.12.18 Radiation Protection Instrumentation Program (PMP 6.4).

5.12.19 Operation and Calibration of the Gamma Spectroscopy System (DI 6-306).

5.12.20 Operation of the Packard Model 4430 Liquid Scintillation (DI 6-316)

5.12.21 Final Status Survey Program (PMP 6.7).

5.12.22 FSS Survey Procedure for Structures, Systems and Soils (PMP 6.7.1).

5.12.23 FSS Survey Unit Classification (PMP 6.7.2).

5.12.24 FSS Quality Control (PMP 6.7.3).

5.12.25 FSS Survey Package Preparation and Control (PMP 6.7.4).

5.12.26 FSS Survey Area Turnover and Control (PMP 6.7.5).

5.12.27 FSS Data Processing and Reporting (PMP 6.7.8).

5.12.28 Selection, Training and Qualification of RP/Waste Personnel (PMP 6.9).

5.12.29 Instrument Quality Assurance (PMP 6.4.1).

5.12.30 Document Control Program (0-17-1).

5.12.31 Operation of the Tennelec LB-5100 Gas Flow Proportional Instrument procedure (DI 6-210).

5.12.32 Instrument Selection and MDC Calculation (EC 009-01).

5.12.33 NRC letter to Maine Yankee, dated August 23, 2002 regarding classification downgrade and other LTP issues.

5.12.34 Use of the SPA-3 Detector for Concrete Scan Surveys (EC 002-03)

5.12.35 White Paper 2002-001, "The Approach for Dealing with Background Radioactivity for Maine Yankee Final Status Surveys"

5.12.36 Revised Report on Eberline Model E-600 Field Testing (MN- 03-009)

5.12.37 Maine Yankee Letter to NRC, MN-03-051, dated September 3, 2003, Technical Basis Document for NRC Review - Forebay FSS Survey Measurement Methods (In-Situ Gamma Spectroscopy) - 30 Day Notice per LTP Requirement

5.12.38 Maine Yankee Letter to NRC, MN-03-067, dated October 21, 2003, Maine Yankee Response to NRC and State of Maine Comments on the Technical Basis Document for NRC Review - Forebay FSS Survey Measurement Methods (In-Situ Gamma Spectroscopy)

ATTACHMENT 5A

Embedded and Buried Pipe
Initial Final Survey Classification Description

Embedded and Buried Piping Remaining on Site:

The following sections of embedded and buried piping will remain on site following demolition of above grade structures. This list includes a description of the piping, the potential for the piping to contain residual contamination and a description and the initial MARSSIM classification of the survey units.

Containment Spray (C0300)

System Description: The function of the Containment Spray (CS) system was to reduce the peak pressure in the containment building following a loss of coolant accident by spraying water into the containment atmosphere, to remove radioactive iodine, which would be released to the containment atmosphere during a loss of coolant accident, and to supply water to the suction of the High Pressure Safety Injection pumps following receipt of a Recirculation Actuation Signal (RAS) to provide the required suction head. The CS system initially took suction from the Refueling Water Storage Tank. The system could take an alternate suction from the containment safeguards sump upon receiving the RAS signal.

Residual Contamination Potential: The Containment Spray piping has a high potential for residual contamination. The portion of the piping that will remain following demolition of above grade structures is embedded in the concrete foundation of the Containment Building. The water source available for the system, Refueling Water Storage Tank, was contaminated.

Survey Units: The Containment Spray piping will be surveyed as a single survey unit. The survey unit will have an initial MARSSIM classification of Class 1. The classification is based on the known presence of contamination in the suction source for the system.

Containment Foundation Drains (C2000)

System Description: The Containment Foundation Drain piping is used to transfer groundwater from around the foundation of the Containment Building to lower the hydrostatic pressure exerted on the foundation. The remaining piping consists of four, two inch ID, horizontal, plastic, transfer pipes at approximately the -46' 6" elevation which run radially from underneath the ICI pit to the Containment Foundation Drain Sump Pumpwell and one, six inch, horizontal, open joint clay pipe at approximately the -18' 6" elevation which runs about 90 degrees around the southwest circumference of the containment foundation from the Spray Building to the Containment Foundation Drain Sump Pumpwell. The horizontal transfer pipes drain to the common, vertical, six foot ID, Containment Foundation Drain Sump Pumpwell which runs from the -52' 3" elevation to grade level.

Residual Contamination Potential: The Containment Foundation Drain piping has a potential for residual contamination, but is not likely to contain residual radioactivity in excess of the $DCGL_w$. The piping is wholly contained in the Restricted Area and there are known instances of contaminated liquid spills in the area around the Containment Building.

Survey Units: The Containment Foundation Drain piping will be surveyed as a single survey unit. The initial MARSSIM classification of the survey unit was Class 1. The basis for classification was operational knowledge of the system and data collected in support of the Radiological Environmental

Monitoring Program. Upon reevaluation of continued characterization data with respect to the balance of plant embedded piping DCGLw, this survey unit has been reclassifed to Class 2.

Sanitary Waste (D0400)

System Description: The Sanitary Waste (SW) piping was used to transfer waste from the various buildings on site to the Sewage Treatment Plant where the waste was treated prior to disposal. The system transferred waste from all areas of the site including sanitary facilities formerly located in the Restricted Area. The portions of the piping that will remain after the demolition of above grade structures will be contained within the Manhole system described in the Storm Drains system. The Radiological Environmental Monitoring Program requires that this outfall be monitored periodically. The original outfall for the system was to the Back River following treatment. In the mid-1980s, the outfall for the system was connected to the city of Wiscasset sewage treatment system.

Residual Contamination Potential: The Sanitary Waste piping has a low potential for residual contamination. The leg of the piping that formerly serviced the sanitary facilities in the Restricted Area was removed from service in the early 1980s. Other portions of the system may have been contaminated with medical isotopes; however, these isotopes are short lived and should be decayed away by the time the system is surveyed.

Survey Units: The abandoned leg of the Sanitary Sewer piping that connected the sanitary facilities in the Restricted Area to the Sewage Treatment Plant will be surveyed as a single survey unit. The initial MARSSIM classification of the piping will be Class 3. The classification is based on operational knowledge of the system and survey data collected during initial site characterization[1].

Circulating Water (D0500)

System Description: The Circulating Water (CW) system supplied cooling water to the main condenser tube bundles. The system took suction from the Back River at the Circulating Water Pump House. Four CW pumps took suction from an individual bay and discharged to an individual tube bundle. The CW in the tube bundle removed heat from the turbine exhaust steam that condensed the steam to condensate water for return to the steam generators. The CW exiting the tube bundles combined and was directed to the seal pit and the forebay. Water from the seal pit and forebay was returned to the Back River. The Circulating Water system is considered a "secondary side" system in that there was a physical barrier (Main Steam and Condensate systems) between the water in the Circulating System and the contaminated systems of the primary plant (Reactor Coolant, etc.).

Residual Contamination Potential: The Circulating Water piping has a very low potential for residual

1 "Initial site characterization" (or ICS) refers to the initial characterization work performed by GTS Duratek as documented in the "Characterization Survey Report for the Maine Yankee Atomic Power Plant," 1998. (See the Reference Section 5.12.) "Continuing characterization" refers to additional characterization which followed the ICS and is an ongoing activity which collects additional data, as required, to support remediation, dose assessment, and FSS activities. See also Section 2.1.

contamination. The piping was separated from the primary system by several interface systems. The Steam Generator U-tubes acted as the separator for the primary and secondary systems, and the main condenser tube bundles acted as the separator for the secondary system (Main Steam, Condensate, etc.) and the CW piping. The operational history of the facility indicates that no significant primary to secondary leakage occurred, implying that there is a very remote chance the system may have become contaminated. Additionally, CW system pressure was maintained above the pressure of the turbine exhaust steam. In the event of a tube bundle leak, the CW system water would have leaked into the Condensate system instead of Condensate leaking into the CW system. During site characterization activities, low levels of detectable activity were identified on the main condenser outlet side of the Circulating Water piping. Continuing Characterization Survey samples collected in the CW piping identify very low levels of plant related radionuclides. The suspected cause of the contamination was recirculation of allowable effluent discharges into the suction side of the Circulating Water Pump House.

Survey Units: The Circulating Water system will be divided into two survey units. The first survey unit will consist of the inlet side piping ending at the floor of the Turbine Hall where the pipes have been cut off at floor level. The second survey unit will consist of the outlet side piping at the floor of the Turbine Hall where the pipes were cut off at floor level and ending at the Seal Pit and Fore Bay area. Both survey units for this survey area will initially be classified as MARSSIM Class 3. The basis for classification of the survey units is operational knowledge of the system, data obtained in support of the Radiological Environmental Monitoring Program, and limited sampling of the piping conducted during site characterization surveys.

<u>Service Water (D0600)</u>

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

The discharge side of the pipe receives the liquid effluent discharge pipe. The waste header is contained within its own local Restricted Area within the Turbine Building. 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 radwaste piping will be removed and disposed of as radioactive waste. The buried inlet portions of the Service Water system will be removed outside of the Turbine Building and the portions beneath the Turbine Building will be abandoned in place. The remaining portions of the service water discharge piping meet the criteria of a Class 3 area and will be surveyed as a single survey unit.

<u>Fire Protection (D0700)</u>

System Description: The water portion of the Fire Protection (FP) system is the only section that will remain following demolition of above grade structures. Water for firefighting was stored in a man-made storage pond located northwest of the plant. Makeup water for the pond was supplied from the Montsweag Reservoir. Water was transferred to the storage pond by two reservoir pumps, which were operated as required to keep the storage pond full. The former storage pond is addressed as part of survey area R0900. Two fire pumps took suction from the storage pond and discharged to the yard loop where they supplied various fire headers and hydrants. The FP system did not supply firefighting water to the Containment Building. The hose stations in the Containment Building were supplied from the Primary Water System. The Fire Protection system is considered a "support system" in that it did not interface with the primary or secondary side of the nuclear steam supply system.

Residual Contamination Potential: The Fire Protection piping has a very low potential for residual contamination. The piping did not interface with either the primary or the secondary side systems of the nuclear steam supply system. Although sections of the piping reside in the Restricted Area, the system operating pressure, even at static head conditions, was sufficient to ensure that any leakage would occur from the system, not into the system. The Fire Water Protection system has been inadvertently cross connected with potentially contaminated systems in the past. Samples collected during the Continuing Characterization Survey have only identified naturally occurring radioactive material. No licensed activity has been identified in the system.

Survey Units: The Fire Protection piping will be surveyed as a single survey unit. The survey unit will consist of all buried and embedded piping remaining after the demolition of the site above grade structures. The initial MARSSIM classification for the Fire Protection piping will be Class 3. The classification is based on knowledge of system operation and samples collected in the storage pond during site characterization surveys and samples of the system collected as part of the Continuing Characterization Survey.

Storm Drains (D3500)

System Description: The Storm Drain (SD) system is used to drain water from the facility to the Back River. The system functions as a gravity drain system to remove the water via a system of drain grates, manholes and piping. The system drains the entire site both inside and outside the Protected Area. Manholes 1 through 3 (Section 1 of the piping) drain the Protected Area outside the Restricted Area and south of the Turbine Building and Service Building. The outfall for this portion of the piping 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 piping) 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 piping 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 piping) 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 piping) drain the upper access road and the upper contractor parking lot. The outfall for this section of the piping is the Back River north of the Information Center building. Manholes 30A, and 31 through 37 (Section 5 of the piping) drain an area inside the Protected Area in the Restricted Area. This leg of the piping 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 Seal Pit Forebay. Manholes 21 through 24 (Section 6 of the piping) 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 piping) 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 piping is routed to Bailey Cove.

Residual Contamination Potential: The Storm Drain piping has a low potential in some legs and a high potential in some legs for residual contamination. Sections 1 through 4 and section 7 upstream of manhole 27 have a low potential for residual contamination. Sections 5 through 7 (downstream of and including manhole 27) have a high potential for residual contamination. Sections 1 through 4 and section 7 upstream of manhole 27 drain areas that have historically been outside the Restricted Area and have a low potential for residual contamination. Sections 5 through 7 (downstream of and including manhole 27) 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.

Survey Units: The Storm Drain piping may be divided into two survey units. The first survey unit will include sections 1 through 4 and section 7 upstream of manhole 27 of the piping. The initial MARSSIM classification for this section of the piping will be Class 3. The basis for classification is operational knowledge, survey data obtained for initial site characterization activities and as part of the Continuing Characterization Survey, and results of the Radiological Environmental Monitoring Program. The second survey unit will consist of sections 5 through 7 (downstream of and including manhole 27) of the piping. The initial MARSSIM classification for this section of the piping will be Class 1. The basis for classification is operational knowledge and survey data obtained during initial site characterization and the Continuing Characterization Survey.

Roof Drains (D3600)

System Description: The Roof Drain (RD) system removed water from the roofs of various site buildings and transferred the water to the Storm Drain system. The Roof Drains from buildings outside the RCA were routed to the Storm Drain piping sections that will be classified as Class 3. The Roof Drains from buildings inside the RCA were routed to the Storm Drain piping sections that will be classified as Class 1.

Residual Contamination Potential: Sections of the Roof Drain system outside the RCA have a low potential for residual contamination. Sections of the Roof Drain system inside the RCA have a high potential for residual contamination.

Survey Units: The portions of the system that will remain following demolition of above grade structures are buried and embedded sections of the system that are associated with the Storm Drain system. For this reason, the Roof Drains will be surveyed as part of the Storm Drain system.

Containment, Primary Auxiliary Building and Containment Spray Building Penetrations (D3700)

System Description: 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 20 linear feet of up to 1" piping
- Approximately 35 linear feet of 1.5" piping
- Approximately 50 linear feet of 2" piping
- Approximately 35 linear feet of 3" piping
- Approximately 55 linear feet of 4" piping
- Approximately 100 linear feet of 6" piping
- Approximately 45 linear feet of 8" piping
- Approximately 5 linear feet of 10" piping
- Approximately 25 linear feet of 16" piping
- Approximately 10 linear feet of 24" piping
- Approximately 20 linear feet of 30" piping
- Approximately 11 linear feet of 40" Fuel Transfer Tube piping

Each of these penetration, except for the Fuel Transfer Tube, consists of a five foot length of pipe penetration through the containment foundation wall. The calculated surface area of this embedded piping is approximately 78 m^2.

The Primary Auxiliary Building and Spray Building Penetrations (60ft). Several non-containment piping penetrations through the Primary Auxiliary Building and Spray Building will remain in the respective building foundations following demolition of the above grade structure. Each of these penetrations consists of a 2 to 3 foot length of pipe penetration through the building foundation wall. The calculated surface area of this embedded piping is approximately 19.5 m^2.

The spent fuel pool liner leak detection system (24ft). Four 1 inch lines embedded in the spent fuel pool structure will remain following demolition of the above grade structure. The calculated surface area of this embedded piping is approximately 0.6 m^2

Residual Contamination Potential: The penetrations that will remain in the Containment Building, Primary Auxiliary Building and Spray 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.

Survey Units: The remaining sections of embedded piping in the Containment Building may be surveyed as a single survey unit. The initial MARSSIM classification assigned to the penetrations is Class 1. The basis for classification is the known presence of contamination in the Containment Spray system, the potential for residual contamination in the remaining piping due to system operation and lack of control of the penetrations to prevent contamination during dismantlement activities in the Containment Building.

Class 1 Survey Units:

Containment Spray System (C0300)

Physical Characteristics: The remaining embedded section of the Containment Spray piping consists of metal piping.

Decontamination: Prior to performing the FSS, the remaining piping will be decontaminated. The decontamination will consist of hydrolasing the embedded piping from the Containment Safeguards Sump to the suction of the Containment Spray Pumps. Following the hydrolasing, the leg of embedded piping will be surveyed for gross removable contamination.

Scan surveys for the Containment Spray piping will be conducted at the accessible ends of the embedded piping. The surface area scanned will be a small percentage of the total area of the system. The location of the measurements will be determined by dividing the total length of the pipe by the number of measurements to be collected. The systematic spacing of the survey measurements is in keeping with the guidance of NUREG-1575 and NUREG-1727. Total Surface Contamination measurements will be collected using a pipe crawler.

Containment Foundation Drains (C2000) - Moved to Class 2 Survey Units

Storm Drains (D3500)

Survey Unit: The Class 1 survey unit for the Storm Drain piping consists of the section of the piping bound by Manholes 30A and 31 through 37 and the section of the piping bound by Manholes 21 through 24. The survey unit includes an unnumbered manhole adjacent to the location of tank TK-16 in the Restricted Area yard.

Physical Characteristics: The remaining sections of buried Storm Drains piping consist of both metal and concrete piping. Some of the metal sections are smooth wall and some are corrugated.

Decontamination: The piping will require decontamination prior to performance of the Final Status Survey. The decontamination will consist of removing the sand and sediment from the piping low points and accesses (the manholes). The sand in the piping contains naturally occurring radioactive material.

Scan Surveys: Although this is Class 1 piping, physical access limits available measurement locations and scan survey locations. Therefore, scan surveys for the Storm Drain piping will be limited to accessible portions of the piping Scan surveys will be performed in areas with the highest potential for contamination based on professional judgment. For this reason, the scan survey will be biased to piping low points and interfaces and the scan survey will be performed in the vicinity of the Total Surface Contamination measurements identified for the piping. Scan surveys will be performed on as much of the interior surfaces of the piping as possible.

Survey Location Designation: Survey measurements for the Storm Drain piping will be collected at existing access points. The locations will be selected based on engineering judgment and biased to areas expected to contain the highest residual activity levels. As the Final Status Survey of the remaining embedded and buried piping for the Storm Drain system will be biased and not random, the minimum number of measurements collected on the system interior surfaces will be the number calculated using the methods described above or 30 measurements, whichever is greater.

Building Penetrations (D3800)

Physical Characteristics: The remaining embedded piping in the Building Penetrations survey unit consists of smooth metal piping surfaces.

Decontamination: The embedded piping remaining in the system will be decontaminated prior to performance of the Final Status Survey.

Scan Survey Coverage: 100% of the accessible system surfaces will receive a scan survey. Sections of embedded piping that are inaccessible will receive 100% gross removable contamination surveys. This will include sections that are too small to allow probe entry into the pipe.

Survey Location Designation: Each penetration will be assigned a number. The number of fixed point measurements will be calculated using the method described in the "sample size determination" section of this plan. The measurements will be randomly assigned to the penetrations. The random measurements will be used due to the difficulty of performing a systematic survey of the penetrations. The penetrations reside at multiple elevations of the building in a non-contiguous manner. These factors make it virtually impossible to perform a systematic survey of the penetrations.

Class 2 Survey Units:

Containment Foundation Drains (C2000)

Physical Characteristics: The remaining buried sections of the Containment Foundation Drains piping consists of plastic and clay piping. The vertical pumpwell wall has perforated sections to allow groundwater to enter the pumpwell. The horizontal piping consists of intact plastic and open joint clay piping.

Decontamination: The Containment Foundation Drain piping is not expected to require decontamination. Samples of the outlet of the piping collected for the Radiological Environmental Monitoring Program have identified Tritium as the only plant related radionuclide in the outlet.

Scan Surveys: Scan surveys for the Containment Foundation Drain piping will be limited to accessible portions of the piping from the Containment Foundation Drain Sump Pumpwell. Scan surveys will be performed on 10 to 100% of the interior surfaces of the piping and pumpwell. The number of measurements will be determined using the sign test and will be applied to the total accessible surface area of the pipe and pumpwell. The systematic spacing of the survey measurements is in keeping with the guidance of NUREG-1575 and NUREG-1727. Total Surface Contamination measurements will be collected using a manually deployed detector. When direct sample locations fall upon surfaces which are not amenable to suface detection (e.g., moisture saturated surfaces or pipe access restraicted by calcium build-up), the volumetric samples of concrete or internal pipe scrapings will be taken and analyzed in accordance with Section 5.5.1.a.

A volumetric sample will also be taken of sediment accumulated at the bottom of the sump pumpwell, if available.

Class 3 Survey Units:

Scan Survey Coverage:

Scan surveys for Class 3 system survey units will be determined based on the Historical Site Assessment (HSA) for the survey unit. In cases where the initial site characterization and the continuing site characterization did not identify the presence of removable contamination or fixed point total surface contamination in excess of the $DCGL_w$, the areal extent of the scanning will be determined by engineering judgment and should be in the range of 1 to 10% of the accessible surfaces of the system. Section 5.5.3 of NUREG-1575 recommends that scan surveys be performed in areas with the highest potential for contamination based on professional judgment. For this reason, the scan survey will be biased to system low points and system interfaces and the scan survey will be performed in the vicinity of the Total Surface Contamination measurements identified for the system.

Sample Size Determination:

The number of samples required for a survey unit is based on the following:

Statistical Test to be used: For Class 3 system survey units, the sign test will be used to test the null hypothesis.

Estimate of Standard Deviation: The estimated standard deviation values for the systems will be derived from characterization data or measurements additional background measurements, if necessary. In the event that there is insufficient data to estimate the standard deviation, the standard deviations developed for Class 3 structural survey units with similar contamination potential as the system (i.e. Turbine Building 21' elevation may be used for the Circulating Water system). The basis for the estimated standard deviation used for the design of the Final Status Survey of the survey area or survey unit will be given in the survey package design instructions.

The previously listed factors directly impact the number of measurements that will be collected in each survey unit. This method of calculating the number of survey measurements is valid regardless of the size of the survey unit or the type of material (i.e. structure or open land area) being surveyed. Experience has shown that this method typically requires that approximately 14 measurements are required for each survey unit at the Maine Yankee site. This method may also be used to determine the number of measurements required to demonstrate compliance in a system survey unit. The basis for the method described is that random designation of survey measurement location allows for a lower sample population to be used for the statistical analysis of the survey unit. As the Final Status Survey of the remaining embedded and buried piping systems will be biased and not random, the minimum number of measurements collected on the system interior surfaces will be the number calculated using the methods described in the "Sample Size Determination" section or 30 measurements, whichever is greater.

MAINE YANKEE

LTP SECTION 6

COMPLIANCE WITH RADIOLOGICAL DOSE CRITERIA

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Attachments

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6.0 COMPLIANCE WITH THE RADIOLOGICAL DOSE CRITERIA

6.1 Introduction

The goal of the MY decommissioning project is to release the site for unrestricted use in compliance with the NRC's annual dose limit of 25 mrem/y plus ALARA and the enhanced State of Maine clean-up criteria of 10 mrem/y or less for all pathways and 4 mrem/y or less for groundwater sources. Both the State and NRC dose limits apply to residual radioactivity that is distinguishable from background. This section provides the methods for calculating the annual dose from residual radioactivity that may remain when the site is released for unrestricted use.

The dose assessment methods are used to determine Derived Concentration Guideline Levels (DCGLs) for nine different potentially contaminated materials. The DCGLs are the levels of residual radioactivity that correspond to the enhanced state clean-up criteria of 10 mrem/y or less for all pathways and 4 mrem/y or less for groundwater sources to the average member of the critical group. The DCGLs developed to demonstrate compliance with the enhanced State criteria are intended to also serve to demonstrate compliance with the NRC's 25 mrem/y plus ALARA regulation.

Maine Yankee intends to dismantle equipment and systems and remediate structures and land areas (per LTP Sections 3 and 4) to ensure that residual radioactivity levels are at, or below, the DCGLs. After remediation is completed, a final site survey will be performed (per LTP Section 5) to verify compliance with the DCGLs. The final survey report will document that the DCGLs have been met and serve to demonstrate that the Radiological Criteria for License Termination, as codified in 10 CFR 20 Subpart E and Maine State Law LD 2688-SP 1084 have been fully satisfied.

A dose assessment will be performed for each of the following materials: 1) contaminated building basement surfaces; 2) embedded pipe; 3) activated concrete/rebar; 4) groundwater; 5) surface water; 6) surface soils; 7) buried piping; 8) deep soils; and 9) Forebay sediment. Appropriate dose models and model input parameters were developed and justified for each material. The dose from each material was evaluated and summed with that from other materials as necessary to determine the total dose to the average member of the critical group.

6.2 Site Condition After Decommissioning

This section provides a brief overview of the planned site condition after decommissioning as well as a summary of site geology and hydrology. Detailed information on the planned final site condition is provided in Section 3.2.4. LTP

Section 8.4 provides a more detailed overview of the geological and hydrological characteristics of the site.

In general, when decommissioning is complete the site will be predominantly a backfilled and graded land area restored with indigenous vegetative cover. The only above grade structures remaining per the current plans include the 345 KV switchyard. The former Low-Level Waste Storage Building (now the ISFSI Security Operations Building) will remain in place until the fuel is removed from the ISFSI. Building basements and foundations greater than three feet below grade will be backfilled and left in place. Buried piping that is at least three feet below grade will be remediated as necessary, surveyed, and abandoned in place.

6.2.1 Site Geology and Hydrology

The site geology consists of a series of ridges and valleys striking north-south that reflect the competency and structural nature of the underlying bedrock. Deep valleys are filled with glaciomarine clay-silt soil and ridges are characterized by exposed bedrock or thin soil cover over rock. Surface drainage moves both to the north and south along the axes of the topographic valleys and also runs east and west down the flanks of the ridges. In the plant area, where the ground surface is relatively flat, manmade underground storm drains and catch basins control the surface runoff. In the area south of Old Ferry Road, drainage from a large area north of Old Ferry Road and the northern half of Bailey Point discharges in underground manmade piping to Bailey Cove.

The groundwater regime at the Maine Yankee facility is comprised of two aquifers: (1) a discontinuous surficial aquifer in the unconsolidated glaciomarine soils and fill material; and (2) a bedrock aquifer. The surficial aquifer is not present continuously across the site, as the overburden soils are thin to non-existent in some portions of the site. This is especially true in the southern portion of Bailey Point. The bedrock aquifer is present below the entire site and vicinity.

Groundwater originating near the surface in the northern portion of the site generally moves vertically into the soil except in the wetland areas where groundwater discharge locally occurs. After slow movement through the soil, the groundwater moves into the deeper bedrock and travels toward the bay, discharging upward in the near-shore area. In the southern portion of the site, groundwater originating near ground surface generally stays near the surface, rather than penetrating deep into the bedrock.

During plant operation, impacts to the groundwater flow regime were limited to draw-down of the groundwater surface caused by foundation drains around the containment structure and, to a lesser extent, draw-down caused by active water supply wells. Following decommissioning of the containment structure, groundwater levels will recover to approximate pre-construction levels.

6.3 Critical Group

The regulations in 10 CFR 20 Subpart E require the dose to be calculated for the average member of the critical group. The critical group is defined in 10CFR20.1003 as "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances." The average member of the critical group is a conservative approach and is also used for demonstrating compliance with the dose criteria in Maine State Law LD 2688-SP 1084. The critical group selected for the MY site dose assessment is the resident farmer.

The resident farmer is a person who lives on the site after the site is released for unrestricted use and derives all drinking and irrigation water from an onsite well. In addition, a significant portion of the resident's diet is assumed to be derived from food grown onsite. NRC guidance in NUREG-1727, NUREG-1549, and NUREG-5512 identify the resident farmer as a conservative onsite critical group. The resident farmer critical group applies to existing open land areas and all site areas where standing buildings have been removed to three feet below grade.

It is unlikely that other future site uses would result in a dose exceeding that calculated for the hypothetical resident farmer. It is more probable that actual future occupants of the site would engage in behaviors that would result in lower doses. For example, it is more likely that a hypothetical future resident would use the municipal water supply, as opposed to well water, since this is the common practice in the vicinity of the site and the yield from onsite test wells has been determined to be low and not suitable for consumption. Further, it is most likely that the site will be limited to industrial use. In this case the future site occupant would be a worker as opposed to the resident farmer. A third example would be an onsite resident who does not derive a significant fraction of dietary needs from an onsite farm. The important conclusion from these examples is that the dose calculated for the hypothetical resident farmer will likely be a conservative estimate of the dose that an actual site occupant or site visitor would receive.

Maine Yankee has assessed the potential for the filled basements to be excavated and occupied at some time in the future and does not believe that this scenario meets the "reasonable expectation" threshold required by the definition of a critical group in 10 CFR 20.1003. As stated in NUREG-1727, page C26, compliance with the dose limit does not require an investigation of all possible scenarios and the use of the average member of the critical group is intended to emphasize the uncertainty and assumptions needed in calculating potential future dose, while limiting "boundless speculation" on possible future exposure scenarios. As discussed above, selecting the resident farmer critical group is a sufficiently conservative projection of future land use. Further assuming that an individual excavates filled basements and attempts to renovate and occupy the basements is not considered plausible and results in excessive conservatism.

Notwithstanding the very low probability of excavation occurring, Maine Yankee will limit the potential activity on basement fill to concentrations below the surface soil DCGL level corresponding to 10 mrem/y. In addition, cost studies conducted to date indicate that it is more expensive to remediate soil than basement surface contamination. As discussed in Section 6.9, the selected Basement Contamination DCGLs are limited in order to maximize soil DCGL levels. The cost optimization process supported selecting Basement Contamination DCGLs that are below the NRC screening values for standing building surfaces. At these levels, the resident farmer dose for contamination on basement surfaces was shown to be low (per Table 6-11) for any credible future land use.

6.4 Conceptual Model

The Conceptual Model for dose to the resident farmer critical group is different to some extent for each contaminated material due to the different physical characteristics of the materials and different source term radionuclides. The Conceptual Model for each material is described in detail in Section 6.6.

In general, the overall site Conceptual Model includes a resident farmer who lives on the site after release for unrestricted use, draws drinking water and irrigation water from the worst-case onsite well location, and derives a substantial percentage of annual food requirements from the onsite resident farm.

The hypothetical dose from each potentially contaminated material is evaluated independently. However, the total resident farmer dose results from the summation of the contributions from all materials and all pathways. The method for summing the doses and selecting DCGLs for all contaminated materials is provided in Section 6.7.

6.5 Environmental Media and Dose Pathways

6.5.1 Contaminated Materials

There are nine contaminated materials that could contribute to dose:

a. Embedded pipe
b. Buried pipe

c. Activated concrete/rebar
d. Groundwater
e. Surface Water
f. Basement surfaces
g. Surface soil
h. Deep soil
i. Forebay Sediment

6.5.2 Environmental Media

After considering radionuclide transfer from the nine contaminated materials, there are five environmental media that could deliver dose to the resident farmer. These are groundwater, surface soil, deep soil, surface water, and basement fill. Groundwater concentration may increase through the transfer of radionuclides from contaminated basement surfaces, activated concrete/rebar, deep soil, and embedded pipe. Note that the "groundwater" environmental medium includes contributions from water contained in building basements as well as other sources. Basement fill may also become slightly contaminated through the transfer of contamination from basement surfaces, embedded piping, and activated concrete/rebar. Table 6-1 indicates which environmental media are affected by the transfer of radionuclides from contaminated materials.

The residual contamination in the Forebay sediment is not transferred to any of the five environmental media and is evaluated independently. Therefore, Forebay sediment is not included in Table 6-1.

6.5.3 Dose Pathways

The five environmental media listed in Table 6-1 deliver dose to the resident farmer through one or more of the following dose pathways: 1) drinking water; 2) direct exposure; 3) ingesting soil, plants, animals, or fish; and 4) inhaling resuspended soil. These pathways are consistent with those listed in NUREG-1549 for the resident farmer. A given environmental medium will not contribute dose through all pathways.

Table 6-2 lists the dose pathways applicable to each environmental medium. Note that groundwater contributes to the plant and animal pathways through irrigation.

6.5.4 Radionuclide Concentrations in Environmental Media

To calculate the dose from each pathway the radionuclide concentrations in each environmental medium must be calculated. The concentrations in the surface soil, deep soil, and surface water can be used directly in the dose assessment since there is no contribution from other contaminated materials. However, the final concentrations in groundwater and basement fill, and the resulting dose, will depend on the transfer of contamination from other materials. Final concentrations in the five environmental media are calculated by summing contributions from various materials as listed below.

The contaminated materials that contribute to each of the environmental media are summarized below. The materials in brackets are those requiring transfer evaluations.

- Groundwater Concentration = [basement surface contamination] + [embedded pipe] + [activated concrete/rebar] + [deep soil] + [buried pipe] + existing groundwater concentration

- Basement Fill Concentration = [basement surface contamination] + [embedded pipe] + [activated concrete/rebar]
- Surface Soil Concentration = surface soil concentration

- Deep Soil Concentration = [buried pipe] + deep soil concentration

- Surface Water Concentration = surface water concentration

Table 6-1
Environmental Media Affected by Transfer from Contaminated Materials

	Ground Water	Surface Soil	Deep Soil	Surface Water	Basement Fill
Basement Contamination	X				X
Surface Soil		X			
Deep Soil	X		X		
Groundwater	X				
Embedded pipe	X				X
Surface Water				X	
Activated concrete/rebar	X				X
Buried Pipe	X		X		

Table 6-2
Environmental Media and Dose Pathways for the Resident Farmer Scenario

	Direct Radiation	Drinking Water	Plant, Animal, Soil Ingestion	Inhalation	Fish Ingestion
Surface Soil	X		X	X	
Deep Soil	X				
Basement Fill	X				
Groundwater		X	X*	X*	
Surface Water		X			X

* These pathways result through irrigation

6.6 Material Specific Dose Assessment Methods and Unitized Dose Factors

Each material has unique characteristics that must be considered when developing the conceptual and mathematical model for dose assessment. This section provides the dose assessment methods and results for each material in a unitized format by expressing the dose as a function of unit concentrations such as 1 dpm/100 cm^2 or 1 pCi/g. The unitized format facilitates the summation of doses from all materials and the selection of material specific DCGLs (see Section 6.7).

6.6.1 Contaminated Basement Surfaces

a. Conceptual Model

The Dose Model for contaminated basement surfaces assumes that the buildings are demolished to three feet below grade. The remaining basements are then decontaminated as necessary, filled with a suitable material (current plans call for fill with Bank Run Sand or flowable fill) and the area restored to grade, which results in a three-foot cover over the top of the filled basements. After the site is restored, rainwater and groundwater infiltrate into the basements and occupy the void space in the fill material. The available void space volume is a function of the fill material porosity.

The entire inventory of contamination on the basement surfaces, including the concrete and steel liner, is assumed to be instantaneously released and mixed with the water that has infiltrated into the basements. In this context, "surface" is intended to include all radioactivity, at all depths (this does not include activated concrete, which is treated as a separate material). Analyses of Maine Yankee concrete have indicated that, on average, the contamination is about 1 mm deep in the concrete. The liner contamination should be true surface contamination, i.e., not at any significant depth.

Using a mass balance approach, the radionuclides that are released from the surfaces are assumed to instantaneously reach equilibrium between the water, fill, and concrete. The relative equilibrium concentrations in the water, fill, and concrete are a function of the material Kd, mass, and porosity.

The critical group is the resident farmer who is assumed to drill a domestic water well into the worst case basement, i.e., that with the highest basement surface area to volume ratio. The amount of activity available for release is assumed to be directly proportional to the surface area of contaminated material. Therefore, the highest surface area/volume ratio results in the maximum radionuclide inventory and maximum concentrations in the water, fill, and concrete. The resident farmer is also assumed to occupy the land immediately above the basement, which maximizes direct exposure through the 3-foot cover. (Since the resident farmer is assumed to receive dose from exposure to surface soil based on 100% stay-time, the additional direct dose from basement fill is a conservative addition to dose. Thus, no credit is taken overall for the absence or presence of the 3 foot cover.)

The conceptual model results in three dose pathways to the resident farmer: 1) drinking water from the well; 2) irrigating with water from the well; and 3) direct radiation from radionuclides in the fill.

b. Mathematical Model

A mathematical model was developed to calculate the equilibrium radionuclide concentrations in the basement water, fill, and concrete after the infiltration of rainwater and groundwater. Contamination is assumed to diffuse into and re-adsorb on concrete surfaces since concrete is a porous media. The re-adsorption on the steel liner is expected to be less than the concrete and is considered to be bounded by the concrete analysis. The mathematical model includes calculations to determine the resident farmer dose from drinking water derived from a well drilled directly into the basement fill, irrigating with the water, and being directly exposed to the covered fill. The model is intended to be a simple, conservative, screening approach.

The radionuclide inventory, water volume, fill volume, and concrete volume subject to re-adsorption are the quantities required to determine the equilibrium radionuclide concentrations in the three materials. The initial condition of the model is that a volume of water has infiltrated into the basement that is equal to the annual volume required for drinking, domestic use, and irrigation by the resident farmer. As stated above, the well is placed directly into the basement fill containing the water. From this initial condition the volumes and masses of the three materials, and the maximum radionuclide inventory released to the water, can be calculated.

The annual resident farmer well-water usage is assumed to be 738 m^3 (justification provided below). This implies that the fill volume is 738 m^3 divided by the porosity of the soil, which is assumed to be 0.3 (justification provided below). Therefore, the model fill volume is 2460 m^3. This is the minimum fill volume required to contain the annual resident farmer water volume. Depending on the infiltration rate, smaller fill volumes could supply the required 738 m^3/y water volume, but this would result in slightly lower average annual concentrations. Assuming a model volume of 2460 m^3, and no dilution through infiltration recharge, is the most conservative approach.

The actual basement open volumes of the PAB, Spray, and Fuel buildings are less than 2460 m^3, but the containment basement volume is greater, i.e., 8217 m^3. The larger containment volume has no effect on the result since the additional hypothetical water volume does not affect the radionuclide concentrations in the water, or the assumed annual water use. In fact, as explained below, using actual containment basement dimensions, including volume and surface area, would reduce water concentrations by a factor of 3.7 since the surface area to volume ratio for the containment basement is lower than that used in the model. The effect of surface area to volume ratio and the rationale for selecting the value used in the model are described below.

The basement surface area to open volume ratios have a direct effect on the results and are necessary for determining two parameters. The most important affected parameter is the maximum radionuclide inventory. Less important, but also related, is the volume of concrete available for re-adsorption of radionuclides. Using the maximum surface area/volume ratio from the four basements maximizes the radionuclide inventory and the resulting water, fill, and concrete concentrations.

The maximum ratio of concrete surface area/basement open volume of 1.7 m^2/m^3 is found in the Spray building basement. The surface area/volume ratios for the Containment, PAB, and Fuel buildings are 0.46 m^2/m^3, 1.03 m^2/m^3, and 0.49 m^2/m^3, respectively. Using the maximum ratio of 1.7 m^2/m^3 results in conservative dose calculations for the Containment, PAB, and Fuel buildings by factors of 3.7, 1.65, and 3.5 respectively. If necessary, as the project proceeds, Maine Yankee may use building-specific surface area/volume ratios based on the data presented in Section 6.6.1(d)(2) to calculate building-specific DCGLs.[1]

Multiplying the 1.7 m^2/m^3 ratio by the fill volume (2460 m^3) results in the maximum contaminated surface area that could contribute to the source term for a given 738 m^3 of water. Accordingly, the maximum surface area in the model would be 4182 m^2, which exceeds the actual surface area of any of the building basements. This occurs because the 1.7 m^2/m^3 ratio is from the Spray building and the maximum surface area of 3775 m^2 is in

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1 For containment, the building specific SA/V ratio of 0.46 m^2/m^3 is used for a model surface area of 1130 m^2

the Containment building. However, consistent with a conservative screening approach, and to maintain the correct mathematical relationships between porosity, annual water volume, and surface area, the 4182 m^2 surface area will be used in the model. Note that using 3775 m^2 would reduce the available source term and thereby reduce water concentrations.

Assuming that the water penetrates to a depth of 1 mm in the concrete, the concrete volume available to re-adsorb radionuclides from contaminated water is 4.2 m^3. The 1 mm depth is based on analyses of contaminated Maine Yankee concrete. Although the conditions are different, i.e., water saturation after decommissioning versus periodic wet contamination events during operation, the penetration of water into the concrete after the basements are filled with water is also assumed to be 1 mm. This is considered a conservative assumption since increasing the concrete penetration depth will decrease the concentrations in the fill and in the water.

The model uses two approximations related to re-adsorption onto concrete that have a very small effect on the final results. First, the fill volume is calculated assuming all of the 738 m^3 water volume is contained in the fill, not mixed between the fill and concrete. An exact solution would require consideration of both the fill and concrete volumes simultaneously. However, the affected concrete volume is very low and the corresponding water volume in the concrete is about 1 m^3. This is less than 1% of the 738 m^3 total and is insignificant. Second, the porosity of 0.3 is assumed to apply to both fill and concrete. The same porosities are used in the model in order to produce the simplified solution provided in Equation 7. However, site-specific measurements indicate that the actual concrete porosity is 0.15. Using a porosity of 0.15 would decrease the volume of water in the concrete to about 0.5 m^3. An exact solution to these two approximations would have a very small effect on the results and is an unnecessary level of detail considering the conservative screening approach used in the model.

The approach assumes uniform mixing among the soil, water, and concrete. Uniform mixing within the fill is not unreasonable considering the surface area to volume ratio of 1.7 m^2/m^3. Assuming a planar geometry, this means that the water is required to mix over a distance of 0.6 m in the backfill. Although assuming planar geometry is a simplification, it demonstrates that water mixing over long distances in the fill is not intrinsic to the validity of the screening model.

The calculations for determining the equilibrium concentrations in the basement water, fill, and concrete are based on a mass balance approach. The total mass in the system, M_t, is the sum of the mass in the water (M_w), the mass sorbed to the fill (M_b), and the mass sorbed to the concrete (M_c). For these calculations, mass is expressed as activity, A. The total activity, A_t, is the total radionuclide inventory in the 4182 m^2 basement concrete surface under consideration. Equations (1) through (7) described below are solved for each radionuclide in the Maine Yankee Radionuclide Mixture.

$$A_t = A_w + A_f + A_c \tag{1}$$

Where: A_t is total activity (pCi)
A_w is the total activity in water (pCi)
A_f is the total activity in the fill (pCi)
A_c is the total activity in the concrete (pCi)

The activity in the water is defined as:

$$A_w = \quad C V_t \tag{2}$$

Where: is the porosity of the fill and concrete
C is the concentration in solution (pCi/l) and,
V_t is the total system volume (sum of the volume of fill and concrete, m^3).

At equilibrium the activity adsorbed to the fill and concrete is directly proportional to the concentration in the water. The proportionality constant used in these calculations is the distribution coefficient, Kd, and has units of cm^3/g. Distribution coefficients are widely accepted measures of sorption onto the solid phase, and the solid/liquid phase ratio, and are accepted for use in risk assessments by national and international regulatory agencies and scientific organizations including the U.S. Nuclear Regulatory Commission and the U.S. Environmental Protection Agency.

The activity adsorbed on the fill and the concrete can be represented as:

$$A_f = \ _f\, Kd_f C\, V_f \qquad (3)$$

Where: $_f$ is fill bulk density (g/cm^3)
Kd_f is fill distribution coefficient
C is water concentration(pCi/l)
V_f is fill volume (m^3)

and

$$A_c = \ _c\, Kd_c\, C\ \ V_c \qquad (4)$$

Where: $_c$ is concrete bulk density (g/cm^3)
Kd_c is concrete distribution coefficient
C is water concentration (pCi/l)
V_c is concrete volume (m^3)

The bulk density of the fill is assumed to be 1.5 g/cm^3 based on analyses of potential fill (reference provided below). For the concrete, a site-specific value of 2.2 g/cm^3 was used (reference provided below). V is the volume of the solid phase; V_f is 2460 m^3 and V_c is 4.2 m^3.

Combining the terms from Equations (2), (3), and (4) gives:

$$A_t = \ C\, V_t + \ _f\, Kd_f\, C\, V_f + \ _c\, Kd_c\, C\ \ V_c \qquad (5)$$

Multiplying the second and third terms by (V_t)/(V_t), i.e., 1, and rearranging gives:

$$A_t = \ C\, V_t + (\ V_t\, C)(\ _f\, Kd_f V_f)\,/(\ V_t\,) + (\ V_t\, C)(\ _c\, Kd_c V_c)/(\ V_t) \qquad (6)$$

Recognizing from Equation (1) that the term, C V_t is the activity in the water phase, A_w, allows Equation 6 to be rewritten as:

$$A_t = A_w(1 + \ _f\,(Kd_f/\ )(V_f/V_t) + \ _c\,(Kd_c/\ )(V_c/V_t)) \qquad (7)$$

To calculate the water concentration, drinking water dose, concentration in the fill, and concentration on the concrete surfaces, Equation (7) is first solved for A_w. All of the terms in Equation (7) are known except A_w. The water concentration, C, is then calculated using Equation (2). After solving for C, the backfill and concrete concentrations are calculated using Equations (3) and (4).

c. Dose Calculations

The concentrations in the basement water and fill are used to calculate dose. There are three dose pathways to the resident farmer after the fill is placed in the basements, the three-foot cover is completed, and water infiltrates the basements. These are drinking water dose, irrigation dose, and direct dose. The dose calculations are described in Equations (8) through (10). The equations are used to calculate dose for each radionuclide in the Maine Yankee mixture.

There will be no ingestion or inhalation associated with the fill because of the presence of the cover. Ingestion or inhalation could occur if the fill were excavated at some time in the future. To account for this possibility, the projected basement fill concentration is limited to ensure that the concentration will not exceed the surface soil DCGL and that the dose will not increase over that calculated with the earthen cover in place. In fact, the hypothetical dose would decrease if the fill were excavated at some time in the future.

1. Drinking Water Dose

Drinking water dose is calculated from the radionuclide concentrations in the basement water. As shown in Table 6-1, the basement water is one of several contributors to drinking water dose. The annual water intake is assumed to be 478 L/y consistent with the default values in the NRC screening code, DandD, Version 1. Dose conversion factors are taken from Federal Guidance Report No. 11.

$$Dose_{dw} = (\ C\ pCi/l)(478\ L/y)(DCF\ mrem\text{-}y/pCi) \qquad (8)$$

Where: C is water concentration in pCi/L
DCF is FGR 11 dose conversion factor

2. Irrigation Dose

Including irrigation dose is conservative because irrigation in Maine is uncommon due to relatively high annual precipitation. However, consistent with a screening approach it is included. The irrigation rate is assumed to be 0.274 $L/m^2/d$ (justification provided below). The source of the water is the resident farmer well placed in the building basement. The annual irrigation volume is mixed in a 15 cm depth of soil, which is consistent with the NRC DandD model as described in NUREG-5512, Volume 1. The dose from the resulting soil concentrations were calculated using the NRC screening values in NUREG-1727, Table C2.3 , converted to mrem/y per pCi/g.

$$Dose_{irrigation} = (C_{soil}\ pCi/g)(NUREG\text{-}1727\ mrem/y\ per\ pCi/g) \quad (9)$$

Where: $Dose_{irrigation}$ is the annual dose from irrigation (mrem/y)
C_{soil} is soil concentration in pCi/g
(NUREG-1727) is the soil screening value from NUREG-1727, Table C2.3 converted to mrem/y per pCi/g

$$C_{soil} = \frac{(pCi/L\ in\ water)(0.274\ L/m^2/d)(365\ d)(1\ m^2)}{(1m^2)(0.15\ m)(1E{+}06\ cm^3/m^3)(1.6\ g/cm^3)} \quad (10)$$

3. Direct Dose

The direct dose was calculated using the Microshield code assuming a three-foot soil cover, 10,000 m^2 area, and 5.8 m depth. The 5.8 m depth represents the deepest basement, i.e., containment. The Microshield result for "Deep Dose Equivalent, Rotational Geometry," was used and is generally referred to as "exposure." The resulting exposure rate was multiplied by the annual outdoor occupancy time of 964 hours (0.1101 x 365 days x 24 hr/day) from the NRC DandD, Version 1, screening code to calculate the annual direct exposure dose. The Microshield output reports are provided in Attachment 6-1.

d. Model Input Parameters

The following section describes and justifies the parameters used in the concentration and dose calculations.

1. Distribution Coefficients, Kd

Fill Kd values were either derived from literature (mean values) or from the results of analyses of site-specific fill materials. The site-specific Kd analyses were performed by Brookhaven National Laboratory (BNL) (results provided in Attachment 6-2). At this time, the most likely fill material is Bank Run Sand or flowable fill. Therefore, the average Kd's for Bank Run Sand or flowable fill from Attachment 6-2 were used in the model. Table 6-3 lists the fill Kd's, and the reference, for each radionuclide.

Concrete Kd values were either derived from literature or from the results of site-specific Kd analyses. The site-specific Kd analyses were performed by BNL (results provided in Attachment 6-3). Table 6-3 lists the concrete Kd's, and the reference, for each radionuclide. It is seen that for cement, a few Kd's were left blank. This indicates data were not available and a value of zero (0) was used in the calculations. A Kd of zero (0) maximizes the concentration in water. In addition, the Krupka reference did not contain Kd information for cobalt or iron. It was assumed that the Kd's for these two metals were the same as nickel. However, the overall effect of the concrete is small, regardless of Kd.

Table 6-3
Selected Kd Values (cm^3/g) for Basement Fill Model

Radionuclide	Mean Flowable Fill Kd	Reference for Mean Kd	Concrete Kd	Reference for Kd in cement
H-3	0		0	
Fe-55	25	Baes, Table 2.13	100	Krupka Table 5.1
Ni-63	128	Attachment 6-2	100	Krupka Table 5.1
Mn-54	50	Sheppard, Table A-1		

Table 6-3 Selected Kd Values (cm^3/g) for Basement Fill Model				
Radionuclide	Mean Flowable Fill Kd	Reference for Mean Kd	Concrete Kd	Reference for Kd in cement
Co-57	128	Attachment 6- 2	100	Krupka Table 5.1
Co-60	128	Attachment 6-2	100	Krupka Table 5.1
Cs-134	79	Attachment 6-2	3	Attachment 6-3
Cs-137	79	Attachment 6-2	3	Attachment 6-3
Sr-90	6	Attachment 6-2	1.0	Attachment 6-3
Sb-125	45	Sheppard, Table A-1		
Pu-238	550	Sheppard, Table A-1	5000	Krupka Table 5.1
Pu-239/240	550	Sheppard, Table A-1	5000	Krupka Table 5.1
Pu-241	550	Sheppard, Table A-1	5000	Krupka Table 5.1
Am-241	1900	Sheppard, Table A-1	5000	Krupka Table 5.1
Cm243/244	4000	Sheppard, Table A-1	5000	Krupka Table 5.1
C-14	5	Sheppard, Table A-1		
Eu-152	400	Onishi, Table 8.35		
Eu-154	400	Onishi, Table 8.35		

2. Maximum Surface Area to Volume Ratio

The building basements that will remain following demolition of site structures include the Containment, PAB, Spray and Fuel Building basements. The open-air volumes of the basements are 8217 m^3, 1584 m^3, 1136 m^3, and 837 m^3 respectively. This represents the volume of fill required in each basement. The wall and floor surface areas are 3775 m^2, 1637 m^2, 1883 m^2, and 409 m^2 respectively. The basement volumes and surface areas were determined in Maine Yankee calculation EC 01-00(MY). The maximum surface area to volume ratio of 1.7 m^2/m^3 is found in the Spray building basement. This ratio is used in the Unitized Dose Factor tables produced below (Tables 6-4, 5, 6A and 6B). The Containment building surface area to volume ratio of 0.46 m^2/m^3

was used in the dose assessment summation for activated concrete (Reference No. 6.10.8) and is shown in Attachment 6-13.

3. Porosity

The porosity of the fill material is assumed to be 0.3. The range of mean porosities for a wide variety of soil types are listed in NUREG-5512, Volume 3, "Residual Radioactive Contamination From Decommissioning. Parameter Analysis," Page 6-64, Table 6.41. The porosities listed in NUREG-5512 ranged from 0.36 to 0.49.

The projected dose from contaminated concrete in the basement fill model decreases with increasing porosity. However, the projected doses from the embedded pipe and activated concrete increase with increasing porosity. This is because the source term for embedded and buried piping is constant and the source term for contaminated concrete is a function of surface area. All three dose assessment models are conservative. However, the activated concrete and embedded piping source term assumptions are much more conservative than those used for the basement concrete and the resulting dose is a small fraction of that from contaminated concrete. Therefore, the porosity effect on the contaminated concrete dose is used to select a porosity at the lower end of the range, e.g., 0.3.

4. Annual Drinking Water Volume

The annual drinking water volume was assumed to be 478 l/y. This is the default volume from NRC DandD, Version 1 screening code.

5. Irrigation Rate and Annual Irrigation Volume

Annual irrigation volume was based on interviews with representatives of the Maine USDA-NRCS. The individuals contacted are documented in a memorandum provided in Attachment 6-4. The USDA representatives indicated that irrigation in Maine is uncommon, but that in drought years irrigation may occur. The Maine USDA representatives indicated that the drought irrigation rate for a family garden would not be expected to exceed 4-5 in/y (10 to 12 cm/y). The 10 cm/y rate was used in the model, which can be converted to 0.274 $l/m^2/d$. To calculate total annual volume, the 10 cm/y rate was multiplied by the default cultivated area of 2400 m^2 from the DandD screening model (NUREG-1727, Appendix C, Section 2.3.2). This results in the annual irrigation volume of 240,000 l/y.

6. Annual Domestic Water Use

Annual domestic water volume is derived from NUREG-5512, Volume 3, Page 6-37, Table 6-19. The per capita consumption rate for the State of Maine is listed as 124,422 l/y. Assuming a family of four, this corresponds to a total domestic water volume of 497,688 l/y. The assumption of four occupants is based on the land occupancy rate from NUREG-1727, Table D2, of 0.0004 persons/m^2 and an assumption that the resident farm size is 10,000 m^2.

7. Total Resident Farmer Annual Well Water Volume

The total annual volume of water from the resident farmer well is the sum of the domestic use plus irrigation use. Domestic use is 497,688 l/y and irrigation use is 240,000 l/y for a total of 737,688 l/y. A rounded value of 738 m^3/y was used in the model.

8. Concrete Density

Concrete density was determined by site-specific analysis to be 2.2 g/cm3 (Attachment 6-5).

9. Fill Material Density

Density of the possible fill material is 1.5 g/cm^3 (Attachment 6-2). This corresponds to Bank Run Sand.

10. Soil Density

Density of soil is 1.6 g/cm^3 based on an average of the densities of Bank Run Sand and Bank Run Gravel from Attachment 6-2. This average is assumed to be representative of the site soil, which is comprised primarily of backfill.

11. Dose Conversion Factors (DCFs)

The DCFs are in units of Committed Effective Dose Equivalent (CEDE) and are taken from Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Table 2.2, EPA-520/1-88-020.

12. Outdoor Occupancy Time

The DandD, Version 1, default value of 0.1101 y or 965 hr/y is used.

e. Unitized Dose Factors for Contaminated Basement Surfaces

Using Equations 1-10 above, the radionuclide concentrations in basement water, fill, and concrete, and the dose to the resident farmer were calculated using a simple spreadsheet application. The activity of each radionuclide in the Maine Yankee mixture for contaminated surfaces was set to1 dpm/100 cm^2 of surface area. The surface was assumed to be concrete for the purpose of the calculation to evaluate the potential effect of re-adsorption on concrete. The spreadsheet output and the resulting unitized dose factors are provided in Table 6-4 (see next page).

6.6.2 Activated Basement Concrete/Rebar

a. Conceptual Model

Activated concrete and rebar is present in the ICI sump area in the containment building. The current plan is to remediate activated concrete down to the containment building liner and any rebar associated with this concrete. The walls and floors consist primarily of concrete with rebar being a small percentage. Characterization results indicate that the total activity concentration in rebar is about 1.9 times higher than the concrete surrounding the rebar. In addition, the radionuclide mixtures for concrete and rebar differ as indicated in Table 2-9. However, as shown in Attachment 6-13, the calculated dose from the rebar is less than the dose from the surrounding concrete (see Table 6-11 for activated concrete dose), accounting for both the higher relative concentration and the rebar radionuclide mixture. Therefore, the walls and floors are conservatively assumed to be comprised entirely of activated concrete in the dose calculation.

Table 6-4
Contaminated Basement Surfaces Unitized Dose Factors

Key Parameters

Porosity	0.30		Fill Volume	2460.0	m^3	Annual Total Well Water Vol	738.0	m^3
Bulk Density	1.50	g/cm^3	Surface Area/Open Vol	1.70	m^2/m^3	Irrigation Rate	0.274	L/m^2-d
Yearly Drinking Water	478.0	L/yr	Concrete Volume	4.18	m^3	Surface Soil Depth	0.15	m
Wall Surface Area	4182.0	m^2	Concrete Density	2.20	g/cm^3			

DOSE CALCULATION FACTORS				Source Term		Kd		WATER, FILL, CONCRETE CONCENTRATION				CONTAMINATED CONCRETE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem per pCi	Microshield mrem/y per pCi/g	Inventory dpm/100 cm2	Inventory pCi	Kd Fill cm3/gm	Kd Concrete cm3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	1.00E+00	1.88E+05	6.02E+01	1.00E+00	3.02E+02	8.45E-04	5.09E-05	8.45E-07	Sr-90	5.74E-05	5.52E-06	0.00E+00	6.29E-05
Cs-134	4.39E+00	7.33E-05	6.09E-05	1.00E+00	1.88E+05	7.91E+01	3.00E+00	3.96E+02	6.44E-04	5.09E-05	1.93E-06	Cs-134	2.26E-05	1.26E-06	3.10E-09	2.38E-05
Cs-137	2.27E+00	5.00E-05	1.20E-05	1.00E+00	1.88E+05	7.91E+01	3.00E+00	3.96E+02	6.44E-04	5.09E-05	1.93E-06	Cs-137	1.54E-05	6.49E-07	6.11E-10	1.60E-05
Co-60	6.58E+00	2.69E-05	6.30E-04	1.00E+00	1.88E+05	1.28E+02	1.00E+02	6.40E+02	3.98E-04	5.09E-05	3.98E-05	Co-60	5.12E-06	1.16E-06	3.20E-08	6.32E-06
Co-57	1.67E-01	1.18E-06	2.80E-08	1.00E+00	1.88E+05	1.28E+02	1.00E+02	6.40E+02	3.98E-04	5.09E-05	3.98E-05	Co-57	2.25E-07	2.96E-08	1.42E-12	2.54E-07
Fe-55	2.50E-03	6.07E-07	0.00E+00	1.00E+00	1.88E+05	2.50E+01	1.00E+02	1.27E+02	2.01E-03	5.01E-05	2.01E-04	Fe-55	5.82E-07	2.23E-09	0.00E+00	5.84E-07
H-3	2.27E-01	6.40E-08	0.00E+00	1.00E+00	1.88E+05	0.00E+00	0.00E+00	1.00E+00	2.55E-01	0.00E+00	0.00E+00	H-3	7.80E-06	2.57E-05	0.00E+00	3.35E-05
Ni-63	1.19E-02	5.77E-07	0.00E+00	1.00E+00	1.88E+05	1.28E+02	1.00E+02	6.40E+02	3.98E-04	5.09E-05	3.98E-05	Ni-63	1.10E-07	2.11E-09	0.00E+00	1.12E-07

With the exception of the source term calculation, and the realistic release rate to the basement, the conceptual model for activated concrete is identical to the conceptual model for contaminated basement surfaces described above. See Reference 6.10.7 for a discussion of the activated concrete dose model.

b. Dose Factors for Activated Concrete

Although activated concrete is present at depth beneath the surface, the dose calculation for activated concrete is based on a total activity (sum of all radionuclides) in the floors and walls of the ICI sump. The total inventory, i.e., source term, includes the radionuclides in the entire volume of activated concrete, including surface and subsurface. The total inventory was determined to be 4.88E+08 pCi as described in Reference 6.10.7.

To determine the inventory of each radionuclide, the total 4.88E+08 pCi inventory must be multiplied by the radionuclide fraction in the activated concrete mixture. The resulting radionuclide specific inventories are input to the "inventory" column in the spreadsheet developed for the contaminated basement surfaces. All of the resulting water, fill, and concrete concentrations and dose calculations are identical to those described for the contaminated basement surfaces in Section 6.6.1.

Table 6-5 - Deleted

6.6.3 Embedded Pipe

a. Conceptual Model

Embedded pipe includes pipes that are encased in the basement concrete walls or floors that will remain after demolition and remediation. The conceptual dose model is identical to that described for contaminated basement surfaces. However, analogous to activated concrete, the source term calculation includes the entire radionuclide inventory contained in all embedded piping, regardless of location. The entire inventory is assumed to be instantaneously released into the worst case 738 m^3 of basement water.

b. Unitized Dose Factors for Embedded Pipe

The total embedded pipe inventory is calculated assuming a unit contamination level of 1 dpm/100 cm^2 over the entire internal surface area of all embedded pipe remaining after decommissioning. A list of the embedded piping planned to remain after decommissioning is provided in Attachment 6-7. The internal surface area of the embedded piping is 154 m^2. Assuming a unit inventory of 1 dpm/100 cm^2 the total inventory was determined to be 6.95E+03 pCi.. The 6.95E+03 pCi inventory applies to each radionuclide at a "unit" concentration of 1 dpm/100 cm^2. Based on this value, an inventory was calculated and input into the spreadsheet developed for the contaminated basement surfaces. The spreadsheet "inventory" column input was calculated by multiplying the pipe surface contamination level, in this case a unitized level of 1 dpm/100 cm^2, by the 6.95E+03 pCi unit inventory. Because two distinct areas (Embedded Spray Pump Piping and BOP Embedded Piping) were created to address embedded piping, two different DCGL calculations (and spreadsheets) were created. Each spreadsheet addresses separate unit inventories that sum to the above total inventory (Spray Pump and BOP embedded inventories are 1.19E+03 and 5.75E+03 respectively). These forms facilitate the use of the spreadsheets in the total dose and DCGL calculations provided in Section 6.7. All of the resulting water, fill, and concrete concentrations, and dose calculations are identical to those described for the contaminated basement surfaces in Section 6.6.1.

The BOP Embedded Piping and Embedded Spray Pump Piping spreadsheets are provided in Tables 6-6A and 6-6B. The results represent the unit dose factors for embedded piping assuming a source term of 1 dpm/100 cm^2, for each radionuclide, on the internal surfaces of the associated pipe.

Table 6-6A
BOP Embedded Piping Unitized Dose Factors
Key Parameters

Porosity	0.30		Fill Volume	2460.0	m^3	Surface Soil Depth	0.15	m
Bulk Density	1.50	g/cm^3	Surface Area/Open Vol	1.70	m^2/m^3	Irrigation Rate	0.274	L/m^2-d
Yearly Drinking Water	478.0	l/yr	Concrete Volume	4.18	m^3	Annual Total Well Water Vol	738	m^3
Wall Surface Area	4182.0	m^2	Concrete Density	2.20	g/cm^3	Embedded Pipe Conversion Factor	5754.5	pCi per dpm/100 cm^2
						Total Inventory	1.00E+00	dpm/100 cm^2

DOSE CALCULATION FACTORS				Source Term		Kd		WATER, FILL, CONCRETE CONCENTRATION				EMBEDDED PIPE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Inventory dpm/100 cm2	Inventory pCi	Kd Fill cm3/gm	Kd Concrete cm3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	1.00E+00	5.75E+03	6.02E+01	1.00E+00	3.01E+02	2.58E-05	1.55E-06	2.58E-08	Sr-90	1.75E-06	1.69E-07	0.00E+00	1.92E-06
Cs-134	4.39E+00	7.33E-05	6.09E-05	1.00E+00	5.75E+03	7.91E+01	3.00E+00	3.96E+02	1.97E-05	1.56E-06	5.90E-08	Cs-134	6.89E-07	3.84E-08	9.47E-11	7.27E-07
Cs-137	2.27E+00	5.00E-05	1.20E-05	1.00E+00	5.75E+03	7.91E+01	3.00E+00	3.96E+02	1.97E-05	1.56E-06	5.90E-08	Cs-137	4.70E-07	1.98E-08	1.87E-11	4.90E-07
Co-60	6.58E+00	2.69E-05	6.30E-04	1.00E+00	5.75E+03	1.28E+02	1.00E+02	6.40E+02	1.22E-05	1.55E-06	1.22E-06	Co-60	1.56E-07	3.56E-08	9.79E-10	1.93E-07
Co-57	1.67E-01	1.18E-06	2.80E-08	1.00E+00	5.75E+03	1.28E+02	1.00E+02	6.40E+02	1.22E-05	1.55E-06	1.22E-06	Co-57	6.86E-09	9.03E-10	4.35E-14	7.76E-09
Fe-55	2.50E-03	6.07E-07	0.00E+00	1.00E+00	5.75E+03	2.50E+01	1.00E+02	1.27E+02	6.13E-05	1.53E-06	6.13E-06	Fe-55	1.78E-08	6.81E-11	0.00E+00	1.78E-08
H-3	2.27E-01	6.40E-08	0.00E+00	1.00E+00	5.75E+03	0.00E+00	0.00E+00	1.00E+00	7.78E-03	0.00E+00	0.00E+00	H-3	2.38E-07	7.85E-07	0.00E+00	1.02E-06
Ni-63	1.19E-02	5.77E-07	0.00E+00	1.00E+00	5.75E+03	1.28E+02	1.00E+02	6.40E+02	1.22E-05	1.55E-06	1.22E-06	Ni-63	3.35E-09	6.43E-11	0.00E+00	3.42E-09

Table 6-6B
Embedded Spray Pump Piping Unitized Dose Factors

Key Parameters

Porosity	0.30		Fill Volume	2460.0	m^3	Surface Soil Depth	0.15	m
Bulk Density	1.50	g/cm^3	Surface Area/Open Vol	1.70	m^2/m^3	Irrigation Rate	0.274	L/m^2-d
Yearly Drinking Water	478.0	l/yr	Concrete Volume	4.18	m^3	Annual Total Well Water Vol	738	m^3
Wall Surface Area	4182.0	m^2	Concrete Density	2.20	g/cm^3	Embedded Pipe Conversion Factor	1191.7	pCi per dpm/100 cm^2
						Total Inventory	1.00E+00	dpm/100 cm^2

DOSE CALCULATION FACTORS				SOURCE TERM		Kd		WATER, FILL, CONCRETE CONCENTRATION				EMBEDDED PIPE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Inventory dpm/100 cm2	Inventory pCi	Kd Fill cm3/gm	Kd Concrete cm3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	1.00E+00	1.19E+03	6.02E+01	1.00E+00	3.01E+02	5.35E-06	3.22E-07	5.35E-09	Sr-90	3.63E-07	3.50E-08	0.00E+00	3.98E-07
Cs-134	4.39E+00	7.33E-05	6.09E-05	1.00E+00	1.19E+03	7.91E+01	3.00E+00	3.96E+02	4.07E-06	3.22E-07	1.22E-08	Cs-134	1.43E-07	7.95E-09	1.96E-11	1.51E-07
Cs-137	2.27E+00	5.00E-05	1.20E-05	1.00E+00	1.19E+03	7.91E+01	3.00E+00	3.96E+02	4.07E-06	3.22E-07	1.22E-08	Cs-137	9.73E-08	4.11E-09	3.87E-12	1.01E-07
Co-60	6.58E+00	2.69E-05	6.30E-04	1.00E+00	1.19E+03	1.28E+02	1.00E+02	6.40E+02	2.52E-06	3.22E-07	2.52E-07	Co-60	3.24E-08	7.37E-09	2.03E-10	4.00E-08
Co-57	1.67E-01	1.18E-06	2.80E-08	1.00E+00	1.19E+03	1.28E+02	1.00E+02	6.40E+02	2.52E-06	3.22E-07	2.52E-07	Co-57	1.42E-09	1.87E-10	9.01E-15	1.61E-09
Fe-55	2.50E-03	6.07E-07	0.00E+00	1.00E+00	1.19E+03	2.50E+01	1.00E+02	1.27E+02	1.27E-05	3.17E-07	1.27E-06	Fe-55	3.68E-09	1.41E-11	0.00E+00	3.70E-09
H-3	2.27E-01	6.40E-08	0.00E+00	1.00E+00	1.19E+03	0.00E+00	0.00E+00	1.00E+00	1.61E-03	0.00E+00	0.00E+00	H-3	4.93E-08	1.63E-07	0.00E+00	2.12E-07
Ni-63	1.19E-02	5.77E-07	0.00E+00	1.00E+00	1.19E+03	1.28E+02	1.00E+02	6.40E+02	2.52E-06	3.22E-07	2.52E-07	Ni-63	6.95E-10	1.33E-11	0.00E+00	7.08E-10

6.6.4 Surface Soil

a. Conceptual Model

Surface soil includes all soil within the first 15 cm of the ground surface. The NRC screening values for soil from NUREG-1727, Table C2.3, are used for the unitized dose calculations Therefore, the conceptual model is identical to that described in NUREG-1727. The screening values include the dose from all pathways. The groundwater contribution to the screening value dose is negligible and is entered as zero. The screening values are used because they were specifically generated by NRC to be conservative calculations of the resident farmer dose and are recommended for use in NUREG-1727.

Verification Conditions (for Surface Soil Screening Values). NUREG-1727, NMSS Decommissioning Standard Review Plan, Appendix C, describes the justification necessary to allow direct use of these screening. Per the NUREG, the following conditions must be satisfied:

10. The initial residual radioactivity (after decommissioning) is contained in the top layer of the surface soil [that is, approximately 6 inches (15cm)].
11. The unsaturated zone and the groundwater are initially free of contamination.
12. The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate.

The above conditions are satisfied for the Maine Yankee site.

Condition One. The direct use of these screening values is only for surface soil (approx. 6 inches). Section 6.6.5 calculated a dose from deep soil (that is, greater than 6 inches) separate from the use of the surface soil screening values. (See Section 6.6.5)

Condition Two. Maine Yankee does not use the surface soil screening values to address potential site groundwater contamination from H-3. H-3 presence in the groundwater and surface water is assumed based upon the highest measured readings and is covered by separate dose assessments. (See Sections 6.6.6 and 6.6.7)

Condition Three. The soils at Maine Yankee that are in areas currently containing nuclides elevated above background, and those soils that are planned to be used to fill the foundations are bank run sand and gravel. The Adams or Hinckley USDA Soil Series would provide the closest approximation. The minimum saturated vertical hydraulic conductivity of these soils is 0.001 cm/sec or 1.417 inches per hour. Average saturated hydraulic conductivity rates would be about 10 times this, or 14 inches per hour. Infiltration capacity is based on land cover type, antecedent moisture condition prior to a rainfall or snowmelt event, and the rate of water supply available for infiltration. The permanent water table at the Maine Yankee site in the area of interest is approximately elevation 10 to 15 feet above Mean Sea Level, indicating a distance of 6 to 11 feet from the existing ground surface to the average water table position. Therefore, this much of the sand fill will be unsaturated. Infiltration capacity is limited by the unsaturated hydraulic conductivity of the soil. The unsaturated hydraulic conductivity of the sand fill is typically from 1/10 to 1/100 of the saturated hydraulic conductivity. Precipitation rates rarely exceed one inch per hour in Maine. Therefore, because the typically expected maximum precipitation rate is less than the minimum saturated hydraulic conductivity, and because the fill is unsaturated for 6 or more feet down and unable to transmit water downward at a rate exceeding the saturated vertical hydraulic conductivity, infiltration rates in the fill must be less than the saturated vertical hydraulic conductivity.

Soil types on the Maine Yankee site are representative of those assumed in the soil screening model. These soil types include: silt loams derived from glaciomarine sediments, fine sandy loams derived from glacial till, and fill that has a wide textural variation. However, the primary fill in the immediate plant area is a sand or loamy sand. The silt loams are most typical over the undisturbed portions of the site. The exceptions are in the knoll and ridge areas where bedrock is exposed or shallow where the fine sandy loams predominate. Fill areas surrounding the plant buildings are sand or loamy sand. Fill areas north of the 345 KV yard tend to have a silt loam surface covering. The most likely foundation fill material will be bank run sand. (See Section 6.6.1d.)

a. Unitized Dose Factors for Surface Soil

The unitized dose factors are generated for each radionuclide directly from the NUREG-1727 screening values by converting the values to mrem/y per pCi/g. Table 6-7 provides the "Surface Soil" unitized dose spreadsheet. The results represent the dose from a unit source term if 1 pCi/g for each radionuclide in the soil mixture.

Table 6-7
Surface Soil Unitized Dose Factors 1.0 pCi/g Cs-137

Key Parameters:

Soil Depth 0.15 m

DOSE CALCULATION FACTORS		SOURCE TERM	SURFACE SOIL ANNUAL DOSE
Nuclide	**NUREG-1727 mrem/y per pCi/g**	**Soil pCi/g**	**Total Dose mrem/yr**
Cs-137	2.27E+00	1.00E+00	2.27E+00
Co-60	6.58E+00	1.00E+00	6.58E+00
H-3	2.27E-01	1.00E+00	2.27E-01
Ni-63	1.19E-02	1.00E+00	1.19E-02

6.6.5 Deep Soil

a. Conceptual Model

Deep soil is defined as soil at depths greater than 15 cm. A separate calculation is required for deep soil because the NRC soil screening values apply to the top 15 cm of soil only. The resident farmer is exposed to deep soil through the direct exposure pathway and groundwater. The deep soil could be brought to the surface at some time in the future through the activities of the resident farmer. Therefore, the deep soil concentration will be limited to the surface soil DCGL.

The conceptual model for deep soil assumes a 15 cm layer of uncontaminated soil for the purpose of calculating the additional direct radiation exposure. The 15 cm cover represents the layer of surface soil. The direct radiation from residual contamination in the top 15 cm soil layer was accounted for in the surface soil screening values. A very large volumetric source term was assumed, i.e., 28,500 m^3, for the purpose of conservatively determining the potential for groundwater contamination from deep soil. This is considered a bounding source term volume and essentially represents the entire volume of soil within the restricted area down to bedrock. After remediation and backfill, the actual remaining volume of deep soil with any significant contamination will be a very small fraction of 28,500 m^3.

b. Unitized Dose Factors for Deep Soil

Unitized dose factors were calculated using unit concentrations of each of the radionuclides in the soil mixture. The contribution from direct radiation was calculated using the Microshield code assuming a 15 cm cover and default values from DandD for indoor occupancy time (0.6571 y), outdoor occupancy time (0.1101 y), and external radiation shielding factor (0.5512). The Microshield output reports, deep dose direct radiation calculations, and resulting dose factors are provided in Attachment 6-8.

The maximum groundwater concentrations were calculated using RESRAD and unit concentrations of each radionuclide in the mixture. The RESRAD groundwater parameters used in the analysis are listed in Table 6-8. Only the parameters pertaining to groundwater transport are listed since the groundwater concentration is the only RESRAD output used. The RESRAD parameters affecting groundwater transport were reviewed by a local hydrologist who is very familiar with the site hydrogeological characteristics (Mr. Robert Gerber, P.E. and Certified Geologist). The parameters in Table 6-3 are recommended site-specific values. The Kd's were derived from Maine Yankee analyses of Bank Run Sand and Bank Run Gravel. The average of these two materials was assumed to represent the material used to backfill the site during plant construction. Finally, site-specific effective porosity was identified as variable at the site. To account for this variability, a sensitivity analysis was conducted over a range of 0.01 to 0.001. The highest groundwater concentration resulted from a value of 0.01, which was used in the analysis.

Table 6-8
Site Specific Parameters used in RESRAD Deep Soil Analysis

Parameter		Value	Units
Contaminated Zone site specific hydraulic conductivity		32	m/y
Contaminated Zone site specific b factor		4.05	
Site Specific Effective Porosity		0.01	
Unsaturated. Zone Site Specific Hydraulic Conductivity		1000	m/y
Site Specific Soil Kds:	Co	335.0	cm3/g
	Sr	152.0	cm3/g
	Cs	1200.0	cm3/g
	Ni	274.0	cm3/g

Attachment 6-9 provides the RESRAD output report. The attachment provides the results for the radionuclides that were projected to migrate to groundwater over a 1000 year period. The RESRAD code was used only to estimate maximum groundwater concentrations, not calculate dose. The dose from the groundwater concentrations listed in Attachment 6-9 were calculated using the same parameters as in the water dose calculations performed for contaminated basement surfaces, activated concrete/rebar, and embedded piping, i.e, 478 l/y annual water intake and FGR 11 Dose Factors. The spreadsheet output and the unitized dose factors for deep soil are provided in Table 6-9.

Table 6-9
Deep Soil Unitized Dose Factors

Key Parameters

Porosity	0.3		Yearly Drinking Water	478	L/y	Surface Soil Depth	0.15	m
Bulk Density	1.6	g/cm^3	Irrigation Rate	0.274	L/m^2-d			

DOSE CALCULATION FACTORS				Source Term			DEEP SOIL ANNUAL DOSE			
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Deep Soil Inventory pCi/g	Derived Water Conversion Units pCi/L per pCi/g	Water Inventory pCi/L	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Cs-137	2.27E+00	5.00E-05	4.00E-01	1.00E+00	9.02E-03	9.02E-03	2.16E-04	8.53E-06	4.00E-01	4.00E-01
Co-60	6.58E+00	2.69E-05	2.40E+00	1.00E+00	2.24E-02	2.24E-02	2.88E-04	6.15E-05	2.40E+00	2.40E+00
H-3	2.27E-01	6.40E-08	0.00E+00	1.00E+00	6.69E+03	6.69E+03	2.05E-01	6.33E-01	0.00E+00	8.37E-01
Ni-63	1.19E-02	5.77E-07	0.00E+00	1.00E+00	6.01E-01	6.01E-01	1.66E-04	2.98E-06	0.00E+00	1.69E-04

6.6.6 Groundwater

This calculation applies to existing groundwater only. As described above, there are additional contributions to the projected total groundwater dose from other contaminated materials.

Groundwater dose is calculated directly from the highest individual groundwater sample result from site monitoring well locations. As reported in Section 2, Attachment B, the only radionuclide identified in site groundwater is H-3 and the maximum concentration was identified in the containment foundation sump at a concentration of 6812 pCi/l. The range of H-3 concentrations identified during characterization sampling of site wells was 441 pCi/l to 6812 pCi/l, for the most part consistent with background levels. The containment sump was re-sampled during continued characterization with 900 pCi/l H-3 identified. In addition, routine containment sump water samples have been collected since February 2000. None of these samples have exceeded the MDC level of about 2500 pCi/l. (Additional sampling and analyses of site groundwater conducted in 2002, including the containment foundation sump, are discussed in Section 2.5.3.d and reported to the NRC in references noted in that section. The additional sampling confirmed the nuclide fraction and conservatism of the H-3 activity level assumed in the dose assessment.)

In general, it appears that current containment sump H-3 water concentrations are within the range expected in area water background. However, to ensure that a conservative water concentration is applied and to avoid the potentially extensive sampling and analyses necessary to demonstrate that the concentrations are at background levels, the 6812 pCi/l H-3 concentration is used in the dose assessment. If, prior to unrestricted release of the site, additional groundwater monitoring data are collected that indicate higher H-3 concentration, or identify other radionuclides, the higher concentrations will be used in the final dose assessment for demonstrating compliance with the 10/4 mrem/yr dose limit.

As discussed in Section 2.5.3.d, additional routine sampling of the containment foundation sump and PAB test pit will be conducted routinely until final status survey has commenced in these two plant areas. The samples will be taken on an approximate monthly basis and will be analyzed by gamma spectroscopy and for H-3. Sample analysis results will be evaluated regarding: (1) the need for additional assessment (such as, additional sampling or "hard to detect" analyses) and (2) any impact to the dose assessment.

There are no unit dose factors or DCGLs for groundwater. The actual dose from the highest measured concentration will be used in the total dose calculation. The groundwater dose is calculated using the FGR 11 DCF for H-3 and a 478 l/y intake. The resulting dose is 0.21 mrem/y. The method for factoring the groundwater dose into the total dose calculation and the DCGL determination for other contaminated materials is described in Section 6.7.

The dose calculation for existing groundwater is provided below.

$$Dose_{GW} = (6812 \text{ pCi/l H-3})(478 \text{ l/y})(6.4\text{E-08 mrem/y/pCi}) = 0.21 \text{ mrem/y} \qquad (12)$$

6.6.7 Surface Water

Site surface water from the Fire Pond and Reflecting Pond was sampled during characterization. The results indicated no plant derived radionuclides in the Fire Pond and a low potential in the Reflecting Pond. Therefore, only the Reflecting Pond was considered in the dose assessment.

Tritium was detected in the Reflecting Pond at a maximum concentration of 960 pCi/l. This activity is not believed to be attributable to Maine Yankee operations. However, a review of available literature on H-3 concentrations in surface water could not conservatively demonstrate that the H-3 concentrations identified were consistent with background levels in the region. Additional characterization and literature review may provide the information needed to demonstrate that the H-3 was not plant derived. However, given the very low dose from these H-3 concentrations, it was not considered cost effective to perform more analyses.

As for groundwater, the dose from surface water was calculated using existing data. The maximum H-3 concentration of 960 pCi/l was used. As with groundwater, if higher concentrations or additional radionuclides are identified at any time prior to unrestricted release of the facility, the higher concentrations will be used in the final dose assessment for demonstrating compliance.

The surface water dose results from drinking water and ingesting fish from the pond. The water dose is calculated using the parameters described above assuming that the resident farmer drinks directly from the surface water source. The dose from fish ingestion is calculated using a water to fish transfer factor of 1 for H-3 (NUREG-5512, Vol. 3, Table 6.30), 20.6 kg fish consumption per year (DandD default value), and using DCFs from FGR No.11.

The calculations for water and fish consumption from onsite surface water with a H-3 concentration of 960 pCi/l is provided below.

$$Dose_{SW} = (960 \text{ pCi/l H-3})(478 \text{ l/y})(6.4\text{E-08 mrem/y/pCi}) = 2.9\text{E-02 mrem/y} \quad (13)$$

$$Dose_{Fish} = (960 \text{ pCi/l})(1.0 \text{ pCi/kg per pCi/l})(20.6 \text{ kg/y}))(6.4\text{E-08 mrem/y/pCi}) = 1.3\text{E-03 mrem/y} \quad (14)$$

6.6.8 Buried Piping

a. Conceptual Model

After decommissioning is completed, some piping and conduit will remain underground at depths greater than three feet below grade. This contaminated material category includes the piping buried in open land, not pipe embedded in concrete basements, which were described in Section 6.6.3. A list of the buried piping that current plans call to remain after decommissioning is provided in Attachment 6-10. The buried piping is expected to contain very limited levels of contamination, if any. The radionuclide mixture is assumed to be the same as for contaminated materials.

The conceptual dose model for the buried piping is very simple and conservative. The piping/conduit is assumed to be uniformly contaminated over the entire internal surface area. The piping is further assumed to eventually disintegrate resulting in the total inventory in the pipe mixing with a volume of soil equal to the pipe volume. Without the assumption of the pipe disintegrating, there is essentially no dose pathway from buried piping. The resulting calculated soil concentrations are treated as deep soil and the dose was calculated using the same methods as described above for deep soil. However, the direct exposure is calculated assuming a three foot cover as opposed to a 15 cm cover. Although not required by the conceptual model, the buried piping DCGLs will be limited to ensure that the projected soil concentrations are below the surface soil DCGLs. This additional measure of conservatism was also applied to deep soil to account for hypothetical future excavation of the buried contamination.

b. Unitized Dose Factors for Buried Piping

The total surface area and total volume were calculated for all of the buried piping planned to remain after decommissioning. Assuming a unit inventory of 1 dpm/100 cm^2 on the internal surfaces, the total inventory of each radionuclide was determined. This total inventory was divided by the total volume and converted to grams of soil assuming a density of 1.6 g/cm^3 to calculate the projected pCi/g soil concentration of each radionuclide. The list of Buried Piping and the calculation of projected pCi/g soil concentration are provided in Attachment 6-10. The resulting concentration is 2.59E-04 pCi/g.

The resulting projected pCi/g soil concentration was entered as the source term in RESRAD for each applicable radionuclide. The RESRAD analysis was performed using the same parameters used for deep soil (Table 6-8) with the exception of the source term geometry. For the buried piping, the source term geometry was assumed to be a 142 m^2 area 1 m deep. This corresponds to the total volume of all buried piping of 142 m^3. This is a conservative assumption since, in reality, the piping is distributed over a fairly large surface area which would result in dilution through groundwater transport compared to the maximum concentration assuming all the pipe is contiguous. The RESRAD output report is provided in Attachment 6-11.

Microshield runs were performed on the unit source term assuming the same 142 m^2 x 1m deep source. The source is assumed to be covered by three feet of soil. The resulting exposure rate was multiplied by the default outdoor occupancy time (0.1101 y) from DandD, Version 1. The Microshield reports

Table 6-10
Buried Piping Unitized Dose Factors

Key Parameters

Porosity	0.3		Yearly Drinking Water	478	L/y
Bulk Density	1.6	g/cm^3	Irrigation Rate	0.274	L/m^2-d
Buried Pipe Conversion Factor	2.59E-04	pCi/g per dpm/100 cm^2	Surface Soil Depth	0.15	m

Dose Calculation Factors				Source Term			Buried Piping Annual Dose			
Nuclide	FGR 11 mrem/pCi	NUREG-1727 mrem/y per pCi/g	Microshield mrem/y per pCi/g	Water Inventory pCi/L per pCi/g	Pipe Surface Inventory dpm/100cm2	Soil Inventory pCi/g	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.42E-04	1.47E+01	0.00E+00	2.15E-02	1.00E+00	2.59E-04	3.77E-07	3.41E-08	0.00E+00	4.12E-07
Cs-134	7.33E-05	4.39E+00	2.21E-05	2.25E-05	1.00E+00	2.59E-04	2.04E-10	1.07E-11	5.72E-09	5.94E-09
Cs-137	5.00E-05	2.27E+00	3.97E-06	3.27E-04	1.00E+00	2.59E-04	2.02E-09	8.01E-11	1.03E-09	3.13E-09
Co-60	2.69E-05	6.58E+00	2.53E-04	8.14E-04	1.00E+00	2.59E-04	2.71E-09	5.78E-10	6.55E-08	6.88E-08
Co-57	1.18E-06	1.67E-01	9.44E-09	1.15E-04	1.00E+00	2.59E-04	1.68E-11	2.07E-12	2.45E-12	2.13E-11
Fe-55	6.07E-07	2.50E-03	0.00E+00	4.30E-05	1.00E+00	2.59E-04	3.23E-12	1.16E-14	0.00E+00	3.24E-12
H-3	6.40E-08	2.27E-01	0.00E+00	1.98E+02	1.00E+00	2.59E-04	1.57E-06	4.85E-06	0.00E+00	6.42E-06
Ni-63	5.77E-07	1.19E-02	0.00E+00	2.09E-02	1.00E+00	2.59E-04	1.49E-09	2.68E-11	0.00E+00	1.52E-09

and Buried Piping Direct Radiation Dose Factors are provided in Attachment 6-12. The spreadsheet output and resulting unitized dose factors (1 dpm/100 cm^2) for buried piping are provided in Table 6-10.

6.6.9 Forebay and Diffuser

a. Forebay Source Term

<u>Forebay Physical Description</u>

The forebay is a basin approximately 400 feet long by 160 feet wide at the top with a granite (ledge) floor, rock/soil walls on two sides, and small concrete walls at each end. The depth is approximately 20 feet. The volume is (64,000 ft^2 bottom + 10,150 ft^2 top incline area) x (20 feet deep) = 1.48E6

ft^3 or 42,000 m^3. The surface area of the bottom plus sides (assuming flat sides) is 7435 m^2. If the rip-rap surface is calculated, the surface area is 2337 m^2. (This assumes the number of circles of 2 foot diameter contained within the forebay wall area then converting those to a half sphere area of 1 foot radius.) The rip-rap volume is estimated at 478 m^3. The total surface area when the forebay is backfilled is 7435 m^2.

There are four potentially contaminated media associated with the forebay: ledge, rip-rap, sediment, and soil. Each of these will be examined separately to determine the dose contribution of each medium. It should be noted that pre-remediation studies conducted to date indicate that the activity in the forebay sediment is very insoluble (i.e., no activity is given up to water nor is there detectable activity in a water filtrate). Most of the activity is contained within the organic layer of sediment or the organic film deposited on the rip-rap of the forebay. Based on solubility, pH, and water chemistry, the conditions for the maximum release of activity from sediment or surface film are occurring now. In spite of these ideal release conditions, no detectable activity is found in the standing water of the forebay. Furthermore, the infiltration water that enters the forebay through pathways in the dikes is brackish which makes the drinking or irrigation pathways doubtful. None the less, drinking and irrigation were evaluated.

Characterization Data

A detailed discussion of forebay / diffuser characterization is provided in Attachment 2H. Table 6-10A (below) provides estimated total activity, where appropriate, for each principal contaminated media. No current contamination data are available for the forebay granite ledge floor; but given its low permeability, the ledge is expected to be clean following remediation. This will be verified. The rip-rap activity is based on the average surface activity of the rip-rap times the entire rip-rap surface area.

Table 6-10A
Estimated Media Activity

Media	Total Activity
Ledge	To be remediated. (See also Table 2H-5 in Attachment 2H.)
Rip-Rap	10.5 uCi Co-60
Marine Sediment	To be remediated. (See also Table 2H-5 in Attachment 2H.)
Soil	1.85E4 uCi Co-60 1.52E3 uCi Cs-137

Drinking Water and Irrigation Dose

The drinking water and irrigation water dose was modeled using the same approach as that used for the basement fill model. The forebay surface area to volume ratio was calculated as 0.177 m^2/m^3, or using the rip-rap surface, as 0.06 m^2/m^3. The surface area of 435 m^2 for the source term was calculated by multiplying the surface area to volume ratio by the volume associated with the annual water usage (738m^3) for the soil porosity 0.3. The source term for the drinking water was then calculated assuming a contamination level equal to the concrete structure DCGL of 36,000 dpm/100cm^2. Thus, the dose contribution from the forebay surface area source term was calculated as 0.004 mrem from drinking water and 0.0008 mrem from irrigation water. These dose contributions are well below and are bounded by the dose contributions from the drinking water and irrigation water sources to the resident farmer from the building basements. Therefore, these dose contributions are considered separate from the resident farmer dose modeling scenario. Furthermore, since this dose is so insignificant and the probability is so low that an individual would be able to successfully place a viable well within the forebay, survey measurements of the forebay surfaces including rip-rap will be limited.

Rock (Rip-Rap) Dose

The exposed surface area of the rip-rap is 2337 m^2. The surface activity is spread over the exposed surface area at 0.1 pCi/g (based on diffuser surface sample and rip-rap sample levels) or 45 pCi/100 cm^2 Co-60. When deposited over the exposed surface area, this level of Co-60 contamination results in a total activity from rip-rap of 10.5 uCi. This activity is assumed to be instantaneously released and mixed within the forebay soil backfill volume. This results in a soil concentration of 1.56E-4 pCi/g Co-60.

Sediment Dose

Several large pockets of sediment were identified on the floor of the forebay during diving inspections. There are also small deposits lying between the rip-rap and also behind the weir (in the seal pit). The activity of the underwater marine sediment averages 19 pCi/g Co-60 and 2 pCi/g Cs-137. (One small area of very high activity was discovered which had Co-60 levels as high as 445 pCi/g.) The sediment within the forebay is all slated for removal by washing, settling, filtering and dewatering. The dewatered sediment will be disposed of as radwaste and will not contribute significantly to dose. Any residual activity remaining following sediment removal would be included in the ledge dose for 36,000 dpm/100 cm^2 surface contamination and the shallow pockets of contaminated sediment which might remain have previously been analyzed and found not to contribute a significant dose (EC 004-01).

Direct Dose Excavated Forebay Soil

Coastal zoning or land use restrictions may prohibit or severely limit excavation or construction activities in the area of the former forebay given its closeness to the shoreline. None the less, the dose from these activities has been evaluated as discussed in this section. Contaminated soil has been detected in approximately a two foot deep band behind the rip-rap. The nuclide fraction is assumed to be the same as the sediment since it originates from the same effluent releases. (See Section 2.5.3 and Attachment 2H for additional discussion of the nuclide fraction and supporting characterization data.) The average activity levels detected were 7.3 pCi/g Co-60 and 0.6 pCi/g Cs-137; maximum levels were 21.3 pCi/g Co-60 and 1.35 pCi/g Cs-137. No Sb-125 was detected in the soil samples. A two foot thick band of contaminated soil 35 feet high by 400 feet long (the forebay wall dimensions) for two dike walls is 1586 m^3 of contaminated soil.

The excavation of two different sized homes are evaluated to determine the volume of soil which must be excavated assuming the worst case volumetric capture of contaminated soil within the excavation volume. An excavation for a 2000 square foot house results in a factor of 11 associated with the worst case capture of contaminated soil with clean soil. An excavation of a 1000 square foot house results in a factor of 7.9 associated with the worst case capture of contaminated soil with clean soil. In neither case, is credit taken for any additional clean soil which would be generated if the excavation was sloped for safety concerns. In both cases, the contaminated soil is assumed to begin at the surface with no cover material, even though the as-left elevation of the forebay will be a few feet above the contaminated zone which exists in the inter tidal zone of the forebay. Therefore, a conservative dilution factor of 7.9 may be applied to determine acceptable levels of radioactive materials of forebay soil in the two feet immediately behind the rip rap.

The dose to a person from the excavation of the contaminated soil is shown in Table 6-10B below, assuming the dilution factors described above and the annual outdoor exposure time for soil at the average activity values and for soil at the 3 pCi/g equivalent activity. The dose reduction due to shielding by a 6" concrete basement floor for the average soil activity is also shown in Table 6-10B.

Table 6-10B
Excavated Soil Direct Dose

	Initial Dose Rate (mrem/h)				Dose at Average Soil Concentration (mrem/y)	Dose at 3 pCi/g Equivalent (mrem/y)
	Average Concentration	3 pCi/g	Hrs/y	Dilution Factor		
Large House	2.14E-02	3.0E-03	964	11	1.87	0.26
Small House	1.31E-02	1.80E-03	964	7.88	1.60	0.22
Basement	1.11E-03	---	5756	---	6.4	---

The excavation scenario dose rates are less than the soil dose rate to the resident farmer, therefore, this scenario is presented as a separate and dose-bounded scenario to the resident farmer.

b. Diffuser Source Term

The source term for the diffuser is the sediment entrained within the diffuser pipes. The sediment activity initially came from plant liquid effluent releases via the forebay and later via the movement of benthic silt back into the diffuser pipes by tidal action. These liquid effluent releases were made in accordance with licensed effluent controls and were routinely reported to the NRC. The effluent reports contained dose assessments which demonstrated compliance with 10 CFR 20 limits. The diffuser consists of 2 pipes 9 feet in diameter and 516 feet long. These two pipes are fed by trunk lines originating at the forebay. The portions of the trunks that are submerged and can contain sediment are 1421.5 feet in length. The volume occupied by the diffusers is 1860 m^3 and the volume of the trunk lines is 2562 m^3. This conservatively results in a potential sediment-filled source of 4422 m^3. For a circumference of 28.3 ft. and a length of 2543.5 ft, the pipe interior would have a surface area of 71,981 ft^2. Converting this area in ft^2 to 100 cm^2 areas results in a value of 6.68E5 100 cm^2 areas.

Coupons of the diffuser pipe were removed and analyzed for surface contamination. The nuclides detected were Co-60 and Cs-137 at nearly equal activity. The combined activities of both nuclides were approximately 0.28 pCi/g. This specific activity multiplied by the sample mass of 125g results in approximately 35 pCi per sample. The samples represent about 100 cm^2. The activity was present as a tightly-adhered, thin film of organic material. Based on the total interior surface area of the diffuser, if all of the activity on the interior surface of the pipes is relocated to the sediment, the additional activity would be 30 uCi.

Sediment samples taken from inside the diffuser and analyzed by gamma spectroscopy gave the following average activity values.

Co-60	1.1 pCi/g
Cs-137	0.15 pCi/g

The sediment nuclide activity was determined by multiplying the activity values by the sediment volumes as shown for Co-60 and Cs-137 for a total activity of 8315 uCi.

Water Activity

The sediment activity is assumed to be instantaneously released non-mechanistically into the waters of Montsweag Bay. (It is likely that the sediment will remain in the diffuser pipes for years to come and the radioactivity slowly be reduced by decay.) Since the Bay is an estuary, the water is considered non-drinkable. The volume of water into which the activity is released was determined by consulting MYC-2035 which discussed the former condenser cooling water "mixing zone". The mixing zone was established for thermal mixing assuming cooling water is released at a rate of 950 cfs. With the cooling water pumps no longer operable, such flow rates are not feasible. However, using the area in which forced mixing of the diffuser water occurred would result in a reasonable estimate for a mixing area for the potential sediment activity released at a much lower flow rate. (Churchill (1980) stated that the same flow model applies to both radionuclide dispersion and hot water dispersion from the plant.)

Using this "mixing zone" and the activities given above for sediment with HTDs included, the water concentrations for each nuclide were calculated. This activity level is assumed to exist for a year, when in fact, it would be dissipated within 56 hours by tidal flushing of the bay. Assuming dilution, the water concentration would be reduced by 6.4E-3 (56 h dilution time/8760 h per year) and the total annual dose would be on the order of 0.005 mrem/y for fish and 0.002 mrem/y for shell fish.

The annual dose rate to the individual who consumes seafood from this contaminated water source was derived by multiplying the water activity by the seafood bioaccumulation factors given in NUREG-5512 by the FGR-11 dose conversion factor for each nuclide times the consumption rate taken from NUREG-5512. Based on a comparison to local marine organism nuclide levels, the NUREG-5512 values are considered to be conservative.

The total dose from eating seafood (fish plus shell fish) grown in the contaminated water is 0.007 mrem/y. The consumption of this food source would actually replace other food sources included in the dose model. If the dose from eating this seafood were simply added to the annual dose to the resident farmer, it would represent a negligible increase compared to the farmer's total annual dose. Therefore, since the dose increase is negligible, this dose has not been added to Table 6-11. Furthermore, since the dose is negligible and the activity would likely be contained in the diffuser for sufficient time for substantial decay of dose significant nuclides, any further survey measurements of the diffuser will be limited.

Sediment Dose

A person could be exposed from direct radiation originating from the contaminated sediment if it were deposited upon a shoreline or mud flat. This portion of the calculation assumes that the total sediment activity is suspended within the area outlined by the "mixing zone" and is then non-mechanistically dewatered to the condition of a mud flat. The area is approximately 52,500 m^2 compared to the entire mud flat of Bailey Cove (130,000 m^2). An area ratio of 0.404 describes that portion of the entire Bailey Cove mud flat that could be covered intact by the postulated release.

The NRC (RG 1.109) adjusts the annual dose from shoreline deposits for the amount of time spent on the shore and for the geometry of the shoreline (shoreline width factor). For tidal basins like Montsweag Bay, the width factor is 1. For river shorelines, like the Back River, the factor is 0.2. For conservatism, a factor of 1 was used. The NRC time for shoreline recreation is 47 hours per year, however, Maine Yankee recognizes (ODCM) the presence of the commercial worm digger on the mud flats for 325 hours per year.

The sediment dose rate (mrem/hr) is the product of the sediment activity divided by the mud flat area factor times the dose rate at 1 m from the resulting activity deposited (from RG 1.109). For Co-60 the dose rate would be: 7315 uCi/5.25E4 m^2 x width factor of 1 x 1E6 pCi/uCi x 1.7E-8 mrem/hr/pCi/m^2 x 0.404 = 9.6E-4 mrem/hr. (Note that Reg. Guide 1.109 does not provide values for Sb-125. The dose rate was estimated at half the Cs-137 value based on the dose rate for soil.)

The total whole body dose rate is 1.01E-3 mrem/h from contaminated mud flats. Using the worm digger exposure time of 325 hr/y x 1.01E-3 mrem/hr = 0.327 mrem/y. If the DandD outdoor time fraction (964 hr/y) is used, the annual dose would be 0.97 mrem/y.

6.6.10 Circulating Water Pump House

The circulating water pump house (CWPH) was the intake for the plant circulating water (CW) system. The water intake was directly from the Back River at high volumes (about 400,000 gpm). The CWPH will be demolished to three feet below grade, backfilled, and stabilized on the river side with rock rip-rap. The intake structure which is below water level will remain in communication with the river. The contamination potential in this structure is very low.

There are three, albeit low potential, exposure pathways from the material that will remain in the demolished and backfilled CWPH: (1) exposure to radionuclides that have leached to the tidal water that saturates the remaining backfilled structure, (2) exposure from the excavation of the limited amount of silt currently on the bottom of the pump house bays, and (3) exposure from contamination that leaches from the structure surfaces, is adsorbed onto fill material, and is excavated at some time in the future.

Exposure to the excavated silt is limited to the same pathways as surface soil. Therefore, the DCGL for the silt will be the same as calculated for surface soil. In addition, the radionuclide mixture is assumed to be the same as that identified for surface soil. This assumption has essentially no effect since the samples will be counted by gamma spectroscopy, which will specifically identify the radionuclides of concern. Limiting the silt DCGL to the surface soil DCGL ensures that there will be no additional dose to the resident farmer, above that already accounted for through the surface soil DCGL, from the hypothetically excavated silt.

The potential for radionuclide leaching from the surfaces of the CWPH is very remote considering the extremely low potential of contamination being present as a result of past operations and the fact that if contamination were present from past operations, the constant tidal flushing of the pump house bays would have already removed any leachable material. Notwithstanding this low potential, one water sample will be collected from each of the four pump house bays prior to draining the bays for final survey. The analytical detection sensitivity will be at the environmental LLD level. If no activity is detected, the water leaching pathway will be eliminated from consideration. Potential leaching to water will be evaluated by direct water sampling only.

If activity above the environmental LLD is detected in the water samples, the positive results will be used to evaluate exposure from fish ingestion using the bioaccumulation factors from NUREG-5512, Vol. 3, Table 6.30, i.e., 20.6 kg fish consumption per year (DandD default value), and DCFs from FGR No.11. If a dose calculation is necessary, the dose will be added to the total dose from the other contaminated materials listed in Table 6-11. Adjustments will be made to the DCGL's for other contaminated materials, if necessary, to ensure compliance with the 10/4 mrem/yr unrestricted use criteria.

Since potential leaching into water is accounted for by direct water sampling, the only remaining exposure pathway to consider is the excavation of fill material hypothetically contaminated by radionuclide transfer from structure surfaces to the fill. The conceptual model developed for the contaminated basement surfaces is adequate to apply to this very low potential pathway. As shown in Attachment 6-13, the DCGL for building basements in Table 6-11 resulted in very low radionuclide concentrations on the basement fill, with all concentrations being less than 1 pCi/g. Note that one of the criteria applied to the selection of the basement fill DCGL is that the calculated fill concentration be less than the surface soil DCGL. In addition, the Kd's used for the basement fill model (Bank Run Sand) are generally higher than the Kd's for Bank Run Gravel which is being considered for backfill. This indicates that the CWPH fill would have lower concentrations than those calculated for basement fill. However, regardless of the fill material used, it is unlikely that the fill concentration would exceed the surface soil DCGL.

Considering all of the arguments presented above, the DCGL calculated for the building basements is appropriate and conservative for application to CWPH surfaces for the purpose of limiting hypothetical dose from the excavated fill pathway (as stated above, the potential leaching to water is addressed by direct sampling of the water). Compliance with the basement fill DCGL will ensure that the fill concentration will not exceed the surface soil DCGLs. Since the concentration of the hypothetically excavated fill would be below the surface soil DCGLs, there will be no additional dose to the resident farmer beyond that already accounted for through the surface soil and no addition to the total dose calculated in Table 6-11 is necessary.

6.7 Material Specific DCGLs and Total Dose Calculation

As described above, calculations were performed to develop conservative dose assessment models and generate unitized dose factors for all contaminated materials at the Maine Yankee site and all radionuclides in the Maine Yankee mixture applicable to each material. When the dose pathways for the resident farmer were evaluated, it was evident that the resident farmer could receive dose from more than one contaminated material. A detailed discussion of the various contaminated materials and dose pathways was provided above. The total dose results from the summation of the contributions from each of contaminated materials. Therefore, the final DCGLs for each of the contaminated materials are inter-dependent.

This section describes the method used to account for the dose from all materials and select the final DCGLs for all materials. The method ensures that the summation of doses from all pathways, at the selected DCGL concentrations for all materials, does not exceed 4 mrem/y drinking water dose and 10 mrem/y total dose. Table 6-11 provides the DCGLs that were selected for the Maine Yankee Site and the resulting total dose for all contaminated materials. (Since the containment basement was the only remaining basement structure to be directly impacted by activated concrete, the basement fill dose calculations were treated in two approaches. One assessment was performed for the containment basement, accounting for the direct impact of the activated concrete present. A second assessment was performed, conservatively modeling the remaining non-containment basement structures. The results are presented in Table 6-11, which comprises two tables: one for containment (Table 6-11a) and one for non-containment (Table 6-11b). For additional discussion on the dose assessment related to activated concrete, see the request for license amendment in Reference 6.10.7, which was approved by the NRC in Reference 6.10.8).

Attachment 6-13 contains the dose calculations for all contaminated materials listed in Table 6-11. The radionuclide mixture for "special areas" differs from the rest of the basement surfaces. Therefore, a separate DCGL was selected and a separate dose calculation was performed for the "special areas". (See Attachment 2F for a discussion of "special areas".)

The DCGLs listed in Table 6-11 are target project DCGLs. The formal unrestricted use criteria are the enhanced State dose criteria of 10 mrem/y or less from all pathways and 4 mrem/y or less from groundwater drinking sources. The DCGL values in Table 6-11 may be adjusted as the project proceeds using the methods and limitations described in this section as long as the dose criteria are satisfied.

Table 6-11a
Containment Contaminated Material DCGL

Basement Contaminated Concrete (gross beta dpm/100 cm^2):	18,000
Special Area Contaminated Concrete (gross beta dpm/100 cm^2)	9,500
Basement Activated Concrete - Released to Basement (pCi):	4.88E+08
Surface Soil (Cs-137 pCi/g):	2.39
Deep Soil (Cs-137 pCi/g):	2.39
BOP Embedded Piping [Limit: 100K], (gross beta dpm/100 cm^2):	100,000
Spray Building Pump Piping [Limit: 800K], (gross beta dpm/100 cm^2):	800,000
Ground Water (H-3, pCi/L):	6,812
Surface Water (H-3, pCi/L):	960
Buried Piping, Conduit and Cable, (gross beta dpm/100 cm^2):	9,800

Containment Contaminated Material Annual Dose

Material	Drinking Water (mrem/y)	Direct, Inhalation & Ingestion (mrem/y)	Total Annual Dose (mrem/y)
Contaminated Concrete	7.32E-02	8.53E-03	8.15E-02
Activated Concrete	1.36E-02	3.30E-02	4.66E-02
Surface Soil	0.00E+00	5.63E+00	5.63E+00
Deep Soil	5.33E-02	1.98E+00	2.04E+00
BOP Embedded Piping	4.59E-02	5.24E-03	5.11E-02
Spray Building Pump Embedded Piping	7.60E-02	8.68E-03	8.47E-02
Ground Water	2.08E-01	0.00E+00	2.08E-01
Surface Water	2.94E-02	1.27E-03	3.06E-02
Buried Piping, Conduit & Cable	6.33E-04	1.89E-03	2.52E-03
Total	0.48 mrem/y	6.79 mrem/y	8.17 mrem/y

Table 6-11b
Non-Containment Contaminated Material DCGL

Basement Contaminated Concrete (gross beta dpm/100 cm^2):	18,000
Special Area Contaminated Concrete (gross beta dpm/100 cm^2)	9,500
Basement Activated Concrete - Released to Basement (pCi):	0.00
Surface Soil (Cs-137 pCi/g):	2.39
Deep Soil (Cs-137 pCi/g):	2.39
BOP Embedded Piping [Limit: 100K], (gross beta dpm/100 cm^2):	100,000
Spray Building Pump Piping [Limit: 800K], (gross beta dpm/100 cm^2):	800,000
Ground Water (H-3, pCi/L):	6,812
Surface Water (H-3, pCi/L):	960
Buried Piping, Conduit and Cable, (gross beta dpm/100 cm^2):	9,800

Non-Containment Contaminated Material Annual Dose

Material	Drinking Water (mrem/y)	Direct, Inhalation & Ingestion (mrem/y)	Total Annual Dose (mrem/y)
Contaminated Concrete	2.70E-01	3.08E-02	3.01E-01
Activated Concrete	0.00E+00	0.00E+00	0.00E+00
Surface Soil	0.00E+00	5.63E+00	5.63E+00
Deep Soil	5.33E-02	1.98E+00	2.04E+00
BOP Embedded Piping	4.59E-02	5.24E-03	5.11E-02
Spray Building Pump Embedded Piping	7.60E-02	8.68E-03	8.47E-02
Ground Water	2.08E-01	0.00E+00	2.08E-01
Surface Water	2.94E-02	1.27E-03	3.06E-02
Buried Piping, Conduit & Cable	6.33E-04	1.89E-03	2.52E-03
Total	0.66 mrem/y	6.78 mrem/y	8.34 mrem/y

The dose summation method is a conservative screening approach. For example, the environmental pathway analysis for deep soil indicated that a low concentration of tritium would reach groundwater three years after the site is released for unrestricted use. The location of the deep soil and corresponding groundwater contamination are obviously different from the location of building basements where the hypothetical resident farmer well was placed. In addition, the peak time for H-3 water concentration from deep soil is different from the peak time for the basement water concentration. Nonetheless, consistent with a screening approach, the peak H-3 concentration in groundwater from deep soil is fully added to the peak basement water concentration and the sum is used in the dose assessment. There was no reduction in concentration due to the differences in peak dose time or dilution through groundwater transport. A more realistic and less conservative environmental pathway analysis would consider these effects.

The Maine Yankee commitment to a conservative screening approach is also seen in the methods for adding the dose contributions from embedded piping, activated concrete/rebar, and contaminated surfaces in the building basements, as well the other contaminated materials. It is important to recognize that the conservative results from the dose summation are in addition to the conservatism already built into the unitized dose factor calculations for the individual contaminated materials.

Soil areas outside of the RA boundary will not require consideration of dose from any other materials. The area of the RA is approximately 10,000 m^2, which represents the size of the resident farmer survey unit and contains the other contaminated materials considered (Refer to Section 5, Figure 5-2).[2] The other contaminated materials have essentially no effect outside of the RA and the dose is assumed to result from the contaminated soil only. In this case, the DCGLs will be based on the NUREG-1727 screening values corrected to represent 10 mrem/y. The soil radionuclide mixture applied to areas outside the RA boundary are assumed to be the same as the mixture listed in Table 2-11 The DCGL for areas outside the RA is 4.2 pCi/g. This DCGL can be calculated most directly by the ratio to the 2.39 pCi/g Cs-137 DCGL provided in LTP Table 6-11, recognizing that the dose from 2.39 pCi/g is 5.63 mrem/yr. This calculation is provided below:

$$4.2 \text{ pCi/g} = (2.39 \text{ pCi/g}) \frac{(10.00 \text{ mrem/yr})}{(5.63 \text{ mrem/yr})}$$

6.7.1 Conceptual Model for Summing Contaminated Material Dose

The conceptual model for summing doses to the resident farmer essentially combines the dose from surface soil and deep soil with the dose from water derived from a well drilled directly into the worst case building basement. The well water is used for irrigation and drinking.

The source term for the well water concentrations includes contributions from basement contamination, activated concrete/rebar, and embedded piping. The model assumes that the residual contamination in all three materials is instantaneously released and mixed with water that has infiltrated the building basement.

The instantaneous release of all contamination is conservative for several reasons. Concrete contamination will be released at a rate associated with the diffusion coefficient for the various radionuclides. Activated concrete/rebar will actually be released to the water at a relatively slow rate more closely linked to physical dissolution of concrete, which is expected be very slow. For embedded piping, the

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2 The Figure 5-2 Class 1 Area represents the size of the Resident Farmer's farm and therefore, consists of that land area to which the Dose Summation Table 6-11 applies. While operational considerations may result in modifications to the RA boundary, the extent of land area to which Table 6-11 applies should not change.

actual contamination release rate is expected to be close to zero because any open pipe end that could be a point of release into a basement will be sealed. Another conservatism is the assumption that all of these sources are mixed in the same worst case 2460 m^3 of basement volume. In actuality, the various sources are in different areas and different buildings. Finally, the source term contributions from groundwater, surface water, and deep soil were added directly to the basement well concentrations without consideration of transport or dilution.

6.7.2 Method and Calculations for Summing Contaminated Material Dose

The primary inputs to the dose summation are the unitized dose factor calculations developed for each contaminated material. The unitized dose spreadsheets were used for the dose calculations without modification. However, the input concentrations and inventories required modification to represent the selected DCGLs as opposed to unit concentrations. The additional calculations required to convert the DCGL values into radionuclide concentrations and inventories are described in the sections below.

To perform the summation and to provide a method to efficiently adjust the DCGLs for various materials, each of the individual material unitized dose spreadsheets was copied and linked in a single spreadsheet entitled DCGL/Total Dose. The spreadsheet output for the DCGL dose calculation for each material is provided in Attachment 6-13. These spreadsheets provide the calculations for the dose values reported in Table 6-11.

Contaminated Basement Surfaces

The DCGL for contaminated concrete is expressed as dpm/100 cm^2 detectable gross beta. This form was required because the final survey will be performed using gross beta measurements. The primary criteria for selecting the gross beta DCGL for basement surfaces was to ensure that the total dose, from all contaminated materials, was less than the 10/4 mrem/yr dose limit. There were two secondary criteria applied to the selection of the DCGL; 1) the DCGL would result in calculated basement fill concentrations below the surface soil DCGL, and 2) the DCGL was less than the NRC surface screening values from NUREG-1727, Table C2.2 (see Attachment 6-18).

To calculate the dose from a given gross beta DCGL, the gross beta concentration is converted to individual radionuclide concentrations based on their respective fractions in the radionuclide mixture. The individual concentrations are then input to the dose calculation spreadsheet for contaminated basement concrete. Characterization data indicated that the radionuclide mixtures for "special areas" differs from the other the basement surfaces (see Table 2-8). Therefore, a separate mixture is applied to the dose assessment for the "special areas", resulting in a different DCGL for the "special areas". The DCGL selected for the "special areas" resulted in a lower dose than that calculated for the rest of the basement surfaces (see Attachment 6-13). Therefore, the total dose shown in Table 6-11 is based on the higher dose calculated for the general radionuclide mixture and DCGL, not the "special areas" mixture.

The individual radionuclide concentrations are calculated as follows:

Convert the detectable gross beta concentration to total radionuclide concentration:

Total dpm/100 cm^2 = (gross beta dpm/100 cm^2)/(gross beta radionuclide fractions) (15)

Where: Total dpm/100 cm^2 is the summation of activity from all radionuclides
Gross beta is the detectable gross beta concentration
gross beta radionuclide fractions is the sum of the fractions of each radionuclide in the Maine Yankee mixture with detectable beta

Calculate each individual radionuclide concentration as follows:

C_R dpm/100 cm^2 = (NF_R)(Total dpm/100 cm^2) (16)

Where: C_R is the concentration of a given radionuclide
NF_R is the nuclide fraction of a given radionuclide

Surface Soil

The DCGL for surface soil is expressed in pCi/g Cs-137. The surface soil dose is calculated by first determining the individual radionuclide concentrations by ratio to Cs-137 using the relative fractions in the Maine Yankee mixture and then entering the individual concentrations into the "inventory" column in the dose calculation spreadsheet for surface soil.

During final survey, and in the final site dose assessment, the non-gamma emitting radionuclides (HTD nuclides) will be accounted for using Cs-137 as a surrogate as described in Equation 17 (from NUREG-1505, Page 11-2, Equation 11-4). The contribution from soil HTD radionuclides will be calculated using the radionuclide fractions listed in Table 2-11. Cs-137 was selected as the surrogate since it is the predominant radionuclide in soil (i.e., 89%) and since many of the soil samples will not result in positively detected Co-60. As seen of page 5 of Attachment 6-13, the dose contribution from the HTD radionuclides in soil (Ni-63 and H-3) is less than 1% of the Cs-137 dose. Therefore, the effect of the surrogate calculation on the Cs-137 $DCGL_w$ value will be minimal.

To calculate the surrogate Cs-137 DCGL, the following equation is used:

$$CS-137s=\frac{1}{\frac{1}{D_1}+\frac{R_2}{D_2}+\frac{R_3}{D_3}+\ldots+\frac{R_n}{D_n}} \qquad (17)$$

Where: Cs-137_s is the surrogate Cs-137 $DCGL_w$;
D_1 is the DCGL for Cs-137;
R_n is the ratio of the HTD radionuclide mixture fraction to the Cs-137 mixture fraction; and
D_n is the $DCGL_w$ of the HTD radionuclide corresponding to 10 mrem/yr. The DCGL's are calculated by inverting the Unitized Dose Factors Listed in the LTP, Table 6-7, and multiplying by 10.

The unitized dose factors were used in the total dose and DCGL calculations. This allowed the dose contribution of each radionuclide to be calculated and reviewed to understand the relative significance of the nuclides in the mixture. The dose calculated from the Cs-137 concentration shown in Table 6-11 will be the same regardless of whether a "surrogate" Cs-137 $DCGL_w$ is used or the unitized dose factors for all radionuclides are used.

The Cs-137 to Co-60 ratio will vary in the final survey soil samples and this will be accounted for using a "unity rule" approach as described in NUREG-1505, Chapter 11.

Before applying the unity rule, the DCGLs, for areas inside the RA, will be adjusted to represent the Table 6-11 total surface soil dose, as opposed to 10 mrem/yr. As seen in Table 6-11, the dose from surface soil is limited because of the additional dose from the other contaminated materials on the site. The unity rule calculation will limit the surface soil dose by multiplying the Cs-137_S and Co-60 DCGL's corresponding to 10 mrem/yr by a factor equal to the Table 6-11 total surface soil dose value divided by 10 mrem/yr. If the dose contribution from surface soil changes in the future, the multiplication factor will change accordingly.

In order to demonstrate compliance with the surface soil DCGL, the gamma spectroscopy results for each soil sample will be converted to a unity rule equivalent using the Table 6-11 surface soil DCGL's in the following equation. After this conversion, the DCGL becomes a unitless value of 1.0 that is equivalent to the total surface soil dose shown in Table 6-11. If the dose contribution from surface soil changes in the future, the dose corresponding to a unity rule equivalent of 1.0 will change accordingly. The unity rule equivalent is calculated per the following equation:

$$\text{Unity Rule Equivalent} \leq 1 = \frac{\text{Cs-137}}{\text{DCGL}_{(\text{Cs-137}_S)}} + \frac{\text{Co-60}}{\text{DCGL}_{(\text{Co-60}_A)}} + \ldots + \frac{R_N}{\text{DCGL}_{(N_A)}}$$

Where: Cs-137 and Co-60 are the gamma spec results,

$\text{DCGL}_{(\text{Cs-137}_S)}$ is the surrogate Cs-137_S DCGL, adjusted to represent the Table 6-11 total surface soil dose, as applicable (inside RA)

$\text{DCGL}_{(\text{Co-60}_A)}$ is the Co-60 DCGL adjusted to represent the Table 6-11 total surface soil dose, as applicable (inside RA)

R_N is any other identified gamma emitting radionuclides, and $\text{DCGL}_{(N_A)}$ is the adjusted DCGL for radionuclide N.

Absent sample-specific information from the final survey, using the radionuclide mixture fractions to represent the final Cs-137/Co-60 ratios is the best method available to estimate dose and determine target soil concentrations for remediation planning.

Activated Concrete/Rebar

The DCGL for activated concrete/rebar is in units of pCi total activity at the wall and floor surfaces. Total activity includes all radionuclides in the Maine Yankee mixture.

Deep Soil

The DCGL for deep soil, as for surface soil, is expressed in pCi/g Cs-137. The deep soil dose is calculated by first determining the individual radionuclide concentrations by ratio to Cs-137 using the relative fractions in the Maine Yankee surface soil mixture and then entering the individual concentrations into the "inventory" column in the dose calculation spreadsheet for deep soil. The surface soil radionuclide mixture is assumed to be representative of the deep soil mixture.

The issues related to compliance using final survey results for gamma emitters and the use of Cs-137 as a surrogate for the HTD radionuclides that were described for surface soil also apply to deep soil.

Groundwater

The existing groundwater concentrations are entered directly into the DCGL/Total Dose spreadsheet. This allows the dose from current groundwater contamination to be accounted for. The entered concentration is not intended to be a DCGL. If Maine Yankee's estimate of existing groundwater concentration changes, the value(s) input to the final dose calculation for compliance with the 10/4 dose criteria will use the most applicable concentrations.

Surface Water

The maximum concentration identified was used in the dose assessment. As with the groundwater concentration, the entered concentration is not a DCGL. If new sample data, if collected, indicates higher concentrations in site surface water, the new data will be used in the final dose assessment to demonstrate compliance with the 10/4 dose criteria.

Buried Piping

The buried piping DCGL is expressed as dpm/100 cm^2 gross beta. The DCGL/Total Dose spreadsheet converts gross beta concentration to individual radionuclide concentrations analogous to contaminated basement surfaces. The resulting concentrations are entered in the dpm/100 cm^2 inventory column in the dose calculation spreadsheet.

Embedded Piping

The embedded piping planned to remain after decommissioning has a total internal surface area of 154.3 m^2. The Spray Building contains 26.5 m^2 of embedded containment spray pump piping surface area with the remaining 127.8 m^2 located in the Containment, Spray Building PAB, and Fuel buildings.

Remediation performed to date on the Spray Building embedded piping has been extensive. Numerous sections of Ric-Wil piping (pipes within a pipe), most less that 5 feet long, that were contained in the concrete walls of the Spray building have been removed. Additionally, two Containment Spray Supply lines were removed by cutting 24- inch diameter cores through five feet of concrete. The cost was approximately $30,000.

The longest run of Spray building piping that remains is approximately 70 linear feet of 16 inch diameter, stainless steel Containment Spray Pump lines (CS-M-91, 92). The two pipes, which are 15 feet apart and cross- connected, extend from the lower level of the Spray building (at El.-14'9") to the safeguards sump (El.-4') in containment and are embedded in over 10 vertical feet and 16 horizontal feet of concrete.

An extensive effort to chemically decontaminate the containment spray pump piping occurred in June 2002. A caustic chemical, which has been successfully used in other facilities, was applied to the piping in four separate applications over a total of 74 hours. Although several sections of the vertical piping were decontaminated to relatively low levels, the majority of piping still contains residual contamination at an average level of ranging from 1E+04 dpm/100 cm^2 to about 1.5E+05 dpm/100 cm^2. The maximum level encountered based on remediation surveys to date is about 4E+05 dpm/100 cm^2. The cost of this project was on the order of $200,000.

The decontamination factors (ratio of before and after contamination levels) were high initially (up to 104). However, the decontamination factors were low for the fourth chemical decontamination effort (as low as 1). Further chemical decontamination is not expected to be effective. The only remaining alternative is removal and disposal as LLRW waste. Estimates to remove the spray building embedded piping range from about $200,00 to $285,000, excluding disposal costs which, for the large volume of concrete required to be removed, are approximately $150,000-175,000.

Assuming that residual contamination were present at an average level of 8E+05 dpm/100 cm^2 in the 26.5 m^2 of spray pump piping, the resident farmer dose contribution would be approximately 0.085 mrem/yr. The 8E+05 dpm/100 cm^2 value was selected to represent the upper range of the average contamination level.

Based on the total projected costs for removal and disposal of the spray pump piping of at least $350,000, the cost per person-rem would be over $4,000,000 per person-rem. This is far in excess of the NRC ALARA criteria of $2000 per person-rem listed in NUREG-1727. Therefore, additional decontamination is not justified.

Maine Yankee has evaluated the contamination potential of the embedded piping in the Containment, PAB, and Fuel building and does not believe the levels of contamination found in the spray pump piping will be encountered in these buildings. Therefore, two different DCGL's will be used for embedded piping. The DCGL for the spray pump piping will be 800,000 dpm/100 cm^2 and the DCGL for the rest of the embedded piping in the Spray Building, Containment, PAB, and Fuel buildings will be 100,000 dpm/100 cm^2.

The inventory for the dose assessment was calculated assuming that the spray pump piping (26.5 m^2) is contaminated at 800,000 dpm/100 cm^2 and that the remaining embedded piping (127.8 m^2) is contaminated at 100,000 dpm/100 cm^2. The entire inventory of embedded piping from all buildings was summed and assumed to be instantaneously released. The dose under these assumptions was calculated to be 0.136 mrem/yr.

The assumption of instantaneous release is conservative since the spray pump embedded piping will be filled with cement grout.

6.8 Area Factors

6.8.1 Basement Contamination

The basement contamination conceptual model described in Section 6.6.1 was based on a worst case surface area of 4182 m^2. The model assumes uniform mixing within a 0.6 m layer of fill in direct contact with the 4182 m^2 surface area. The conceptual model assumes that the activity released from the wall is mixed with the 738 m^3 volume of water contained in the 0.6 m fill layer, but does not require the contamination to be uniformly distributed over the entire 4182 m^2 surface area. The model source term is the total inventory over the surface and is not dependent on the distribution of the contamination on the surface. Therefore, consistent with the conceptual model, the area factor could be a simple linear relationship between total activity and area. The area factor formula would then be described using the following equation:

$$AF = 4182\ m^2/(\text{elevated area}) \tag{18}$$

where: AF is the area factor
(elevated area) is the size of the area exceeding the $DCGL_W$

Maine Yankee evaluated this potential approach and believes that it is consistent with NUREG-1575 and NUREG-1727 guidance which acknowledges that the area factors should be based on the dose model used to calculate the DCGL. However, it appears that substantially better remediation performance can be achieved than is reflected in Equation (18) and that leaving elevated areas at the levels allowed by the equation is not sufficiently conservative. Accordingly, the area factors for contaminated basement concrete will be calculated using Equation (19), which represents a considerably more conservative approach. The area factor used in the unity rule for contaminated basement concrete will be calculated using Equation (20) which ensures that the number of elevated areas in a survey unit are restricted to limit the inventory of activity allowed in a survey unit and maintain compliance with the release criteria.

$$AF = 50\ m^2/(\text{elevated area}) \quad (19)$$

$$AF_{\text{unity rule}} = (\text{survey unit})/(\text{elevated area}) \quad (20)$$

where: AF is the area factor
(elevated area) is the size of the area exceeding the $DCGL_W$
(survey unit) is the size of the survey unity

The 50 m^2 area was selected after qualitative consideration of the potential residual contamination that could remain in elevated areas after a comprehensive remediation effort. Areas greater than 50 m^2 are required to be at or below the $DCGL_w$. Area factors can apply to elevated areas on any surface, but are expected to be applied primarily to contamination in cracks and crevices, or other geometries, that are not efficiently remediated. It is not expected that a large number of elevated areas will remain. The number of elevated areas allowed to remain is limited by the formula presented in Section 5.6.3. . The survey unit size is determined in accordance with Section 5.3.1.a.

6.8.2 Surface Soil and Deep Soil Area Factors

The NRC screening values were used to calculate the surface soil DCGLs. This approach does not provide a direct method of linking the area factor calculation to the dose model. The surface soil area factors were determined based on the change in direct radiation as a function of area. The relative exposure was determined using Microshield. The output reports are provided in Attachment 6-14.

Using direct radiation only is a conservative approach since area factors based on the ingestion and inhalation dose pathways increase at a faster rate than those based on the direct radiation pathway. This is evident from inspection of Table 5.6 in NUREG-1575 which shows, for example, the higher area factors for Am-241 as compared to Cs-137 and Co-60. The area factors for surface and deep soil are listed in Table 6-12.

Table 6-12
Area Factors (AF) for Surface Soil and Deep Soil
Survey Unit = 10,000 m^2

Area m^2	1	2	4	6	8	16	25	50	100	500	1,000	10,000
Cs-137 (AF)	11.9	6.7	4.1	3.2	2.8	2.0	1.7	1.5	1.3	1.2	1.1	1.0
Co-60 (AF)	12.7	7.2	4.4	3.1	2.9	2.1	1.8	1.5	1.2	1.2	1.1	1.0
MY Mix (AF)*	12.0	6.8	4.1	3.2	2.8	2.0	1.8	1.5	1.3	1.2	1.1	1.0

* Where MY mix is the surface and deep soil radionuclide mixture.

6.8.3 Embedded Piping Area Factors

Since the dose model for embedded piping is the same as the basement fill model, the same area factor equation would apply.

$$AF = \frac{50\,m^2}{elevated\ area}$$

An evaluation of contamination potential and remediation effectiveness in embedded piping concluded that area factors can be limited to 2.0. Area factors larger than 2.0 can readily be justified on a dose basis using the above equation. However, a conservative application of ALARA was applied to limit the embedded piping area factor to 2.0

The number of elevated areas in embedded piping will be limited to ensure that the source term inventory (and annual dose) relative to the selected DCGL(s) is not exceeded.

6.8.4 Buried Piping Area Factors

Buried piping contributes less than one-tenth of one percent of the total dose to the resident farmer. The volume of piping expected to remain on site is 142.0 m^3. The radioactive contaminants associated with buried pipe are considered to be excavated to the soil surface uniformly mixed in the top 0.15 m of soil. Under these conditions area factors for soil would apply.

The following equation calculates an area factor that is ALARA and conserves the survey unit total inventory. As a measure of conservatism, a limit of 10 is placed on area factors for buried piping. The $DCGL_{EMC}$ (the DCGL used for the elevated measurement criteria) is calculated using the same equation.

$$Area\ Factor = \frac{Buried\ Piping\ Survey\ Unit\ Size(m^2)}{Buried\ Piping\ Elevated\ Area\,(m^2)}$$

For example, a 20 m^2 survey unit containing a 1.0 m^2 elevated area and using the DCGL of 9.50E+03 dpm/100 cm^2 would result in an area factor (AF) of 20:

$$Area\ Factor = 20 = \frac{20\,m^2}{1.0\,m^2}$$

The AF would be limited to 10 as stated above so the allowable activity in the elevated area would be 9.50E+04 dpm/100 cm^2. The $DCGL_{EMC}$ calculated by the equation would be 20 times the DCGL or 1.90E+05 dpm/100 cm^2.

If the maximum concentration of the elevated area (i.e., 9.50E+04 dpm/100 cm^2) were the only activity in the survey unit, the unity rule application would be as follows:

$$Unity\ Rule = \frac{9.50E+04\frac{dpm}{100cm^2}}{1.90E+05\frac{dpm}{100cm^2}} = 0.5\ which\ is < 1.0$$

6.8.5 Activated Concrete/Rebar Area Factors

The activated concrete/rebar conceptual model is conservatively treated in a similar manner as the basement contamination model. Activated concrete includes the source term in the entire volume of activated concrete (surface and subsurface). Unlike the basement fill model however, the activated radionuclide inventory is realistically released. Since the dose models are similar, the area factor for the Basement Fill Model (Section 6.8.1, equation 19 or 20 of the LTP) will be used for activated concrete.

6.9 Standing Building Dose Assessment and DCGL Determination

6.9.1 Dose Assessment Method

This dose assessment applies to the occupancy of a standing building and does not apply to the filled building basement. Current plans call for only one building to remain standing after decommissioning, i.e., the switchyard relay house. The NRC screening values from NUREG-1727, Table C2.2 were used for building occupancy dose assessment and DCGL determination. The screening values were adjusted to correspond to 10 mrem/y.

NUREG-1727, NMSS Decommissioning Standard Review Plan, Appendix C, describes the justification necessary to allow direct use of these screening values. When using the screening approach licensees need to demonstrate that the particular site conditions (e.g., physical and source term conditions) are compatible and consistent with the DandD model assumptions.

The following site conditions are specified for use of the Standing Building screening values:

1. The contamination on building surfaces (e.g., walls, floors, ceilings) should be surficial and non-volumetric (e.e., less than 0.4 in (10 mm)).
2. Contamination on surfaces is mostly fixed (not loose), with the fraction of loose contamination not to exceed 10 percent of the total surface activity.
3. The screening criteria are not applied to surfaces such as buried structures (e.g., drainage or sewer pipes) or mobile equipment within the building; such structures and buried surfaces will be treated on a case-by-case basis.

The above conditions are satisfied for the Maine Yankee site.

6.9.2 Standing Building DCGLs

The standing building DCGL was calculated as shown in Table 6-13. The DCGLs were calculated using Equation 4-4 in NUREG-1727 as adjusted for gross beta by multiplying the results by the gross beta radionuclide fraction in the mixture. The DCGL was expressed as gross beta since the final survey of a standing building, if necessary, will be performed using gross beta measurements.

Table 6-13 Gross Beta DCGL For Standing Buildings (Not Applicable to Basements to be Filled)				
Nuclide	**Nuclide Fraction (nf)**	**Screening Level dpm/100 cm²**	**Beta Fraction**	**nf/Screening Level**
H-3	2.36E-02	4.96E+07		4.75E-10
Fe-55	4.81E-03	1.80E+06		2.67E-09
Co-57	3.06E-04	8.44E+04		3.63E-09
Co-60	5.84E-02	2.82E+03	5.84E-02	2.07E-05
Ni-63	3.55E-01	7.28E+05		4.88E-07
Sr-90	2.80E-03	3.48E+03	2.80E-03	8.04E-07
Cs-134	4.55E-03	5.08E+03	4.55E-03	8.95E-07
Cs-137	5.50E-01	1.12E+04	5.50E-01	4.91E-05
		Sum	6.16E-01	7.20E-5
				DCGL 8.554E+03 β dpm/100 cm² (10 mrem/y)

6.9.3 Standing Building Area Factors

As discussed above for soil, using the NRC screening values for DCGL determination does not allow for direct determination of area factors. Consistent with the method used for soil, Microshield runs were used to generate the area factors by starting with an area of 100 m^2 and calculating the relative exposure rate as the area is decreased. The ratio of the 100 m^2 exposure rate to the respective smaller area exposure rate represents the area factor for the given elevated area size. Attachment 6-15 contains the Microshield runs and Table 6-14 provides the resulting area factors

Table 6-14 Area Factors (AF) for Standing Buildings (Does Not Apply to Building Basements To Be Filled) Survey Unit Size = 100 m^2									
Area m^2	0.5	1	2	4	8	16	25	50	100
Cs-137 (AF)	23.5	12.6	7.1	4.3	2.8	1.9	1.6	1.2	1.0
Co-60 (AF)	23.5	12.6	7.1	4.3	2.8	1.9	1.6	1.2	1.0
MY Mix (AF)	23.5	12.6	7.1	4.3	2.8	1.9	1.6	1.2	1.0

* Where MY mix is the Contaminated Concrete radionuclide mixture.

6.10 References

6.10.1 Baes, C.F., R.D. Sharp, A.L. Sjorren, and R.W. Shor, 1984. "A Review and Analysis of Parameters for Assessing Transport of Environmentally Released Radionuclides through Agriculture," ORNL-5786, Oak Ridge National Laboratory.

6.10.2 U.S. Environmental Protection Agency, 1988. "External Exposure to Radionuclides in Air Water and Soil, Federal Guidance Report No. 11," EPA 520/1-88-020, U. S. EPA Office of Radiation and Indoor Air.

6.10.3 Krupka, K.M., and R.J. Serne, 1998. "Effects on Radionuclide Concentrations by Cement/Ground-Water Interactions in Support of Performance Assessment of Low-Level Radioactive Waste Disposal Facilities," NUREG/CR-6377, PNNL-14408.

6.10.4 Onishi, Y., R.J. Serne, R.M. Arnold, C.E. Cowan, and F.L. Thompson, 1981. "Critical Review: Radionuclide Transport, Sediment Transport, and Water Quality Mathematical Modeling; and Radionuclide Adsorption/Desorption Mechanisms," NUREG/CR-1322, PNL-2901.

6.10.5 Sheppard, M.I. and D.H. Thibault, 1990. "Default Soil Solid/Liquid Partition Coefficients."

6.10.6 Maine Yankee Engineering Calculation, Diffuser and Forebay Dose Assessment, EC-041-01 (MY), Revision 0.

6.10.7 Maine Yankee letter to NRC (MN-03-049), dated September 11, 2003, Proposed Change: Revised Activated Concrete DCGL and More Realistic Activated Concrete Dose Modeling - License Condition 2.B.(10), License Termination

6.10.8 NRC letter to Maine Yankee, dated February 18, 2004, Issuance of Amendment No. 170 to Facility Operating License No. DPR-36 - Maine Yankee Atomic Power Station (TAC No. M8000).

Attachment 6-1
Fill Direct Dose Microshield Output

Basement Fill Direct Dose
Unitized Values

This attachment provides the Microshield outputs for direct dose factors for basement fill. The area size is 10,000 m2 by 5.8 m deep. Fill density is 1.5 g/cm3. The dose point is 1 meter above the soil surface. The shielded data assume 1 m of clean soil has been placed on top of the basement fill material. The dose factor assumes 964 hours exposure time per year.

MicroShield v5.05 (5.05-00201)
Maine Yankee

Page : 1
DOS File : SOILFL.MS5
Run Date : March 15, 2001
Run Time: 10:45:17 AM
Duration : 00:00:10

File Ref: ______
Date: ______
By: ______
Checked: ______

Case Title: Soil Fill
Description: Using the RESRAD land area approx.10000 m2
Geometry: 13 - Rectangular Volume

Source Dimensions

Length	580.0 cm	19 ft 0.3 in
Width	1.0e+4 cm	328 ft 1.0 in
Height	1.0e+4 cm	328 ft 1.0 in

Dose Points

	X	Y	Z
# 1	780 cm	5000 cm	5000 cm
	25 ft 7.1 in	164 ft 0.5 in	164 ft 0.5 in

Shields

Shield Name	Dimension	Material	Density
Source	5.80e+10 cm³	SiO2	1.5
Shield 1	100.0 cm	SiO2	1.5
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	8.7000e-002	3.2190e+009	1.5000e-006	5.5500e-002
Cs-137	8.7000e-002	3.2190e+009	1.5000e-006	5.5500e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

DOS File : SOILFL.MS5
Run Date: March 15, 2001
Run Time: 10:45:17 AM
Duration : 00:00:10

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	6.664e+07	2.688e-67	7.288e-28	2.239e-69	6.071e-30
0.0322	1.230e+08	3.170e-65	1.403e-27	2.551e-67	1.129e-29
0.0364	4.474e+07	2.967e-50	8.637e-28	1.686e-52	4.907e-30
0.6616	2.896e+09	1.398e-07	7.706e-06	2.711e-10	1.494e-08
TOTALS:	3.131e+09	1.398e-07	7.706e-06	2.711e-10	1.494e-08

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:45:17 AM
Dose Point # 1 - (780,5000,5000) cm

Cs-137

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	2.114e-007	1.165e-005
Photon Energy Fluence Rate	MeV/cm²/sec	1.398e-007	7.706e-006
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.711e-010	1.494e-008
Absorbed Dose Rate in Air	mGy/hr	2.367e-012	1.304e-010
"	mrad/hr	2.367e-010	1.304e-008
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.802e-012	1.544e-010
o Opposed	"	2.244e-012	1.236e-010
o Rotational	"	2.244e-012	1.236e-010
o Isotropic	"	1.984e-012	1.093e-010
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.978e-012	1.641e-010
o Opposed	"	2.829e-012	1.559e-010
o Rotational	"	2.829e-012	1.559e-010
o Isotropic	"	2.121e-012	1.169e-010
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.478e-012	1.365e-010
o Posterior/Anterior	"	2.187e-012	1.205e-010
o Lateral	"	1.622e-012	8.936e-011
o Rotational	"	1.954e-012	1.077e-010
o Isotropic	"	1.664e-012	9.168e-011

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:47:42 AM
Dose Point # 1 - (780,5000,5000) cm

Cs-134

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.411e-006	4.701e-005
Photon Energy Fluence Rate	MeV/cm²/sec	1.390e-006	4.050e-005
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.526e-009	7.528e-008
Absorbed Dose Rate in Air	mGy/hr	2.205e-011	6.572e-010
"	mrad/hr	2.205e-009	6.572e-008
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.539e-011	7.628e-010
o Opposed	"	2.134e-011	6.318e-010
o Rotational	"	2.134e-011	6.318e-010
o Isotropic	"	1.898e-011	5.604e-010
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.706e-011	8.132e-010
o Opposed	"	2.594e-011	7.774e-010
o Rotational	"	2.594e-011	7.774e-010
o Isotropic	"	2.019e-011	5.976e-010
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.273e-011	6.806e-010
o Posterior/Anterior	"	2.064e-011	6.127e-010
o Lateral	"	1.605e-011	4.694e-010
o Rotational	"	1.854e-011	5.493e-010
o Isotropic	"	1.618e-011	4.756e-010

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:53:24 AM
Dose Point # 1 - (780,5000,5000) cm

Co-57

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	5.133e-010	2.688e-008
Photon Energy Fluence Rate	MeV/cm²/sec	3.520e-010	1.819e-008
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	6.800e-013	3.515e-011
Absorbed Dose Rate in Air	mGy/hr	5.936e-015	3.068e-013
"	mrad/hr	5.936e-013	3.068e-011
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	7.006e-015	3.624e-013
o Opposed	"	5.637e-015	2.913e-013
o Rotational	"	5.637e-015	2.913e-013
o Isotropic	"	4.984e-015	2.576e-013
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	7.458e-015	3.857e-013
o Opposed	"	7.091e-015	3.666e-013
o Rotational	"	7.091e-015	3.666e-013
o Isotropic	"	5.330e-015	2.755e-013
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	6.204e-015	3.209e-013
o Posterior/Anterior	"	5.492e-015	2.839e-013
o Lateral	"	4.091e-015	2.113e-013
o Rotational	"	4.909e-015	2.538e-013
o Isotropic	"	4.188e-015	2.164e-013

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:46:39 AM
Dose Point # 1 - (780,5000,5000) cm

Co-60

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.904e-005	3.434e-004
Photon Energy Fluence Rate	MeV/cm²/sec	2.435e-005	4.366e-004
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	4.264e-008	7.655e-007
Absorbed Dose Rate in Air	mGy/hr	3.722e-010	6.683e-009
"	mrad/hr	3.722e-008	6.683e-007
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	4.224e-010	7.586e-009
o Opposed	"	3.641e-010	6.536e-009
o Rotational	"	3.641e-010	6.536e-009
o Isotropic	"	3.254e-010	5.839e-009
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	4.496e-010	8.075e-009
o Opposed	"	4.330e-010	7.776e-009
o Rotational	"	4.330e-010	7.776e-009
o Isotropic	"	3.446e-010	6.185e-009
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	3.804e-010	6.832e-009
o Posterior/Anterior	"	3.506e-010	6.295e-009
o Lateral	"	2.792e-010	5.008e-009
o Rotational	"	3.158e-010	5.668e-009
o Isotropic	"	2.796e-010	5.018e-009

MN-54

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 11:04:04 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	9.506e-007	3.449e-005
Photon Energy Fluence Rate	MeV/cm²/sec	7.936e-007	2.879e-005
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	1.502e-009	5.448e-008
Absorbed Dose Rate in Air	mGy/hr	1.311e-011	4.756e-010
"	mrad/hr	1.311e-009	4.756e-008
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.523e-011	5.524e-010
o Opposed	"	1.257e-011	4.559e-010
o Rotational	"	1.257e-011	4.559e-010
o Isotropic	"	1.111e-011	4.030e-010
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.630e-011	5.914e-010
o Opposed	"	1.557e-011	5.647e-010
o Rotational	"	1.557e-011	5.647e-010
o Isotropic	"	1.189e-011	4.314e-010
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	1.358e-011	4.928e-010
o Posterior/Anterior	"	1.220e-011	4.426e-010
o Lateral	"	9.291e-012	3.370e-010
o Rotational	"	1.093e-011	3.963e-010
o Isotropic	"	9.419e-012	3.417e-010

5b-125

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 11:05:19 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	6.198e-008	4.227e-006
Photon Energy Fluence Rate	MeV/cm²/sec	3.724e-008	2.473e-006
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	7.255e-011	4.820e-009
Absorbed Dose Rate in Air	mGy/hr	6.333e-013	4.208e-011
"	mrad/hr	6.333e-011	4.208e-009
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	7.563e-013	5.038e-011
o Opposed	"	5.980e-013	3.969e-011
o Rotational	"	5.980e-013	3.969e-011
o Isotropic	"	5.288e-013	3.510e-011
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	8.007e-013	5.328e-011
o Opposed	"	7.588e-013	5.047e-011
o Rotational	"	7.588e-013	5.047e-011
o Isotropic	"	5.648e-013	3.749e-011
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	6.669e-013	4.439e-011
o Posterior/Anterior	"	5.848e-013	3.887e-011
o Lateral	"	4.291e-013	2.845e-011
o Rotational	"	5.220e-013	3.467e-011
o Isotropic	"	4.422e-013	2.935e-011

Pu-238

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 11:09:58 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	5.584e-026	2.808e-024
Photon Energy Fluence Rate	MeV/cm²/sec	3.088e-027	1.553e-025
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	6.890e-030	3.465e-028
Absorbed Dose Rate in Air	mGy/hr	6.015e-032	3.025e-030
"	mrad/hr	6.015e-030	3.025e-028
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.034e-031	5.202e-030
o Opposed	"	5.340e-032	2.686e-030
o Rotational	"	5.051e-032	2.540e-030
o Isotropic	"	4.944e-032	2.486e-030
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.032e-031	5.188e-030
o Opposed	"	7.038e-032	3.540e-030
o Rotational	"	7.038e-032	3.540e-030
o Isotropic	"	5.229e-032	2.630e-030
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	7.884e-032	3.965e-030
o Posterior/Anterior	"	5.515e-032	2.774e-030
o Lateral	"	3.257e-032	1.638e-030
o Rotational	"	4.650e-032	2.338e-030
o Isotropic	"	3.783e-032	1.903e-030

Pu-239

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 11:10:41 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.152e-016	4.535e-014
Photon Energy Fluence Rate	MeV/cm²/sec	1.301e-017	5.121e-015
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.013e-020	7.920e-018
Absorbed Dose Rate in Air	mGy/hr	1.757e-022	6.914e-020
"	mrad/hr	1.757e-020	6.914e-018
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.842e-022	1.118e-019
o Opposed	"	1.600e-022	6.294e-020
o Rotational	"	1.609e-022	6.331e-020
o Isotropic	"	1.535e-022	6.042e-020
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.830e-022	1.114e-019
o Opposed	"	2.263e-022	8.904e-020
o Rotational	"	2.263e-022	8.904e-020
o Isotropic	"	1.636e-022	6.439e-020
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.462e-022	9.688e-020
o Posterior/Anterior	"	1.947e-022	7.663e-020
o Lateral	"	1.205e-022	4.742e-020
o Rotational	"	1.662e-022	6.540e-020
o Isotropic	"	1.325e-022	5.216e-020

Pu-240

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 11:11:24 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.825e-026	8.603e-025
Photon Energy Fluence Rate	MeV/cm²/sec	9.912e-028	4.674e-026
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.276e-030	1.073e-028
Absorbed Dose Rate in Air	mGy/hr	1.987e-032	9.369e-031
"	mrad/hr	1.987e-030	9.369e-029
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	3.395e-032	1.601e-030
o Opposed	"	1.750e-032	8.253e-031
o Rotational	"	1.654e-032	7.797e-031
o Isotropic	"	1.617e-032	7.623e-031
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	3.388e-032	1.598e-030
o Opposed	"	2.307e-032	1.088e-030
o Rotational	"	2.307e-032	1.088e-030
o Isotropic	"	1.715e-032	8.089e-031
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.566e-032	1.210e-030
o Posterior/Anterior	"	1.786e-032	8.420e-031
o Lateral	"	1.053e-032	4.966e-031
o Rotational	"	1.508e-032	7.112e-031
o Isotropic	"	1.226e-032	5.782e-031

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:52:18 AM
Dose Point # 1 - (780,5000,5000) cm

AmZ41

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.751e-020	1.852e-018
Photon Energy Fluence Rate	MeV/cm²/sec	1.177e-021	1.259e-019
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.084e-024	2.204e-022
Absorbed Dose Rate in Air	mGy/hr	1.819e-026	1.924e-024
"	mrad/hr	1.819e-024	1.924e-022
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	3.223e-026	3.412e-024
o Opposed	"	1.690e-026	1.791e-024
o Rotational	"	1.618e-026	1.716e-024
o Isotropic	"	1.603e-026	1.701e-024
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	3.196e-026	3.382e-024
o Opposed	"	2.259e-026	2.396e-024
o Rotational	"	2.259e-026	2.396e-024
o Isotropic	"	1.664e-026	1.764e-024
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.617e-026	2.780e-024
o Posterior/Anterior	"	1.925e-026	2.049e-024
o Lateral	"	1.162e-026	1.239e-024
o Rotational	"	1.622e-026	1.727e-024
o Isotropic	"	1.324e-026	1.410e-024

Cm-243

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:55:04 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.210e-010	3.335e-008
Photon Energy Fluence Rate	MeV/cm²/sec	3.258e-011	8.907e-009
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	6.086e-014	1.661e-011
Absorbed Dose Rate in Air	mGy/hr	5.313e-016	1.450e-013
"	mrad/hr	5.313e-014	1.450e-011
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	7.061e-016	1.930e-013
o Opposed	"	4.852e-016	1.324e-013
o Rotational	"	4.853e-016	1.324e-013
o Isotropic	"	4.379e-016	1.195e-013
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	7.177e-016	1.960e-013
o Opposed	"	6.628e-016	1.809e-013
o Rotational	"	6.628e-016	1.809e-013
o Isotropic	"	4.691e-016	1.281e-013
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	6.090e-016	1.664e-013
o Posterior/Anterior	"	5.048e-016	1.379e-013
o Lateral	"	3.428e-016	9.360e-014
o Rotational	"	4.418e-016	1.207e-013
o Isotropic	"	3.665e-016	1.001e-013

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:56:00 AM
Dose Point # 1 - (780,5000,5000) cm

Cm-244

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	2.011e-025	1.127e-023
Photon Energy Fluence Rate	MeV/cm²/sec	1.143e-026	6.409e-025
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.445e-029	1.370e-027
Absorbed Dose Rate in Air	mGy/hr	2.134e-031	1.196e-029
"	mrad/hr	2.134e-029	1.196e-027
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	3.703e-031	2.075e-029
o Opposed	"	1.916e-031	1.074e-029
o Rotational	"	1.816e-031	1.018e-029
o Isotropic	"	1.780e-031	9.979e-030
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	3.689e-031	2.067e-029
o Opposed	"	2.527e-031	1.416e-029
o Rotational	"	2.527e-031	1.416e-029
o Isotropic	"	1.874e-031	1.050e-029
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.857e-031	1.601e-029
o Posterior/Anterior	"	2.014e-031	1.129e-029
o Lateral	"	1.192e-031	6.680e-030
o Rotational	"	1.694e-031	9.496e-030
o Isotropic	"	1.381e-031	7.738e-030

MicroShield v5.05 (5.05-00201)

Maine Yankee

Conversion of calculated exposure in air to dose

FILE: C:\MS5\DATA\SOILFL.MS5

Case Title: Soil Fill

This case was run on Thursday, March 15, 2001 at 10:58:51 AM

Dose Point # 1 - (780,5000,5000) cm

Eu-152

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	6.572e-006	1.107e-004
Photon Energy Fluence Rate	MeV/cm²/sec	9.187e-006	1.467e-004
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	1.567e-008	2.528e-007
Absorbed Dose Rate in Air	mGy/hr	1.368e-010	2.207e-009
"	mrad/hr	1.368e-008	2.207e-007
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.550e-010	2.506e-009
o Opposed	"	1.347e-010	2.168e-009
o Rotational	"	1.347e-010	2.168e-009
o Isotropic	"	1.207e-010	1.940e-009
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.643e-010	2.659e-009
o Opposed	"	1.588e-010	2.567e-009
o Rotational	"	1.588e-010	2.567e-009
o Isotropic	"	1.272e-010	2.048e-009
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	1.397e-010	2.256e-009
o Posterior/Anterior	"	1.293e-010	2.084e-009
o Lateral	"	1.041e-010	1.669e-009
o Rotational	"	1.167e-010	1.879e-009
o Isotropic	"	1.039e-010	1.667e-009

Eu-154

MicroShield v5.05 (5.05-00201)
Maine Yankee
Conversion of calculated exposure in air to dose
FILE: C:\MS5\DATA\SOILFL.MS5
Case Title: Soil Fill
This case was run on Thursday, March 15, 2001 at 10:59:56 AM
Dose Point # 1 - (780,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.002e-005	1.593e-004
Photon Energy Fluence Rate	MeV/cm²/sec	1.440e-005	2.195e-004
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.442e-008	3.749e-007
Absorbed Dose Rate in Air	mGy/hr	2.132e-010	3.273e-009
"	mrad/hr	2.132e-008	3.273e-007
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.414e-010	3.712e-009
o Opposed	"	2.103e-010	3.223e-009
o Rotational	"	2.103e-010	3.223e-009
o Isotropic	"	1.886e-010	2.887e-009
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.556e-010	3.934e-009
o Opposed	"	2.473e-010	3.802e-009
o Rotational	"	2.473e-010	3.802e-009
o Isotropic	"	1.985e-010	3.043e-009
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	2.176e-010	3.343e-009
o Posterior/Anterior	"	2.018e-010	3.094e-009
o Lateral	"	1.629e-010	2.488e-009
o Rotational	"	1.822e-010	2.792e-009
o Isotropic	"	1.624e-010	2.483e-009

Attachment 6-2
BNL Kd Report for Fill

Backfill Materials for the Maine Yankee Site

Bulk Density and Partition Coefficients for Co, Cs, Sr, and Ni

Revised October 17, 2001

Mark Fuhrmann and Biays Bowerman
Environmental Sciences Department
Brookhaven National Laboratory

Partition Coefficients

Method

To determine the partition coefficients (K_D) of Co, Cs, Sr, and Ni, four materials from the Maine Yankee site were exposed to low activity tracers of ^{57}Co, ^{137}Cs, ^{85}Sr, and ^{63}Ni. The tracers each were prepared by initially diluting them from the "as received" concentrations of 100 Ci/mL to 4.76 Ci/mL. Two mL of each of the first three tracers (^{57}Co, ^{85}Sr, and ^{137}Cs) were mixed together and the pH was adjusted to 6.0, giving a final concentration of each tracer in the mixture of 0.476 Ci/mL. Stock solutions of ^{63}Ni were prepared separately because this pure beta-emitter had to be counted in a liquid scintillation counter.

^{57}Co, ^{85}Sr, and ^{137}Cs

For each sample of material to be tested, the contact solution was prepared by weighing out 44 g of distilled water into a plastic bottle and adding 1.0 mL of mixed tracer solution. The contact solution had a concentration of each tracer of 0.01 Ci/mL. The solution was mixed and 5 mL were removed and pipetted into a plastic counting vial. These 5 mL samples became the reference solutions against which the samples of liquid were compared after contact with the solids. Approximately 2 grams of each solid was weighed out and placed in the individual bottles of tracer. Four samples of each solid were prepared.

One of the bottles of each set was sampled at 24 hours, again at 72 hours, and a third time at 168 hours to check the uptake kinetics. Sampling was done by removing about 5 mL of solution by plastic syringe and then filtering the liquid through a syringe filter (0.45 m). This liquid was then pipetted into preweighed vials, which were reweighed to get the weight of the liquid.

Both the reference samples and the actual contact solutions were counted on an intrinsic germanium gamma detector with a Canberra spectroscopy system. The ^{57}Co, ^{85}Sr, and ^{137}Cs were measured at the 122, 514, and 661 keV gamma energies respectively. Because reference solutions were used for each of the triplicate samples, there was no need to calculate activities of the post-contact samples. Instead counts per minute per gram (CPM/g) were compared directly and used in calculation of K_D.

The first set of tracer solutions was sampled after contact with the solids for 24, 72, and 168 hours. The other three from each set were left in contact for 144 hours. Kinetics results are shown in Figures 1 to 3, indicating that uptake for both tracers was essentially complete. The partition coefficient is calculated as the concentration of an element of interest sorbed on the solid phase, divided by that elements final concentration the liquid with which the solid was in contact. Results for ^{57}Co are shown in Table 1. Results for ^{85}Sr are shown in Table 2. Results for ^{137}Cs are shown in Table 3. The pH of samples was measured after 336 hours; Clay A = 5.5, Crushed Rock A = 6.85, Sand = 5.32, and Gravel = 4.95.

Table 1.
Partition Coefficients for ^{57}Co

Sample	Individual K_D Values for ^{57}Co	Average K_D
Clay	350, 455, 467, 516	447
Crushed Rock	156, 224, 204, 206	198
Bank Run Sand	368, 512, 557, 536	493
Bank Run Gravel	126, 460, 524, 493	401

Table 2.
Partition Coefficients for ^{85}Sr

Sample	Individual K_D Values for ^{85}Sr	Average K_D
Clay	436, 438, 486, 474	459
Crushed Rock	41, 32, 48, 40	40
Bank Run Sand	143, 264, 275, 274	239

Bank Run Gravel	69, 160, 178, 181	147

Table 3.
Partition Coefficients for ^{137}Cs

Sample	Individual K_D Values for ^{137}Cs	Average K_D
Clay	1056, 998, 1136, 1337	1133
Crushed Rock	810, 1387, 1166, 1401	1191
Bank Run Sand	1105, 2250, 2140, 2337	1958
Bank Run Gravel	384, 1110, 1288, 1224	1001

^{63}Ni

Solutions containing ^{63}Ni were prepared separately to allow for liquid scintillation counting. Preparations of the contact solutions were identical to those for the gamma emitting radionuclides. Sampling for measurements were different in that 1.0 mL of contact solution was withdrawn for counting, and mixed with 10 mL of Packard Ultima Gold liquid scintillation cocktail. An initial experimental solution was prepared and sampled after 24 hours. The remaining three solutions were sampled after 144 hours, since it was assumed that Ni would exhibit sorption kinetics similar to Co. Blank samples were also prepared to verify that leachable chemical constituents of the materials tested did not affect the quenching properties of the scintillator material. Samples were counted on a Wallac DSA for one minute each. Results for ^{63}Ni are shown in Table 4. Values for pH of the blank contact solutions were determined after 168 hours.

Table 4.
Partition Coefficients for ^{63}Ni

Sample	Individual K_D Values for ^{63}Ni	Average K_D
Clay	262, 420, 565, 258	376
Crushed Rock	198, 199, 215, 224	209

Bank Run Sand	220, 446, 521, 542	432
Bank Run Gravel	121, 240, 294, 402	264

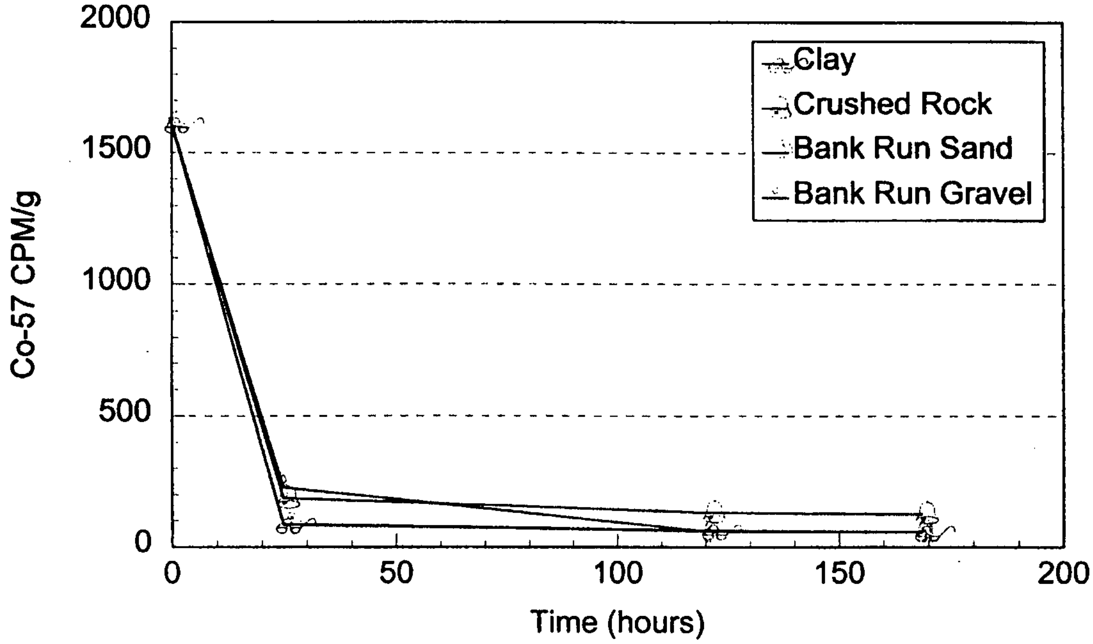

Figure 1. Uptake kinetics for Co-57 for the backfill materials.

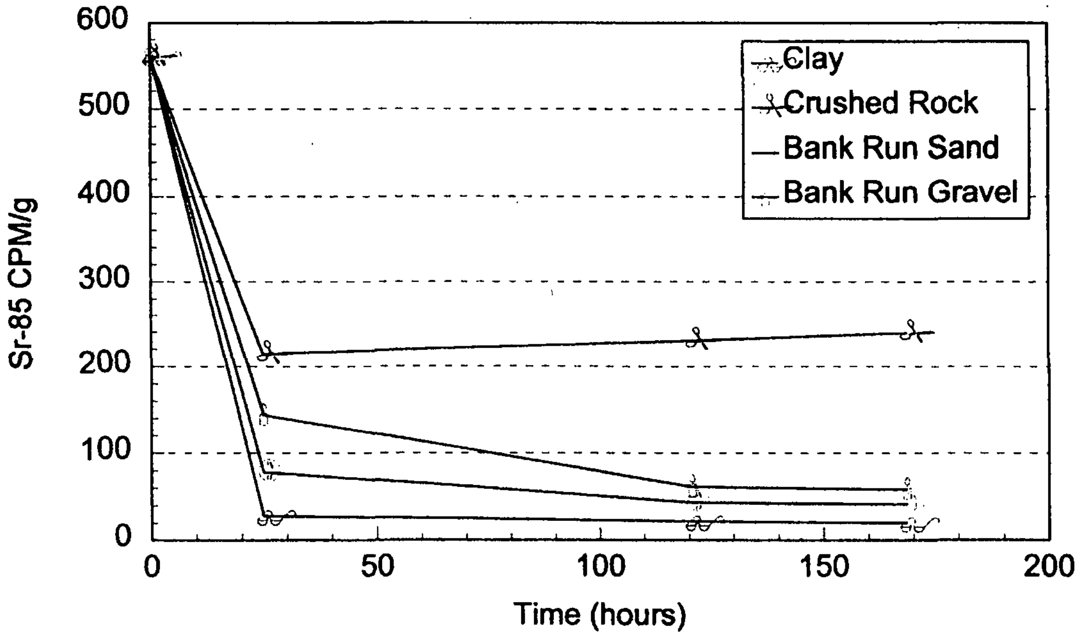

Figure 2. Uptake kinetics for Sr-85 for the backfill materials.

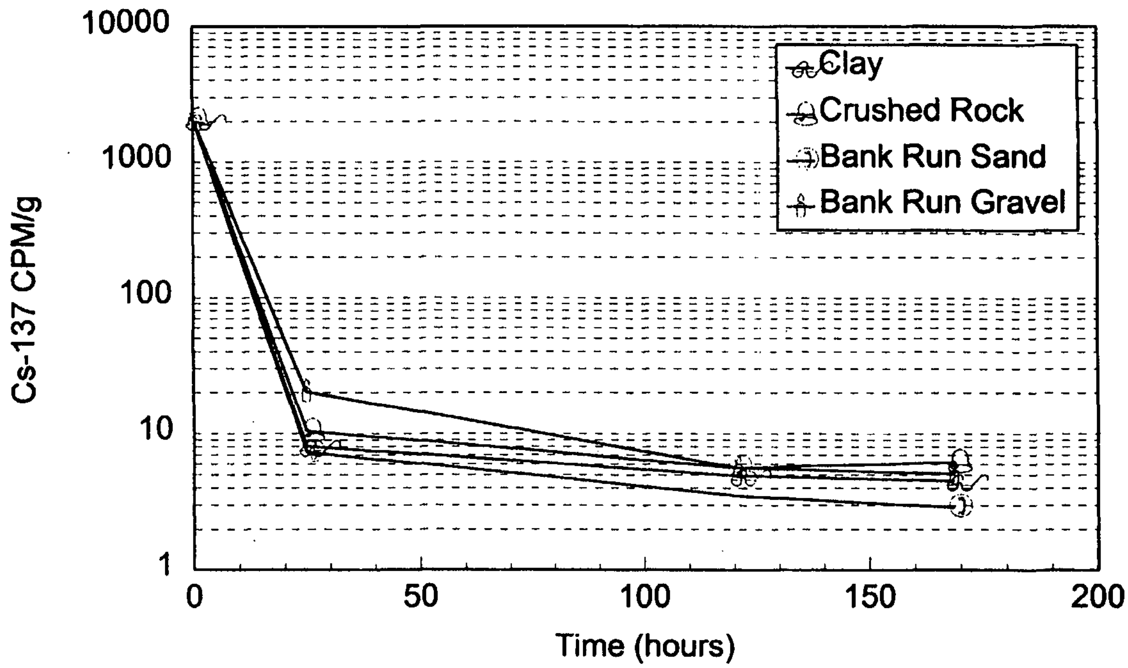

Figure 3. Uptake kinetics of Cs-137 for the backfill materials.

Please note that for the bank run gravel samples, we were not able to do the experiments with a large enough mass of solid that it would provide a representative value. Consequently the bank run gravel material was passed through a 4 mm (#5) sieve and that material was used for the K_D tests. To obtain the proper K_D for this material, the K_D should be multiplied by the fraction of material that passed the 4 mm sieve. We determined that 0.44 (44%) of the material was less than 4 mm; so each K_D value should be multiplied by 0.44 to obtain the correct value. Because our sample was relatively small for material containing so much gravel, it is advisable to check with the supplier to find out what fraction of the bulk material passes a No.5 sieve, and then correcting the K_D with that value.

Bulk Density

Method

Bulk density was determined in triplicate for the four materials. The bulk density of the clay was determined by placing a large (about 200 cc), preweighed, bolus of clay into a measured volume of water in a large graduated cylinder. The volume of the clay was determined by displacement. The other samples were not coherent and were poured into a graduated cylinder and were tamped down. The volume was measured from the graduations on the cylinder and then the sample was decanted and weighed.

Results

These data are plotted on Figure 4, with the slope being the bulk density. All results are linear (typical R^2 values were 0.98 or better) indicating good reproducibility. However, the bank run gravel samples did not produce a line that approached the origin. This indicates that we were not able to get a consistent mixture of sand and gravel for the samples. Bulk density values determined both as the slope from Figure 4 and by average are given in Table 5. It is recommended that the average values be used; the plots of the data and slope values are included to illustrate the small scatter in these determinations.

Table 5.
Bulk Density of Backfill Materials

Sample	Bulk density from the slope (g/cc)	Bulk density from averages (g/cc)
Clay	2.13	2.18
Bank Run Sand	1.31	1.47
Bank Run Gravel	1.20	1.70
Crushed Rock	1.63	1.63

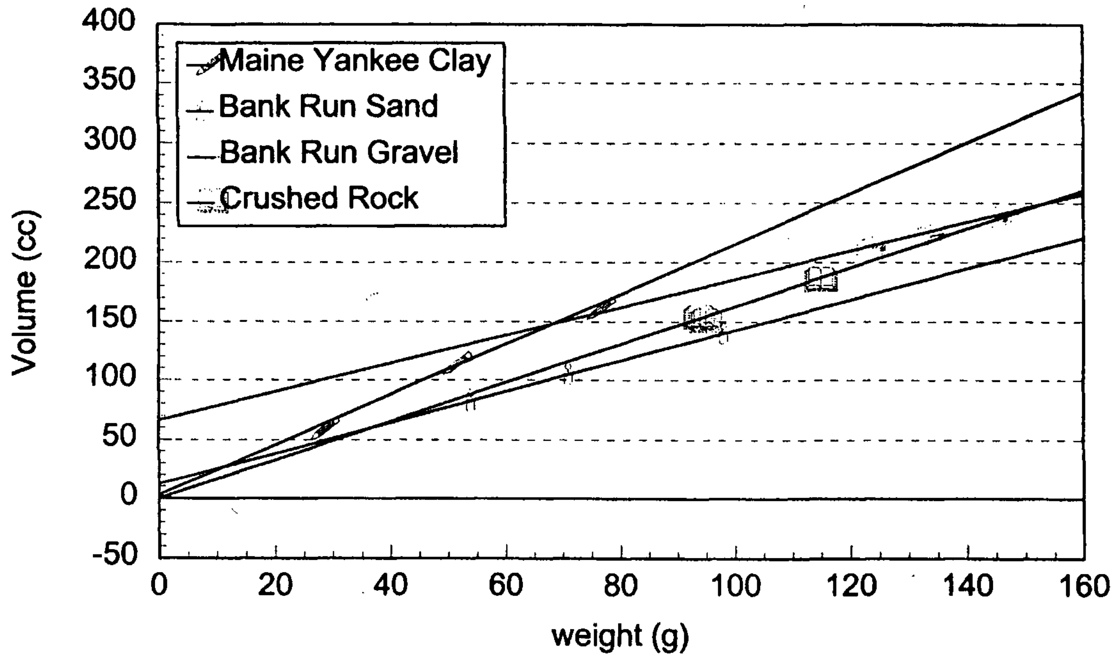

Figure 4. Data for bulk density of backfill material.

Partition Coefficients for Co, Cs, and Sr for Sand and Cemetitious Backfill Materials for the Maine Yankee Site

October 18, 2001

Mark Fuhrmann
Environmental Sciences Department
Brookhaven National Laboratory

Partition Coefficients

Method

To determine the partition coefficients (K_D) of Co, Cs, and Sr, two materials from the Maine Yankee site were exposed to low activity tracers of ^{57}Co, ^{137}Cs, and ^{85}Sr. The tracers each were prepared by initially diluting them from the "as received" concentrations of 100 Ci/mL to 4.76 Ci/mL. Two mL of each were mixed together and the pH was adjusted to 6.0, giving a final concentration of each tracer in the mixture of 0.476 Ci/mL.

For each sample of material to be tested, the contact solution was prepared by weighing out 44 g of distilled water into a plastic bottle and adding 0.5 mL of mixed tracer solution. The contact solution had a concentration of each tracer of 0.01 Ci/mL, or less. The solution was mixed and 2 mL were removed and pipetted into a plastic counting vial. These 2 mL samples became the reference solutions against which the samples of liquid were compared after contact with the solids. Approximately 2 grams of each solid (material that passed a 1mm sieve) were weighed out and placed in the individual bottles of tracer. After the 114 hour contact time, liquid samples were taken by removing about 3 mL of solution by plastic syringe and then filtering the liquid through a syringe filter (0.45 m). Two mL of this liquid were then pipetted into preweighed vials, which were reweighed to get the weight of the liquid.

Both the reference samples and the actual contact solutions were counted on an intrinsic germanium gamma detector with a Canberra spectroscopy system. The ^{57}Co, ^{85}Sr, and

^{137}Cs were measured at the 122, 514, and 661 keV gamma energies respectively. Because reference solutions were used for each of the triplicate samples, there was no need to calculate activities of the post-contact samples. Instead counts per minute per gram (CPM/g) were compared directly and used in calculation of K_D

The partition coefficient is calculated as the concentration of an element of interest sorbed on the solid phase, divided by that elements final concentration the liquid with which the solid was in contact. Results for ^{57}Co are shown in Table 1. Results for ^{85}Sr are shown in Table 2. Results for ^{137}Cs are shown in Table 3.

Table 1.
Partition Coefficients for ^{57}Co

Material	Individual K_D Values for ^{57}Co (mL/g)	Average K_D
Trial batch 99-932.2	188, 192, 186	189
Wiscasset Sand	633, 525, 597	585

Table 2.
Partition Coefficients for ^{85}Sr

Material	Individual K_D Values for ^{85}Sr (mL/g)	Average K_D
Trial batch 99-932.2	102, 88, 77	89
Wiscasset Sand	1031, 761, 770	854

Table 3.
Partition Coefficients for ^{137}Cs

Material	Individual K_D Values for ^{137}Cs (mL/g)	Average K_D
Trial batch 99-932.2	109, 130, 113	117
Wiscasset Sand	30980, 24400, 23340	26,200

The K_D values for the sand are significantly higher than those of the cement mix (Trial Batch 99-932.2). The cement supplies a large quantity of ions to solution, which compete with the radionuclide tracers for sorption sites on the sand. In addition it has agglomerated the sand so that sorption, which is a surface area based process, cannot proceed very effectively. The very high K_D values for ^{137}Cs on the sand is an estimate because almost all of the ^{137}Cs was sorbed by the sand, removing it from solution. Consequently the count rates for the liquid were very low; about 0.2 cpm/g, giving poor statistics even though count times were as long as 833 minutes. As a result, small changes in the very low count rate result in large changes in the K_D.

Both materials tested were passed through a 1mm sieve. The Wiscasset Sand contained 72.5% material that was less than 1 mm. The cement material was gently disaggregated with a spatula. It contained 67.6% material less than 1 mm. Assuming that material greater than 1mm has little or no capacity to sorb, the K_D needs to be corrected for the coarse fraction. Each K_D value should be multiplied by 0.725 or 0.676 (for the sand or cement materials respectively) to obtain the correct value. Because our sample was relatively small for material containing so much gravel, it is advisable to check with the supplier to find out what fraction of the bulk material passes a 1 mm sieve, and then correcting the K_D with that value.

Attachment 6-3
BNL Kd Report for Concrete

TECHNICAL EVALUATION
OF
BNL K_d and Diffusion Coefficient Determination

TE-99-041

Purpose

This evaluation documents the determination of partition and diffusion coefficients for concrete samples from Maine Yankee which were used to support the dose evaluation section (section 6) of the License Termination Plan. The studies were conducted by Brookhaven National Laboratory.

References

1. "Leaching and Sorption of Radionuclides: Structural Concrete from Maine Yankee Nuclear Power Station", BNL, October 21, 1999.

Assumptions

None

Method

Six samples of contaminated concrete and three sets of clean concrete were crushed and submitted for testing. An Accelerated Leach Test was performed on five of the contaminated samples using ASTM C-1308 methods. The leachant volume was 10 times the surface area of the solid samples and was composed of 1 liter of distilled/deionized water. Forty milliliter aliquots were removed at specified intervals for gamma spectroscopy. Estimated detection limits for both Cs-137 and Co-60 were 40 pCi/l. Count rates were converted to pCi/l and input to the Accelerated Leach Test (ALT) computer model. The ALT code output is a table of Incremental Fraction Leached (IFL) and the Cumulative Fraction Leached (CFL). The effective diffusion coefficient and goodness-of-fit were determined for both the Diffusion and Partition models.

The partition coefficient (K_d) was determined for Cs-137 and Sr-90. Pieces of crushed concrete were immersed in distilled water containing the nuclide of interest. Uptake kinetics were determined by taking aliquots periodically, counting them and then returning them to the sample container. At the end of the test period, samples were filtered and counted. Sample count rates and reference count rates were determined. The K_d value was determined by dividing the count rate per gram of sample by the count rate per milliliter in the liquid. The pH of the leachate was also determined.

The DUST-MS code was used to determine the best fit effective diffusion coefficients from the experimental data.

The details of the analytical methods are contained in reference 1 attached.

Conclusions

Based on the goodness-of-fit results, diffusion is the transport mechanism for concrete. The effective diffusion coefficient for Cs-137 was 2E-10 cm^2/sec. K_d values for cesium and strontium averaged 3.0 mL/g and 1.0 mL/g respectively.

Plant specific values of diffusion and partition coefficients have been determined for use in performing dose assessment calculations to support section 6 of the LTP.

Prepared By: [signature] Date: 11/24/99

Reviewed By: Robert F Decker Date: 11/29/99

Leaching and Sorption of Radionuclides: Structural Concrete from Maine Yankee Nuclear Power Station

November 9, 1999

Mark Fuhrmann and Terrence Sullivan
516-344-2224
fuhrmann@bnl.gov

Environmental and Waste Technology Group
Brookhaven National Laboratory
Upton, NY 11973-5000

Summary

Six samples of contaminated structural concrete from the Maine Yankee Nuclear Power Station were received at Brookhaven National Laboratory (BNL) for leach testing. The leach test used is designed to determine if diffusion is the dominant rate controlling release mechanism from porous materials. If so the test method and computer code associated with it can be used to quantify the effective diffusion coefficient (D_e). This approach assumes a homogeneously distributed contaminant in the leached sample. However, there is evidence that the contaminants are actually in a thin layer (1 mm or less) on the surface of the concrete core samples. To estimate an effective diffusion coefficient that is more representative of this condition, the DUST code was applied to the experimental data. As described in Appendix B, after reevaluating the leach rate relative to the geometry of the contaminant, the best fit D_e for ^{137}Cs from the sample with the greatest leach rate (sample 4A) was 2 x 10^{-10} cm^2/sec.

Three sets of uncontaminated, crushed concrete were tested to determine partition coefficients (K_d) for ^{137}Cs and ^{85}Sr. With these tests the quantities of ^{137}Cs and ^{85}Sr that sorbed onto the fractured concrete were determined. Uptake of ^{137}Cs yielded a K_d of 3.0. For ^{90}Sr the K_d was 3.0. These values can be used as input to the DUST code to determine how much sorption reduces releases from the facility.

METHODS

The Accelerated Leach Test (ASTM C-1308) was started for five samples on September 14, 1999. With the observation that these samples all had coatings of paint or epoxy on them, one of the samples was removed from testing and two additional samples (with the epoxy removed) were sent to BNL. These samples were added to the test set, starting on September 20, 1999. The leach test was run according to the test protocol. The leachant volume was 10 times the surface area of the solid sample, with the volume of distilled/deionized water used for each sample, in each interval, being about 1.0 liter. All weighing was done on calibrated and recently certified balances. Sample parameters are given in Table 1.

Aliquots of 40 mL were taken at each interval for gamma spectroscopy. All samples were counted in the same geometry containers. Two intrinsic germanium gamma detectors were used. Each was calibrated with a NIST traceable mixed gamma standard (#678-59) from Isotope Products Laboratories. This standard contained both ^{137}Cs and ^{60}Co, which allowed direct comparisons to the leaching samples. The standard was diluted and counted in the same geometry as the samples, on each detector. At the end of the counting campaign, samples of distilled/deionized water were counted as blanks, again in the same geometry, for 2000 minutes each. One detector was observed to have a low background for ^{137}Cs, which was subtracted from the data obtained with that detector.

Count rates were converted to pCi/L on an Excel spread sheet, and then input to the Accelerated Leach Test (ALT) computer model. Estimated detection limits for both ^{137}Cs and ^{60}Co are 40 pCi/L. The parameters used in the calculations are shown in Table 1 for each sample. Spread sheets for each sample are included in Appendix A. Output of the code is a sheet that tabulates the Incremental Fraction Leached (IFL) and the Cumulative Fraction Leached (CFL). The effective diffusion coefficient and "goodness-of-fit" parameter are also given for both the Diffusion and Partition models. Figures showing the CFL as a function of time are also included in Appendix A.

The partition coefficient (K_d) was experimentally determined for ^{137}Cs and ^{90}Sr. Pieces of broken concrete from Maine Yankee were contacted with distilled water (adjusted to pH = 7.0) that contained the radionuclide of interest. As shown in Fig.1, 3-6 pieces were used, each being about 2 cm. Uptake kinetics were determined by taking periodic samples of the water, counting it and then returning it to the experiment. Experiments were started by weighing out distilled water, adding tracer, and then taking an aliquot as a reference. At the end of the experiment about 5 mL were withdrawn with a plastic syringe, and the water was pushed through a 0.45 micron syringe filter. Aliquots were pipetted into preweighed vials; the vials were then reweighed and counted. Uptake on the concrete was determined by taking the difference in count rates between the sample and the reference. The K_d is the count rate per gram on the solid divided by the count rate per mL in the liquid, at steady-state. After sampling, pH measurements were taken from the leachate. The instrument was standardized with newly made pH reference solutions at 7.0 and 4.01.

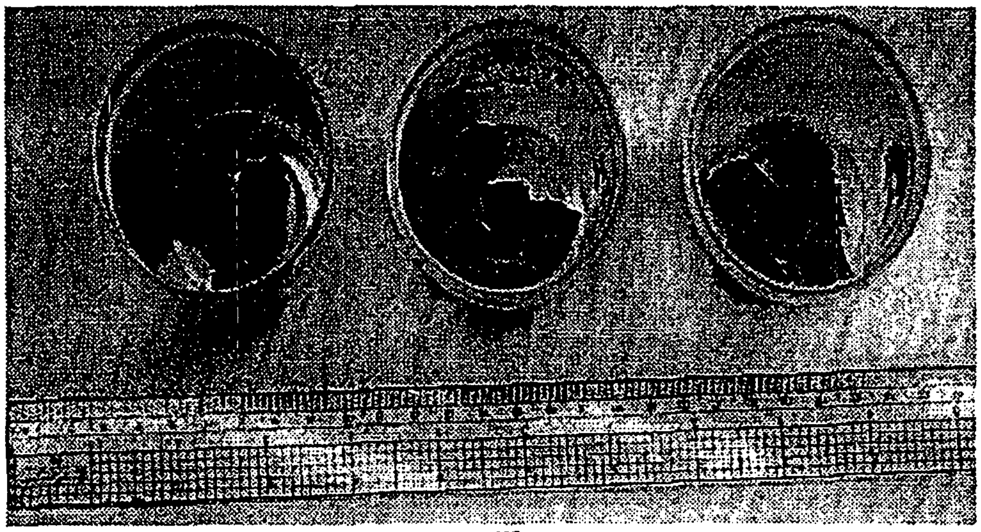

Figure 1. Samples of concrete used in the ^{137}Cs experiment for K_d determination.

Table 1.

Concrete Samples for Leach Testing

Sample	Weight (g)	Diameter (cm)	Height (cm)	Leachant Volume (L)	Source Term Cs-137 (pCi)	Source Term Co-60 (pCi)
1A	117.4	6.97	1.41	1.070	372,100	200
2A	105.4	6.89	1.36	1.040	553,300	359,700
4A	125.2	6.91	1.46	1.070	249,200	200
8A*	103.8	6.92	1.16	1.000	25,000	31,900
32A	93.8	6.95	1.17	1.010	336,000	19,100
41A	101.0	6.92	1.42	1.060	113,000	21,100

* Sample 8A contains 30,100 pCi of Eu-152.

RESULTS

Results of the leach tests are summarized in Table 2. Effective diffusion coefficients for ^{137}Cs range from 9 x 10^{-9} to 5 x 10^{-11} cm^2/sec. One of the bare concrete samples leached the fastest while the other was very slow. Alternatively, there may still be some epoxy in the pores of the bare concrete with the low leach rate. Effective diffusion coefficients for ^{60}Co were about the same or somewhat lower than ^{137}Cs. In some of the samples, inventories were so low that no ^{60}Co could be detected in the leachate. No releases of ^{152}Eu were observed from sample 8A.

Releases of ^{137}Cs fit the diffusion model very well. Generally if the goodness-of-fit parameter is less than 2%, the fit of the model to the data indicates diffusion as the transport mechanism. All of the samples have goodness-of-fit values for ^{137}Cs lower than 2%; with most being significantly lower than 1%. The samples (2A and 41A) with high values had thick layers of epoxy on their contaminated surfaces. For these samples several processes or rates may be controlling ^{137}Cs releases. It is likely that diffusion at several rates (from the epoxy and the concrete) presents an averaged rate to produce the observed leaching curve. For ^{60}Co, sample 2A had a goodness-of-fit value of 0.35% while 8A had 2.38%. Leaching data, ALT output, and figures showing the cumulative fraction released as a function of time, are given in Appendix A.

The results presented above assume that the source term is distributed, homogeneously, through the entire sample. However, there is evidence that the activity on the concrete is actually in a 1 mm thick layer at the concrete surface. Because the diffusion coefficient is very sensitive to the path length through the sample, the ALT model was run using a 1mm thickness for sample 4A as well as the measured 14 mm. At 14 mm, D_e for ^{137}Cs was 8 x 10^{-9} cm^2/sec. When the thickness was altered to 1 mm, D_e for ^{137}Cs was 7 x 10^{-11} cm^2/sec, with no change in the "goodness-of-fit" parameter. The diffusion coefficient responds to the reduction in thickness by becoming lower by almost two orders of magnitude, in order to keep the fraction released the same (as was observed experimentally). This estimated D_e is based on releases from both sides of a uniformly contaminated cylinder. There is evidence that the contamination actually resides in a 1mm thick layer which is backed by clean concrete. To examine this case in detail the DUST-MS computer code was used. Results of this analysis are discussed in Appendix B. From this modeling a best fit effective diffusion coefficient for Cs (for sample 4A) was estimated to be 2×10^{-10} cm^2/sec. This is based on a 1 mm thick contaminated layer with clean concrete on one side and water on the other.

Effective diffusion coefficients for ^{60}Co were determined for only two of the six samples; those with the greatest inventories of the radionuclide. Concentrations were below detection limits in the leachate from the other samples. Observed values of D_e were 1×10^{-10} and 3×10^{-11}, which were calculated based on a homogeneous distribution of the contaminant in the sample. It is believed that ^{60}Co is actually present in a layer of about 0.2 mm thickness. This being the case, the value of D_e would decrease by an order of magnitude or more.

No ^{152}Eu was observed in the leachate although it was specifically searched for. This is not surprising because rare earths typically partition strongly to the solid phase. Moreover, sample 8A, the only one containing observable activities of this radionuclide, represents material that was activated. One would therefore expect ^{152}Eu to be retained in minerals in the aggregate (or the cement itself) that contained the traces of Eu that were activated.

Partition coefficients were determined for ^{137}Cs and ^{85} Sr in contact with structural concrete obtained from the site. Kinetics of uptake were examined for ^{137}Cs to determine the correct contact time for the experiment. Figure 2 shows that ^{137}Cs uptake follows a square root of time function and is mostly complete by 160 hours. The ^{137}Cs K_d sampling was done at that time. Results for ^{137}Cs and ^{85}Sr are given in tables 3 and 4 respectively. Values of K_d for ^{137}Cs average 3.0 mL/g; while for ^{85}Sr the average is 1.0 mL/g.

Table 2.

Summary of Radionuclide Leaching from Structural Concrete

Maine Yankee Nuclear Power Station

Sample	Description	D_e Cs-137 cm^2/S	Cs-137 goodness of fit	Fraction Leached Cs-137	D_e Co-60 cm^2/S	Fraction Leached Co-60	pH of final sample
1A	Bare concrete, epoxy removed	5.6×10^{-11}	0.22	0.019	---	---	7.9
2A	Concrete with two layers of epoxy, ~1mm thick	4.9×10^{-11}	1.74	0.015	1.03×10^{-10}	0.027	6.9
4A	Bare concrete, epoxy removed	8.0×10^{-9}	0.35	0.195	---	---	6.5
8A	Concrete with a thin layer of white paint	3.3×10^{-9}	0.07	0.146	3.3×10^{-11}	0.019	7.6
32A	Concrete with a thin layer of white paint	4.6×10^{-10}	0.51	0.058	---	---	7.5
41A	Concrete with three layers ~ 2mm thick	8.0×10^{-11}	1.21	0.020	---	—	6.3

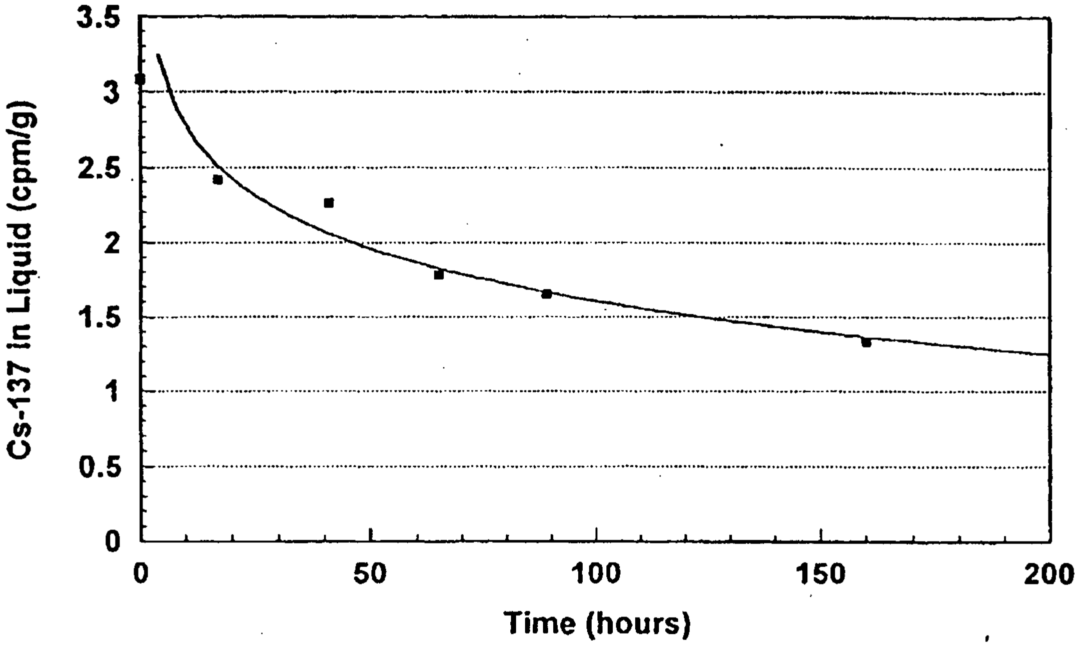

Figure 2. Kinetics of Cs-137 uptake on structural concrete from Maine Yankee.

Table 3.

K_d Values for Cs-137 in Contact with Yankee Concrete Samples

Contact Time: 160 hours

Sample	Start CPM/g	End CPM/g	Δ CPM/g	Liquid Volume (g)	Concrete Wt (g)	Counts on Solid CPM/g	K_d Cs-137	pH
1B	3.08	1.33	1.75	48.4	16.900	5.01	2.9	11.2
2B	2.93	1.82	1.11	48.6	18.294	2.95	2.7	11.5
3B	2.93	1.46	1.47	48.4	14.575	4.88	3.3	11.32

Table 4

K_d Values for Sr-85 in Contact with Yankee Concrete Samples

Contact Time: 143 hours

Sample	Start CPM/g	End CPM/g	Δ CPM/g	Liquid Volume (mL)	Concrete Wt (g)	Counts on Solid CPM/g	K_d Sr-85	pH
1	16.96	13.13	3.83	50.0	15.35	9.54	0.73	11.1
2	18.05	14.60	3.45	50.0	12.32	14.00	0.96	11.3
3	13.78	10.85	2.93	50.0	10.70	13.69	1.26	11.2

Attachment 6-4
Irrigation Memorandum

MEMO

Date: 10-20-99

To: George Pillsbury

From: Robert F. Decker

Subject: TE99-020 documentation of telephone calls from USDA/NRCS representatives

On September 1, 1999 an e-mail was sent to Norman Kalloch, USDA-NRCS representative in the Orono, ME (207 990-9100) regarding local (Lincoln County) well irrigation rates. Unfortunately, his office was in the process of changing over to a new server system and the message was not received until some time later. On September 10, 1999 Mr. Kalloch contacted me by telephone. During the conversation Mr. Kalloch confirmed that agriculture in Maine does not rely to any significant extent on well or surface water irrigation. The majority of irrigation occurs in the northern portion of the State and is primarily associated with potato crops. To provide local agricultural irrigation information Mr Kalloch directed me to contact Ms. Mary Thompson in Warren Maine. Ms. Thompson is the local Lincoln county extension representative (207 273- 2005).

On September 10, 1999 following the conversation with Mr. Kalloch I contacted Mary Thompson. Ms. Thompson stated that precipitation (rain) is the principal source of irrigation for family gardens. She said that local irrigation rates from wells would not be expected to exceed 4-5 inches per year for family gardens and 7-8 inches per year for commercial growers. She stated that these rates are relative to drought years, normal years would result in less well irrigation. She also stated that pumping cost for family and commercial growers is a contributing factor for the stated irrigation rates as are low well production rates especially during drought years where there is a greater concern for conserving the water for domestic usage. Ms. Thompson also stated that in the coastal region salt water intrusion of the well is also a consideration by local residents. This latter concern is a significant consideration during drought years. Ms. Thompson forwarded the latest copy of the USDA report for local irrigation and farm usage. Ms. Thompson concluded that the principal local commercial crops irrigated are strawberries.

On September 16, 1999 I was contacted by Paul Hughes (207 990-9100 #3) the USDA agronomist for Maine. Mr. Hughes confirmed the conditions and rates provided by Mr Kalloch and Ms. Thompson and reiterated the reasons and conditions provided by Ms. Thompson regarding local well usage. Mr. Hughes concluded that recommendations to commercial strawberries growers was to provide the crops one inch of water per week. The recommendations to commercial growers is to supplement their crops to a 1 inch per week rate with irrigation water if the weekly rainfall is less than one inch per week.

Attachment 6-5
Concrete Density

TECHNICAL EVALUATION
OF
Concrete Porosity and Density

TE-99-039

Purpose

This TE documents the porosity and density of Maine Yankee concrete samples as determined by Earth Engineering Services, Inc. of Baltimore, Maryland.

References

1. Letter dated June 9, 1999 from Earth Engineering.

2. ASTM C 642-97, "Standard Test Method for Density, Absorption, and Voids in Hardened Concrete.

Assumptions

None

Method

Clean concrete samples from Maine Yankee were submitted to Earth Engineering Services, Inc. for analysis in order to determine the porosity and density. ASTM test method C 642-97 was followed for performing the analyses.

The results of the analyses were as follows;

	Core# A9900/01FL2	Core# A9900/01MC3
Bulk Density	2.21 g/cc	2.27 g/cc
Porosity	14.6%	13.7%

Conclusion

The density and porosity of concrete at the Maine Yankee site have been determined using standard test methods.

Prepared By: [signature] Date: 11/22/99

Reviewed By: Robert H Decker Date: 12/3/99. Calculations (attached) checked for correctness & verified RHD

E2Si Earth Engineering & Sciences, Inc.

3401 CARLINS PARK DRIVE BALTIMORE, MARYLAND 21215 (410) 466-1400 FAX: (410) 466-7371

June 9, 1999

Maine Yankee Atomic Power Plant
Stone & Webster Decommissioning Team
Old Ferry and Bailey Point Road
Wiscasset, ME 04578

Attention: Mr. Robert J. Tozzie
Stone & Webster/ Radiological Services, Inc.

Re: Concrete Core Samples
Bulk Density & Porosity Tests
E2Si Project No. 99-160

Dear Mr. Tozzie:

Test results for the 2 core samples are summarized below.

	CORE #A9900/01FL2	CORE #A9900/01MC3
Bulk Density (Dry)	2.21	2.27
Bulk Density After Immersion	2.34	2.38
Bulk Density After Immersion & Boiling	2.36	2.40
Apparent Density	2.59	2.63
Permeable Pore Space (voids)	14.6%	13.7%

* Density in units of grams per cubic centimeter

If we can be of further assistance, please contact us.

Very truly yours,

EARTH ENGINEERING & SCIENCES, INC.

Paul A. D'Amato

Paul A. D'Amato, P.E.
Vice President

PAD/ew/proj.doc/S&W

Earth Engineering & Sciences, Inc.	SUBJECT Maine Yankee Concrete Cores - Bulk Density and Porosity	SHEET NO. 1 OF 1 JOB NO. 99-160 BY PAD DATE 6/8/99 CHK'D DATE REV. DATE

Reference: ASTM C 642

Core # A9900/01FL2 (5.8" Dia)

Oven Dry Weight = 2956.8 gm
Saturated Wt. After Immersion = 3125.0 gm
Saturated Wt. After Boiling = 3151.6 gm
Immersed Wt. = 1816.0 gm.

Bulk Density (dry) = 2956.8/(3151.6 - 1816.0) = 2.21

Bulk Density (After Immersion) = 3125.0/(3151.6-1816.0) = 2.34

Bulk Density (After Immersion & Boiling) = 3151.6/(3151.6-1816.0) = 2.36

Apparent Density = 2956.8/(2956.8 - 1816.0) = 2.59

Permeable Pore Space (voids) = (3151.6 - 2956.8)/(3151.6 - 1816.0)
= 0.146 = 14.6%

Core # A9900/01ME3 (2.7" Dia)

Oven Dry Weight = 1082.1 gm
Saturated Wt. After Immersion = 1139.3 gm
Saturated Wt. After Boiling = 1147.7 gm
Immersed Weight = 670.0 gm

Bulk Density (Dry) = 1082.1/(1147.7 - 670.0) = 2.27
Bulk Density After Immersion = 1139.3/(1147.7-670.0) = 2.38
Bulk Density After Immersion & Boiling = 1147.1/(1147.1-670.0) = 2.40

Apparent Density = 1082.1/(1082.1-670.0) = 2.63

Permeable Pore Space (voids) = (1147.7-1082.1)/(1147.7 - 670.0)
= 0.137 = 13.7%

Designation: C 642 – 97

Standard Test Method for Density, Absorption, and Voids in Hardened Concrete[1]

This standard is issued under the fixed designation C 642; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This test method covers the determinations of denisty, percent absorption, and percent voids in hardened concrete.

1.2 The text of this test method references notes and footnotes which provide explanatory information. These notes and footnotes (excluding those in tables and figures) shall not be considered as requirements of this standard.

2. Significance and Use

2.1 This test method is useful in developing the data required for conversions between mass and volume for concrete. It can be used to determine conformance with specifications for concrete and to show differences from place to place within a mass of concrete.

3. Apparatus

3.1 *Balance*, sensitive to 0.025 % of the mass of the specimen.

3.2 *Container*, suitable for immersing the specimen and suitable wire for suspending the specimen in water.

4. Test Specimen

4.1 Whenever possible, the sample shall consist of several individual portions of concrete, each to be tested separately. The individual portions may be pieces of cylinders, cores, or beams of any desired shape or size, except that the volume of each portion shall be not less than 350 cm^3 (or for normal weight concrete, approximately 800 g); and each portion shall be free from observable cracks, fissures, or shattered edges.

5. Procedure

5.1 *Oven-Dry Mass*—Determine the mass of the portions, and dry in an oven at a temperature of 100 to 110°C for not less than 24 h. After removing each specimen from the oven, allow it to cool in dry air (preferably in a desiccator) to a temperature of 20 to 25°C and determine the mass. If the specimen was comparatively dry when its mass was first determined, and the second mass closely agrees with the first, consider it dry. If the specimen was wet when its mass was first determined, place it in the oven for a second drying treatment of 24 h and again determine the mass. If the third value checks the second, consider the specimen dry. In case of any doubt, redry the specimen for 24-h periods until check values of mass are obtained. If the difference between values obtained from two successive values of mass exceeds 0.5 % of the lesser value, return the specimens to the oven for an additional 24-h drying period, and repeat the procedure until the difference between any two successive values is less than 0.5 % of the lowest value obtained. Designate this last value *A*.

5.2 *Saturated Mass After Immersion*—Immerse the specimen, after final drying, cooling, and determination of mass, in water at approximately 21°C for not less than 48 h and until two successive values of mass of the surface-dried sample at intervals of 24 h show an increase in mass of less than 0.5 % of the larger value. Surface-dry the specimen by removing surface moisture with a towel, and determine the mass. Designate the final surface-dry mass after immersion *B*.

5.3 *Saturated Mass After Boiling*—Place the specimen, processed as described in 5.2, in a suitable receptacle, covered with tap water, and boil for 5 h. Allow it to cool by natural loss of heat for not less than 14 h to a final temperature of 20 to 25°C. Remove the surface moisture with a towel and determine the mass of the specimen. Designate the soaked, boiled, surface-dried mass *C*.

5.4 *Immersed Apparent Mass*—Suspend the specimen, after immersion and boiling, by a wire and determine the apparent mass in water. Designate this apparent mass *D*.

6. Calculation

6.1 By using the values for mass determined in accordance with the procedures described in Section 5, make the following calculations:

$$\text{Absorption after immersion, } \% = [(B - A)/A] \times 100 \quad (1)$$

$$\text{Absorption after immersion and boiling, } \% = [(C - A)/A] \times 100 \quad (2)$$

$$\text{Bulk density, dry} = [A/(C - D)] \cdot \rho = g_1 \quad (3)$$

$$\text{Bulk density after immersion} = [B/(C - D)] \cdot \rho \quad (4)$$

$$\text{Bulk density after immersion and boiling} = [C/(C - D)] \cdot \rho \quad (5)$$

$$\text{Apparent density} = [A/(A - D)] \cdot \rho = g_2 \quad (6)$$

$$\text{Volume of permeable pore space (voids), } \% = (g_2 - g_1)/g_2 \times 100 \text{ or } (C - A)/(C - D) \times 100 \quad (7)$$

where:

A = mass of oven-dried sample in air, g
B = mass of surface-dry sample in air after immersion, g
C = mass of surface-dry sample in air after immersion and boiling, g
D = apparent mass of sample in water after immersion and boiling, g
g_1 = bulk density, dry, Mg/m^3 and
g_2 = apparent density, Mg/m^3
ρ = density of water = 1 Mg/m^3 = 1 g/cm^3.

[1] This test method is under the jurisdiction of ASTM Committee C-9 on Concrete and Concrete Aggregates and is the direct responsibility of Subcommittee C09.69 on Miscellaneous Tests.

Current edition approved Jan. 10, 1997. Published March 1997. Originally published as C 642 – 69 T. Last previous edition C 642 – 90.

7. Exampl
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7. Example

7.1 Assume a sample having the following characteristics:

7.1.1 Mass of the solid part of the specimen = 1000 g.

7.1.2 Total volume of specimen (including solids, "permeable" voids, and "impermeable" voids) = 600 cm^3.

7.1.3 Absolute density of solid part of specimen = 2.0 Mg/m^3.

7.1.4 Void space in specimen contains initially only air (no water).

7.2 Then, it follows that there are 500 cm^3 of solids and 100 cm^3 of voids making up the specimen, and the void content is $1/6$ = 16.67 %.

7.3 Assume that on immersion 90 mL of water is absorbed.

7.4 Assume that after immersion and boiling 95 mL of water is absorbed.

7.5 Based on the assumptions given in 7.1 to 7.4 above, the data that would be developed from the procedures given in Section 5 would be as follows:

7.5.1 Oven-dry mass, A = 1000 g.

7.5.2 Mass in air after immersion, B = 1090 g.

7.5.3 Mass in air after immersion and boiling, C = 1095 g.

7.5.4 Apparent mass in water after immersion and boiling, D = 495 g.

NOTE 1—Since loss of mass in water is equal to mass of displaced water, and volume of specimen = 600 cm^3, mass of specimen in water after immersion and boiling is 1095 − 600 = 495 g.

7.6 By using the data given above to perform the calculations described in Section 6, the following results will be obtained (*Note 2*):

Absorption after immersion, % $= [(B - A)/A] \times 100 = [(1090 - 1000)/1000] \times 100 = 9.0$

Absorption after immersion and boiling, % $= [(C - A)/A] \times 100 = [(1095 - 1000)/1000] \times 100 = 9.5$

Bulk density, dry $= [A/(C - D)] \cdot \rho = [1000/(1095 - 495)] \times 1 = 1.67\ Mg/m^3 = g_1$

Bulk density after immersion $= [B/(C - D)] \cdot \rho = [1090/(1095 - 495)] \times 1 = 1.82$

Bulk density after immersion and boiling $= [C/(C - D)]\ \rho = [1095/(1095 - 495)] \times 1 = 1.83\ Mg/m^3$

Apparent density $= [A/(A - D)]\ \rho = [1000/(1000 - 495)] \times 1 = 1.98\ Mg/m^3 = g_2$

Volume of permeable voids, %
$= [(g_2 - g_1)/g_2] \times 100 = [(1.98 - 1.67)/1.98] \times 100$
$= 15.8$, or $[(C - A)/(C - D)] \times 100$
$= [(1095 - 1000)/(1095 - 495)] \times 100 = 15.7$

NOTE 2—This test method does not involve a determination of absolute density. Hence, such pore space as may be present in the specimen that is not emptied during the specified drying or is not filled with water during the specified immersion and boiling or both is considered "impermeable" and is not differentiated from the solid portion of the specimen for the calculations, especially those for percent voids. In the example discussed it was assumed that the absolute density of the solid portion of the specimen was 2.0 Mg/m^3, the total void space was 16.67 %, and the impermeable void space was 5 cm^3. The operations, if performed, and the calculations, if performed as described, have the effect of assuming that there are 95 cm^3 of pore space and 505 cm^3 of solids, and indicate that the solid material, therefore, has an apparent density of 1.98 rather than the absolute density of 2.00 Mg/m^3 and the specimen has a percentage of voids of 15.8 rather than 16.67.

Depending on the pore size distribution and the pore entry radii of the concrete and on the purposes for which the test results are desired, the procedures of this test method may be adequate, or they may be insufficiently rigorous. In the event that it is desired to fill more of the pores than will be filled by immersion and boiling, various techniques involving the use of vacuum treatment or increased pressures may be used. If a rigorous measure of total pore space is desired, this can only be obtained by determining absolute density by first reducing the sample to discrete particles, each of which is sufficiently small so that no impermeable pore space can exist within any of the particles. If the absolute density were determined and designated g_3, then:

Total void volume, % $= (g_3 - g_1)/g_3 \times 100 = (2.00 - 1.67)/2.00 \times 100 = 16.5$

8. Precision and Bias

8.1 *Precision*—At present there are insufficient data available to justify attempting to develop a precision statement for this test method.

8.2 *Bias*—Bias for this test method cannot be determined since there is no reference standard available for comparison.

9. Keywords

9.1 absorption; concrete-hardened; density; voids

Attachment 6-6
Deleted

Attachment 6-7
Embedded Piping List

Remaining Embedded Piping

No.	Location	Pen No.	Length (ft)	Embedded Diameter (in.)	Surface Area (m^2)
	Primary Auxiliary Building To Containment				
1	Steam Generator Blowdown	45	5.0	16.0	1.945
2	Reactor Coolant Letdown	32	5.0	16.0	1.945
3	Primary Component Cooling Reactor Coolant Pump	6	5.0	6.0	0.729
4	Primary Drain Transfer Pump Discharge	39	5.0	3.0	0.365
5	Spare	41	5.0	2.0	0.243
6	Air Recirculation Cooling Water (out)	21	5.0	6.0	0.729
7	Air Recirculation Cooling Water (out)	20	5.0	6.0	0.729
8	Air Recirculation Cooling Water (out)	19	5.0	6.0	0.729
9	Air Recirculation Cooling Water (out)	18	5.0	6.0	0.729
10	Air Recirculation Cooling Water (out)	17	5.0	6.0	0.729
11	Air Recirculation Cooling Water (in)	10	5.0	6.0	0.729
12	Air Recirculation Cooling Water (in)	11	5.0	6.0	0.729
13	Air Recirculation Cooling Water (in)	12	5.0	6.0	0.729
14	Air Recirculation Cooling Water (in)	13	5.0	6.0	0.729
15	Air Recirculation Cooling Water (in)	14	5.0	6.0	0.729
16	Steam Generator Blowdown Lines	47	5.0	16.0	1.945
17	Auxiliary Steam	46	5.0	16.0	1.945
18	Spare	82	5.0	4.0	0.486
19	Air Recirculation Cooling Water (out)	16	5.0	6.0	0.729
20	Air Recirculation Cooling Water (out)	15	5.0	6.0	0.729
21	Containment Isolation & Safeguards Pressure Switch Header	63	5.0	8.0	0.973
22	Pressurizer Steam Interface and Pressurizer Safety Valve Loop	62	5.0	8.0	0.973
23	Containment Leak Detection	61	5.0	8.0	0.973
24	Seal Water From Reactor Coolant Pump	23	5.0	3.0	0.365
25	Spare	25	5.0	4.0	0.486
26	Primary Component Cooling Water From CRDM Coolers	86	5.0	4.0	0.486
27	Primary Vent Header	24	5.0	2.0	0.243
28	Aerated Vent Header	42	5.0	2.0	0.243
29	Steam Generator Auxiliary Feedwater	75	5.0	16.0	1.945

Remaining Embedded Piping

No.	Location	Pen No.	Length (ft)	Embedded Diameter (in.)	Surface Area (m^2)
30	Charging	22	5.0	3.0	0.365
31	High Pressure Safety Injection	72	5.0	4.0	0.486
32	Spare	69	5.0	8.0	0.973
33	Spare	40	5.0	3.0	0.365
34	Spare, Chemical Cleaning During Construction	74	5.0	8.0	0.973
35	Spare	2	5.0	1.5	0.182
36	Steam Generator Auxiliary Feedwater	76	5.0	8.0	0.973
37	Spare	77	5.0	8.0	0.973
38	Primary Component Cooling From High Pressure/Low Pressure Shield Tank Coolers	5	5.0	6.0	0.729
39	Primary Component Cooling Water From CRDM Coolers	89	5.0	4.0	0.486
40	Spare	1	5.0	1.5	0.182
41	Spare	88	5.0	4.0	0.486
42	Spare	87	5.0	4.0	0.486
43	Injection Seal Water to Reactor Cooling Pump	26	5.0	1.5	0.182
44	Primary Component Cooling From High Pressure/Low Pressure Drain Shield Coolers a& Neutron Tank Shield Coolers	4	5.0	6.0	0.729
45	Primary Component Cooling to Reactor Cooling Pumps	3	5.0	6.0	0.729
46	Demineralizer Water to Quench Tank	37	5.0	2.0	0.243
47	Nitrogen To Quench and Safety Injection Tanks	44	5.0	1.0	0.122
48	Instrument Air	49	5.0	1.5	0.182
49	Spare	90	5.0	6.0	0.729
50	Spare	80	5.0	6.0	0.729
51	Service Air	48	5.0	2.0	0.243
52	High Pressure Safety Injection	71	5.0	4.0	0.486
53	High Pressure Safety Injection	73	5.0	4.0	0.486
54	Spare	43	5.0	0.75	0.091
55	Post Accident Purge	84	5.0	2.0	0.243
56	Refueling Cavity Purification (out)	79	5.0	6.0	0.729
57	Spare	91	5.0	6.0	0.729
58	Spare	83	5.0	4.0	0.486
59	Spare	85	5.0	3.0	0.365

Remaining Embedded Piping

No.	Location	Pen No.	Length (ft)	Embedded Diameter (in.)	Surface Area (m^2)
60	Spare	38	5.0	4.0	0.486
61	Sump Pump Discharge	92	5.0	2.0	0.243
62	Air Monitor Sample	59	5.0	1.0	0.122
63	Primary Component Cooling to and from Penetration Coolers	81	5.0	8.0	0.973
64	Spare	78	5.0	8.0	0.973
65	Neutron Shield Tank Fill	35	5.0	1.5	0.182
66	Air Monitor Sample	60	5.0	1.0	0.122
67	Spare	7	5.0	3.0	0.365
68	Spare	8	5.0	3.0	0.365
69	Injection Seal Water to Reactor Coolant Pump	28	5.0	1.5	0.182
70	Injection Seal Water to Reactor Coolant Pump	27	5.0	1.5	0.182
71	Reactor Coolant Loop Fill	36	5.0	2.0	0.243
72	Fire Water Supply (North Wall)	154	2.0	8.0	0.389
	South Wall Primary Auxiliary Building To Yard				
73	Ric-Wil	23	2.0	20.5	0.997
74	Primary Water	276	2.0	2.0	0.097
75	Ric-Wil	24	2.0	22.0	1.070
76	Secondary Component Cooling	282	2.0	1.5	0.073
77	Demineralized Water to Storage Tank	5	2.0	4.0	0.195
78	Secondary Component Cooling	4	2.0	2.0	0.097
79	Primary Component Cooling	3	2.0	16.0	0.778
80	Primary Component Cooling	41	2.0	16.0	0.778
81	Borated Water to Sump	26	2.0	4.0	0.195
	Containment Spray Pump Building				
82	Drain Line from Main Steam Valve House and Personnel Hatch (East Wall)		2.25	6.0	0.328
83	Ric-Wil (East Wall)		2.25	16.0	0.875
84	Ric-Wil (East Wall)		2.25	16.0	0.875
85	Ric-Wil (South East Wall)		3.0	19.0	1.386
86	Ric-Wil (South East Wall)		3.0	12.0	0.875
87	Ric-Wil (South East Wall)		3.0	14.0	1.021

Remaining Embedded Piping

No.	Location	Pen No.	Length (ft)	Embedded Diameter (in.)	Surface Area (m^2)
88	Secondary Component Cooling Supply to E-3B (South East Wall)		3.0	16.0	1.167
89	Secondary Component Cooling Return to E-3B (South East Wall)		3.0	16.0	1.167
90	Purification to Refueling Water Storage Tank (South West Wall)		3.0	6.0	0.438
91	Ric-Wil (South West Wall)		3.0	19.0	1.386
92	Ric-Wil (South West Wall)		3.0	22.0	1.605
93	Ric-Wil (South West Wall)		3.0	22.0	1.605
94	Primary Component Cooling Supply to E-3A (South West Wall)		3.0	14.0	1.021
95	Primary Component Cooling Return From E-3A (South West Wall)		3.0	14.0	1.021
	Containment Spray Pump Building To Containment				
96	Residual Heat Remover (out)	9	5.0	30.0	3.647
97	Low Pressure Steam Injection and Residual Heat Remover (in)	29	5.0	30.0	3.647
98	Low Pressure Steam Injection and Residual Heat Remover (in)	30	5.0	30.0	3.647
99	Low Pressure Steam Injection and Residual Heat Remover (in)	31	5.0	30.0	3.647
100	Low Pressure Steam Injection and Containment Spray Pump Suction	33	34.0	16.0	13.228
101	Low Pressure Steam Injection and Containment Spray Pump Suction	34	34.0	16.0	13.228
102	Containment Spray Supply[1]	50	5.0	24.0	2.918
103	Containment Spray Supply (See Note 1 related to Item No. 102.)	51	5.0	24.0	2.918
104	Safety Injection Test and Safety Injection Tank Liquid Sample	68	5.0	2.0	0.243
105	Spare	70	5.0	10.0	1.216
106	Containment Spray Pump Casing Vent	93	5.0	2.0	0.243
	Fuel Building To Containment				
107	Fuel Transfer Tube	52	11.0	40.0	10.669
	Miscellaneous				
108	Liner Leak Detection system		4 x 6.0	4 ea. @ 1.0	0.584
109	Containment Foundation Drain		122.0	6.0	17.799
110	Containment Foundation Drain		256.0	2.0	12.450
	Totals:		940.75		154.2

[1] The subject 24" containment spray supply lines have been removed; however, the associated activity inventory was included in the dose assessment. See also, Section 6.7.2.

Attachment 6-8
Deep Soil Microshield Output

Microshield
Deep Soil Direct Dose
Unitized Values

This attachment provides the Microshield outputs for direct dose factors for deep soil. No output is provided for H-3 or Ni-63 because they have no contribution to direct dose. The area size is 10,000 m^2 by 2.85 m deep. Soil density is 1.6 g/cm^3. The dose point is 1 meter above the soil surface.

Then direct docs factors are determined by multiplying the fraction of the time spent indoors (0.6571) by the external gamma shielding factor (0.5512) then adding the fraction of time spent outdoors (0.1101). The resulting number is multiplied by 24 hours per day for 365 days per year.

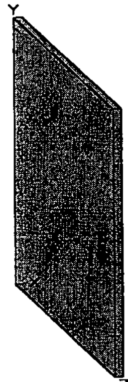
Nuclide	Direct Dose Factor (mrem/y per pCi/g)
H-3	0.00E+00
Ni-63	0.00E+00
Co-60	2.4E+00
Cs-137	4.00E-01

MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power

Page : 1
DOS File: Case1
Run Date: September 24, 2002
Run Time: 2:30:37 PM
Duration: 00:00:02

File Ref:
Date:
By:
Checked:

Case Title: Deep Soil Cs-137
Description: Direct Dose Rate for Unit Activity
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	285.0 cm	9 ft 4.2 in
Width	1.0e+4 cm	328 ft 1.0 in
Height	1.0e+4 cm	328 ft 1.0 in

Dose Points

	X	Y	Z
# 1	400 cm	5000 cm	5000 cm
	13 ft 1.5 in	164 ft 0.5 in	164 ft 0.5 in

Shields

Shield Name	Dimension	Material	Density
Source	2.85e+10 cm³	SiO2	1.6
Shield 1	15.0 cm	SiO2	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	4.5600e-002	1.6872e+009	1.6000e-006	5.9200e-002
Cs-137	4.5600e-002	1.6872e+009	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	3.493e+07	4.855e-18	1.532e-17	4.044e-20	1.276e-19
0.0322	6.445e+07	2.210e-17	7.140e-17	1.778e-19	5.746e-19
0.0364	2.345e+07	1.818e-14	7.670e-14	1.033e-16	4.358e-16
0.6616	1.518e+09	8.579e-03	6.022e-02	1.663e-05	1.167e-04
TOTALS:	1.641e+09	8.579e-03	6.022e-02	1.663e-05	1.167e-04

Attachment 6-8
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MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power
Conversion of calculated exposure in air to dose
FILE: Case1
Case Title: Deep Soil Cs-137
This case was run on Tuesday, September 24, 2002 at 2:40:54 PM
Dose Point # 1 - (400,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.297e-002	9.101e-002
Photon Energy Fluence Rate	MeV/cm²/sec	8.579e-003	6.022e-002
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	1.663e-005	1.167e-004
Absorbed Dose Rate in Air	mGy/hr	1.452e-007	1.019e-006
"	mrad/hr	1.452e-005	1.019e-004
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.719e-007	1.207e-006
o Opposed	"	1.376e-007	9.661e-007
o Rotational	"	1.376e-007	9.661e-007
o Isotropic	"	1.217e-007	8.543e-007
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.827e-007	1.282e-006
o Opposed	"	1.735e-007	1.218e-006
o Rotational	"	1.735e-007	1.218e-006
o Isotropic	"	1.301e-007	9.132e-007
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	1.520e-007	1.067e-006
o Posterior/Anterior	"	1.342e-007	9.418e-007
o Lateral	"	9.950e-008	6.983e-007
o Rotational	"	1.199e-007	8.416e-007
o Isotropic	"	1.021e-007	7.165e-007

MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power
Conversion of calculated exposure in air to dose
FILE: Case1
Case Title: Deep Soil Co-60
This case was run on Tuesday, September 24, 2002 at 2:35:00 PM
Dose Point # 1 - (400,5000,5000) cm

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	7.381e-002	3.092e-001
Photon Energy Fluence Rate	MeV/cm²/sec	9.300e-002	3.884e-001
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	1.634e-004	6.829e-004
Absorbed Dose Rate in Air	mGy/hr	1.427e-006	5.962e-006
"	mrad/hr	1.427e-004	5.962e-004
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.620e-006	6.770e-006
o Opposed	"	1.394e-006	5.826e-006
o Rotational	"	1.394e-006	5.826e-006
o Isotropic	"	1.245e-006	5.203e-006
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.725e-006	7.209e-006
o Opposed	"	1.660e-006	6.939e-006
o Rotational	"	1.660e-006	6.939e-006
o Isotropic	"	1.320e-006	5.515e-006
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	1.459e-006	6.096e-006
o Posterior/Anterior	"	1.343e-006	5.613e-006
o Lateral	"	1.068e-006	4.460e-006
o Rotational	"	1.209e-006	5.053e-006
o Isotropic	"	1.070e-006	4.470e-006

Attachment 6-9
Deep Soil RESRAD Output

Table of Contents

Part I: Mixture Sums and Single Radionuclide Guidelines

Dose Conversion Factor (and Related) Parameter Summary
File: FGR 13 Morbidity

Menu	Parameter	Current Value	Default	Parameter Name
B-1	Dose conversion factors for inhalation, mrem/pCi:			
B-1	Co-60	2.190E-04	2.190E-04	DCF2(1)
D-1	Dose conversion factors for ingestion, mrem/pCi:			
D-1	Co-60	2.690E-05	2.690E-05	DCF3(1)
D-34	Food transfer factors:			
D-34	Co-60 , plant/soil concentration ratio, dimensionless	8.000E-02	8.000E-02	RTF(1,1)
D-34	Co-60 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	2.000E-02	2.000E-02	RTF(1,2)
D-34	Co-60 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	2.000E-03	2.000E-03	RTF(1,3)
D-5	Bioaccumulation factors, fresh water, L/kg:			
D-5	Co-60 , fish	3.000E+02	3.000E+02	BIOFAC(1,1)
D-5	Co-60 , crustacea and mollusks	2.000E+02	2.000E+02	BIOFAC(1,2)

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	1.000E+04	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	2.850E+00	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	1.000E+01	2.500E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.900E-01	1.000E+00	---	T(2)
R011	Times for calculations (yr)	2.000E-01	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+00	1.000E+01	---	T(4)
R011	Times for calculations (yr)	7.100E+00	3.000E+01	---	T(5)
R011	Times for calculations (yr)	4.270E+01	1.000E+02	---	T(6)
R011	Times for calculations (yr)	1.300E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.340E+02	1.000E+03	---	T(8)
R011	Times for calculations (yr)	1.500E+02	0.000E+00	---	T(9)
R011	Times for calculations (yr)	1.000E+03	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Co-60	1.000E+00	0.000E+00	---	S1(1)
R012	Concentration in groundwater (pCi/L): Co-60	not used	0.000E+00	---	W1(1)
R013	Cover depth (m)	1.500E-01	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	3.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	4.050E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	1.000E+06	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	1.000E-03	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	3.000E-01	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	1.000E-02	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	2.000E-01	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	2.000E-02	2.000E-02	---	HGWT
R014	Saturated zone b parameter	4.050E+00	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	1.000E-03	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	1.000E+01	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW
R015	Number of unsaturated zone strata	1	1	---	NS

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	1.000E+04	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	2.850E+00	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	1.000E+01	2.500E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.900E-01	1.000E+00	---	T(2)
R011	Times for calculations (yr)	2.000E-01	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+00	1.000E+01	---	T(4)
R011	Times for calculations (yr)	7.100E+00	3.000E+01	---	T(5)
R011	Times for calculations (yr)	4.270E+01	1.000E+02	---	T(6)
R011	Times for calculations (yr)	1.300E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.340E+02	1.000E+03	---	T(8)
R011	Times for calculations (yr)	1.500E+02	0.000E+00	---	T(9)
R011	Times for calculations (yr)	1.000E+03	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Co-60	1.000E+00	0.000E+00	---	S1(1)
R012	Concentration in groundwater (pCi/L): Co-60	not used	0.000E+00	---	W1(1)
R013	Cover depth (m)	1.500E-01	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	3.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	4.050E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	1.000E+06	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	1.000E-03	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	3.000E-01	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	1.000E-02	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	2.000E-01	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	2.000E-02	2.000E-02	---	HGWT
R014	Saturated zone b parameter	4.050E+00	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	1.000E-03	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	1.000E+01	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW
R015	Number of unsaturated zone strata	1	1	---	NS

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	1.000E+04	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	2.850E+00	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	1.000E+01	2.500E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.900E-01	1.000E+00	---	T(2)
R011	Times for calculations (yr)	2.000E-01	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+00	1.000E+01	---	T(4)
R011	Times for calculations (yr)	7.100E+00	3.000E+01	---	T(5)
R011	Times for calculations (yr)	4.270E+01	1.000E+02	---	T(6)
R011	Times for calculations (yr)	1.300E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.344E+02	1.000E+03	---	T(8)
R011	Times for calculations (yr)	1.500E+02	0.000E+00	---	T(9)
R011	Times for calculations (yr)	1.000E+03	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Ni-63	1.000E+00	0.000E+00	---	S1(1)
R012	Concentration in groundwater (pCi/L): Ni-63	not used	0.000E+00	---	W1(1)
R013	Cover depth (m)	1.500E-01	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	3.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	4.050E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	1.000E+06	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	1.000E-03	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	3.000E-01	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	1.000E-02	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	2.000E-01	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	2.000E-02	2.000E-02	---	HGWT
R014	Saturated zone b parameter	4.050E+00	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	1.000E-03	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	1.000E+01	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW
R015	Number of unsaturated zone strata	1	1	---	NS

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	1.000E+04	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	2.850E+00	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	1.000E+01	2.500E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.900E-01	1.000E+00	---	T(2)
R011	Times for calculations (yr)	2.000E-01	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+00	1.000E+01	---	T(4)
R011	Times for calculations (yr)	7.100E+00	3.000E+01	---	T(5)
R011	Times for calculations (yr)	4.270E+01	1.000E+02	---	T(6)
R011	Times for calculations (yr)	1.340E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.500E+02	1.000E+03	---	T(8)
R011	Times for calculations (yr)	1.000E+03	0.000E+00	---	T(9)
R011	Times for calculations (yr)	not used	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): H-3	1.000E+00	0.000E+00	---	S1(1)
R012	Concentration in groundwater (pCi/L): H-3	not used	0.000E+00	---	W1(1)
R013	Cover depth (m)	1.500E-01	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	3.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	4.050E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	1.000E+06	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	1.000E-03	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	3.000E-01	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	1.000E-02	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	2.000E-01	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	2.000E-02	2.000E-02	---	HGWT
R014	Saturated zone b parameter	4.050E+00	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	1.000E-03	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	1.000E+01	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW
R015	Number of unsaturated zone strata	1	1	---	NS

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Unsat. zone 1, thickness (m)	0.000E+00	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	1.600E+00	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	3.000E-01	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	2.000E-01	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	4.050E+00	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	1.000E+03	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Cs-137				
R016	Contaminated zone (cm**3/g)	1.200E+03	1.000E+03	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	1.200E+03	1.000E+03	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	1.200E+03	1.000E+03	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	9.136E-05	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R017	Inhalation rate (m**3/yr)	not used	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	not used	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	not used	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	not used	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	not used	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	not used	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	not used	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Unsat. zone 1, thickness (m)	0.000E+00	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	1.600E+00	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	3.000E-01	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	2.000E-01	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	4.050E+00	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	1.000E+03	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Co-60				
R016	Contaminated zone (cm**3/g)	3.350E+02	1.000E+03	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	3.350E+02	1.000E+03	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	3.350E+02	1.000E+03	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	3.272E-04	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R017	Inhalation rate (m**3/yr)	not used	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	not used	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	not used	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	not used	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	not used	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	not used	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	not used	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Unsat. zone 1, thickness (m)	0.000E+00	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	1.600E+00	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	3.000E-01	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	2.000E-01	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	4.050E+00	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	1.000E+03	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Ni-63				
R016	Contaminated zone (cm**3/g)	2.740E+02	1.000E+03	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	2.740E+02	1.000E+03	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	2.740E+02	1.000E+03	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	4.000E-04	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R017	Inhalation rate (m**3/yr)	not used	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	not used	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	not used	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	not used	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	not used	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	not used	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	not used	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Unsat. zone 1, thickness (m)	0.000E+00	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	1.600E+00	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	3.000E-01	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	2.000E-01	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	4.050E+00	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	1.000E+03	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for H-3				
R016	Contaminated zone (cm**3/g)	0.000E+00	0.000E+00	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	0.000E+00	0.000E+00	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	0.000E+00	0.000E+00	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	8.506E-01	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R017	Inhalation rate (m**3/yr)	not used	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	not used	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	not used	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	not used	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	not used	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	not used	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	not used	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R018	Fruits, vegetables and grain consumption (kg/yr)	not used	1.600E+02	---	DIET(1)
R018	Leafy vegetable consumption (kg/yr)	not used	1.400E+01	---	DIET(2)
R018	Milk consumption (L/yr)	not used	9.200E+01	---	DIET(3)
R018	Meat and poultry consumption (kg/yr)	not used	6.300E+01	---	DIET(4)
R018	Fish consumption (kg/yr)	not used	5.400E+00	---	DIET(5)
R018	Other seafood consumption (kg/yr)	not used	9.000E-01	---	DIET(6)
R018	Soil ingestion rate (g/yr)	not used	3.650E+01	---	SOIL
R018	Drinking water intake (L/yr)	4.785E+02	5.100E+02	---	DWI
R018	Contamination fraction of drinking water	1.000E+00	1.000E+00	---	FDW
R018	Contamination fraction of household water	not used	1.000E+00	---	FHHW
R018	Contamination fraction of livestock water	not used	1.000E+00	---	FLW
R018	Contamination fraction of irrigation water	not used	1.000E+00	---	FIRW
R018	Contamination fraction of aquatic food	not used	5.000E-01	---	FR9
R018	Contamination fraction of plant food	not used	-1	---	FPLANT
R018	Contamination fraction of meat	not used	-1	---	FMEAT
R018	Contamination fraction of milk	not used	-1	---	FMILK
R019	Livestock fodder intake for meat (kg/day)	not used	6.800E+01	---	LFI5
R019	Livestock fodder intake for milk (kg/day)	not used	5.500E+01	---	LFI6
R019	Livestock water intake for meat (L/day)	not used	5.000E+01	---	LWI5
R019	Livestock water intake for milk (L/day)	not used	1.600E+02	---	LWI6
R019	Livestock soil intake (kg/day)	not used	5.000E-01	---	LSI
R019	Mass loading for foliar deposition (g/m**3)	not used	1.000E-04	---	MLFD
R019	Depth of soil mixing layer (m)	not used	1.500E-01	---	DM
R019	Depth of roots (m)	not used	9.000E-01	---	DROOT
R019	Drinking water fraction from ground water	1.000E+00	1.000E+00	---	FGWDW
R019	Household water fraction from ground water	not used	1.000E+00	---	FGWHH
R019	Livestock water fraction from ground water	not used	1.000E+00	---	FGWLW
R019	Irrigation fraction from ground water	not used	1.000E+00	---	FGWIR
R19B	Wet weight crop yield for Non-Leafy (kg/m**2)	not used	7.000E-01	---	YV(1)
R19B	Wet weight crop yield for Leafy (kg/m**2)	not used	1.500E+00	---	YV(2)
R19B	Wet weight crop yield for Fodder (kg/m**2)	not used	1.100E+00	---	YV(3)
R19B	Growing Season for Non-Leafy (years)	not used	1.700E-01	---	TE(1)
R19B	Growing Season for Leafy (years)	not used	2.500E-01	---	TE(2)
R19B	Growing Season for Fodder (years)	not used	8.000E-02	---	TE(3)
R19B	Translocation Factor for Non-Leafy	not used	1.000E-01	---	TIV(1)
R19B	Translocation Factor for Leafy	not used	1.000E+00	---	TIV(2)
R19B	Translocation Factor for Fodder	not used	1.000E+00	---	TIV(3)
R19B	Dry Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RDRY(1)
R19B	Dry Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RDRY(2)
R19B	Dry Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RDRY(3)
R19B	Wet Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RWET(1)
R19B	Wet Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RWET(2)
R19B	Wet Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RWET(3)
R19B	Weathering Removal Constant for Vegetation	not used	2.000E+01	---	WLAM
C14	C-12 concentration in water (g/cm**3)	not used	2.000E-05	---	C12WTR
C14	C-12 concentration in contaminated soil (g/g)	not used	3.000E-02	---	C12CZ
C14	Fraction of vegetation carbon from soil	not used	2.000E-02	---	CSOIL
C14	Fraction of vegetation carbon from air	not used	9.800E-01	---	CAIR

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
C14	C-14 evasion layer thickness in soil (m)	not used	3.000E-01	---	DMC
C14	C-14 evasion flux rate from soil (1/sec)	not used	7.000E-07	---	EVSN
C14	C-12 evasion flux rate from soil (1/sec)	not used	1.000E-10	---	REVSN
C14	Fraction of grain in beef cattle feed	not used	8.000E-01	---	AVFG4
C14	Fraction of grain in milk cow feed	not used	2.000E-01	---	AVFG5
C14	DCF correction factor for gaseous forms of C14	not used	8.894E+01	---	CO2F
STOR	Storage times of contaminated foodstuffs (days):				
STOR	Fruits, non-leafy vegetables, and grain	1.400E+01	1.400E+01	---	STOR_T(1)
STOR	Leafy vegetables	1.000E+00	1.000E+00	---	STOR_T(2)
STOR	Milk	1.000E+00	1.000E+00	---	STOR_T(3)
STOR	Meat and poultry	2.000E+01	2.000E+01	---	STOR_T(4)
STOR	Fish	7.000E+00	7.000E+00	---	STOR_T(5)
STOR	Crustacea and mollusks	7.000E+00	7.000E+00	---	STOR_T(6)
STOR	Well water	1.000E+00	1.000E+00	---	STOR_T(7)
STOR	Surface water	1.000E+00	1.000E+00	---	STOR_T(8)
STOR	Livestock fodder	4.500E+01	4.500E+01	---	STOR_T(9)
R021	Thickness of building foundation (m)	not used	1.500E-01	---	FLOOR1
R021	Bulk density of building foundation (g/cm**3)	not used	2.400E+00	---	DENSFL
R021	Total porosity of the cover material	not used	4.000E-01	---	TPCV
R021	Total porosity of the building foundation	not used	1.000E-01	---	TPFL
R021	Volumetric water content of the cover material	not used	5.000E-02	---	PH2OCV
R021	Volumetric water content of the foundation	not used	3.000E-02	---	PH2OFL
R021	Diffusion coefficient for radon gas (m/sec):				
R021	in cover material	not used	2.000E-06	---	DIFCV
R021	in foundation material	not used	3.000E-07	---	DIFFL
R021	in contaminated zone soil	not used	2.000E-06	---	DIFCZ
R021	Radon vertical dimension of mixing (m)	not used	2.000E+00	---	HMIX
R021	Average building air exchange rate (1/hr)	not used	5.000E-01	---	REXG
R021	Height of the building (room) (m)	not used	2.500E+00	---	HRM
R021	Building interior area factor	not used	0.000E+00	---	FAI
R021	Building depth below ground surface (m)	not used	-1.000E+00	---	DMFL
R021	Emanating power of Rn-222 gas	not used	2.500E-01	---	EMANA(1)
R021	Emanating power of Rn-220 gas	not used	1.500E-01	---	EMANA(2)
TITL	Number of graphical time points	32	---	---	NPTS
TITL	Maximum number of integration points for dose	17	---	---	LYMAX
TITL	Maximum number of integration points for risk	257	---	---	KYMAX

Summary of Pathway Selections

Pathway	User Selection
1 -- external gamma	suppressed
2 -- inhalation (w/o radon)	suppressed
3 -- plant ingestion	suppressed
4 -- meat ingestion	suppressed
5 -- milk ingestion	suppressed
6 -- aquatic foods	suppressed
7 -- drinking water	active
8 -- soil ingestion	suppressed
9 -- radon	suppressed
Find peak pathway doses	active

Contaminated Zone Dimensions

Area: 10000.00 square meters
Thickness: 2.85 meters
Cover Depth: 0.15 meters

Initial Soil Concentrations, pCi/g

Cs-137 1.000E+00

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 1.000E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.340E+02	1.500E+02	1.000E+03
TDOSE(t):	6.647E-06	9.153E-06	9.285E-06	1.961E-05	8.650E-05	2.158E-04	8.644E-05	8.121E-05	6.275E-05	1.070E-12
M(t):	6.647E-07	9.153E-07	9.285E-07	1.961E-06	8.650E-06	2.158E-05	8.644E-06	8.121E-06	6.275E-06	1.070E-13

Maximum TDOSE(t): 2.158E-04 mrem/yr at t = 42.70 ± 0.09 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 4.270E+01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Cs-137	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 4.270E+01 years

Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Cs-137	2.158E-04	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.158E-04	1.0000
Total	2.158E-04	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.158E-04	1.0000

*Sum of all water independent and dependent pathways.

RESRAD, Version 6.1 T½ Limit = 0.5 year 09/24/2002 12:27 Page 8
Summary : RESRAD Default Parameters File: deepsoil061802Cob405.RAD

Contaminated Zone Dimensions

Area: 10000.00 square meters
Thickness: 2.85 meters
Cover Depth: 0.15 meters

Initial Soil Concentrations, pCi/g

Co-60 1.000E+00

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 1.000E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.340E+02	1.500E+02	1.000E+03
TDOSE(t):	4.721E-05	6.401E-05	6.487E-05	1.264E-04	2.891E-04	1.516E-05	4.670E-10	2.843E-10	3.870E-11	0.000E+00
M(t):	4.721E-06	6.401E-06	6.487E-06	1.264E-05	2.891E-05	1.516E-06	4.670E-11	2.843E-11	3.870E-12	0.000E+00

Maximum TDOSE(t): 2.891E-04 mrem/yr at t = 7.11 ± 0.01 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 7.110E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	frac
Co-60	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.00
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.00

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 7.110E+00 years

Water Dependent Pathways

Radio-Nuclide Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	frac
Co-60	2.891E-04	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.891E-04	1.00
Total	2.891E-04	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	2.891E-04	1.00

*Sum of all water independent and dependent pathways.

Contaminated Zone Dimensions

Area: 10000.00 square meters
Thickness: 2.85 meters
Cover Depth: 0.15 meters

Initial Soil Concentrations, pCi/g

Ni-63 1.000E+00

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 1.000E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.344E+02	1.500E+02	1.000E+03
TDOSE(t):	1.655E-06	2.283E-06	2.317E-06	4.948E-06	2.400E-05	1.048E-04	1.658E-04	1.659E-04	1.649E-04	4.036E-07
M(t):	1.655E-07	2.283E-07	2.317E-07	4.948E-07	2.400E-06	1.048E-05	1.658E-05	1.659E-05	1.649E-05	4.036E-08

Maximum TDOSE(t): 1.659E-04 mrem/yr at t = 134.4 ± 0.3 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.344E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	frac
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.00
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.00

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.344E+02 years

Water Dependent Pathways

Radio-Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	frac
Ni-63	1.659E-04	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.659E-04	1.00
Total	1.659E-04	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.659E-04	1.00

*Sum of all water independent and dependent pathways.

Contaminated Zone Dimensions	
Area:	10000.00 square meters
Thickness:	2.85 meters
Cover Depth:	0.15 meters

Initial Soil Concentrations, pCi/g	
H-3	1.000E+00

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 1.000E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.340E+02	1.500E+02	1.000E+03
TDOSE(t):	1.241E-01	1.158E-01	1.146E-01	3.942E-02	1.005E-05	1.577E-28	0.000E+00	0.000E+00	0.000E+00
M(t):	1.241E-02	1.158E-02	1.146E-02	3.942E-03	1.005E-06	1.577E-29	0.000E+00	0.000E+00	0.000E+00

Maximum TDOSE(t): 1.241E-01 mrem/yr at t = 0.000E+00 years

Concentration of radionuclides in environmental media
at t = 4.270E+01 years

Radio-Nuclide	Contaminated Zone	Surface Soil*	Air Particulate	Well Water	Surface Water
	pCi/g	pCi/g	pCi/m**3	pCi/L	pCi/L
Cs-137	3.714E-01	1.057E-01	1.790E-06	9.020E-03	9.020E-05

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

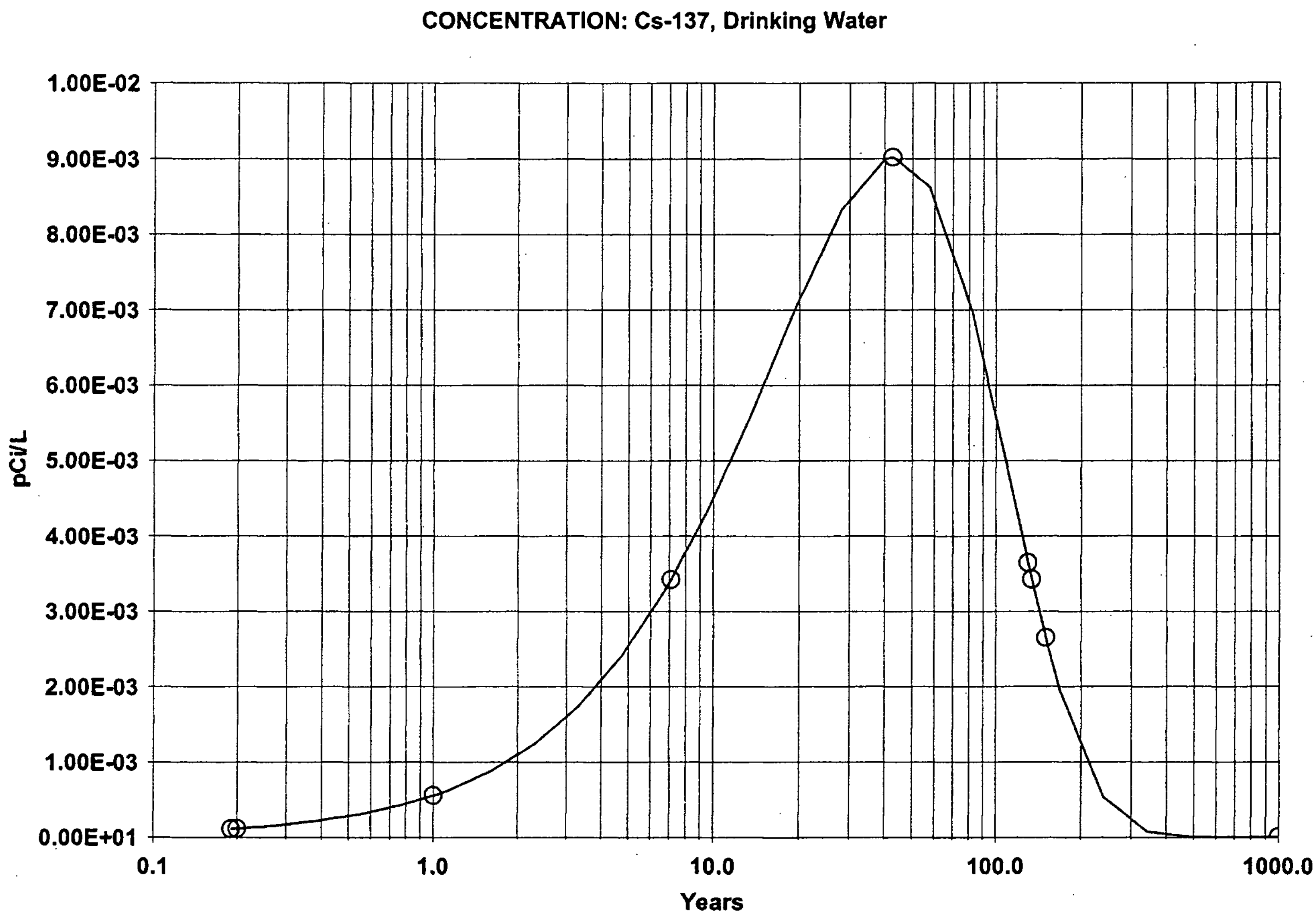
Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 4.270E+01 years*

Radio-Nuclide	Drinking Water	Nonleafy Vegetable	Leafy Vegetable	Fodder Meat	Fodder Milk	Meat	Milk	Fish	Crustacea
	pCi/L	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/L	pCi/kg	pCi/kg
Cs-137	9.020E-03	1.309E+01	1.310E+01	1.312E+01	1.310E+01	2.832E+01	6.198E+00	1.803E-01	9.016E-03

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.



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Concentration of radionuclides in environmental media
at t = 7.100E+00 years

Radio-Nuclide	Contaminated Zone	Surface Soil*	Air Particulate	Well Water	Surface Water
	pCi/g	pCi/g	pCi/m**3	pCi/L	pCi/L
Co-60	3.922E-01	1.856E-02	3.143E-07	2.244E-02	2.244E-04

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

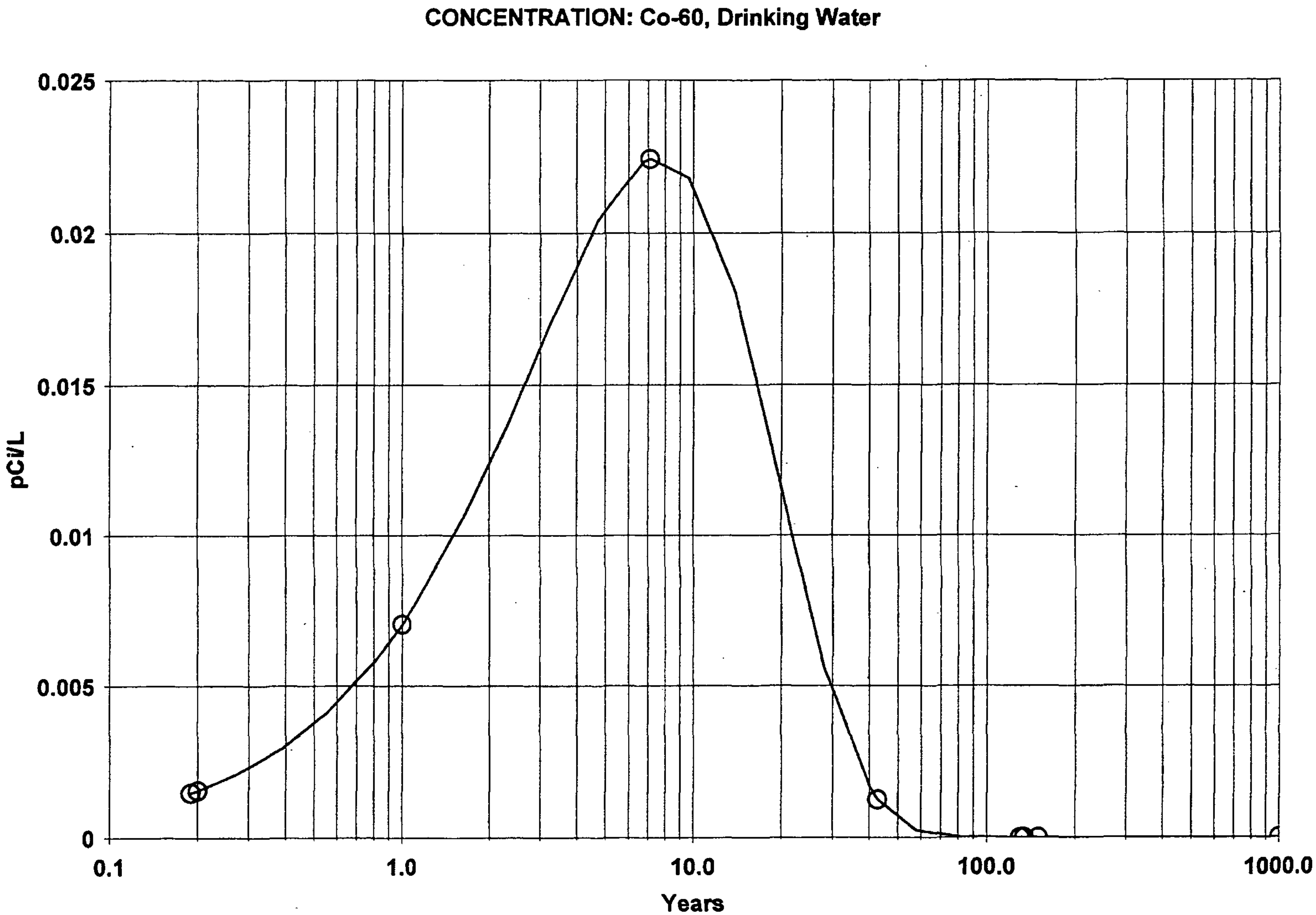
Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 7.100E+00 years*

Radio-Nuclide	Drinking Water	Nonleafy Vegetable	Leafy Vegetable	Fodder Meat	Fodder Milk	Meat	Milk	Fish	Crustacea
	pCi/L	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/L	pCi/kg	pCi/kg
Co-60	2.243E-02	2.640E+01	2.643E+01	2.662E+01	2.644E+01	3.615E+01	2.933E+00	6.712E-02	4.475E-02

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.



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Concentration of radionuclides in environmental media
at t = 1.344E+02 years

Radio-Nuclide	Contaminated Zone pCi/g	Surface Soil* pCi/g	Air Particulate pCi/m**3	Well Water pCi/L	Surface Water pCi/L
Ni-63	3.591E-01	3.218E-01	5.447E-06	6.008E-01	6.008E-03

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

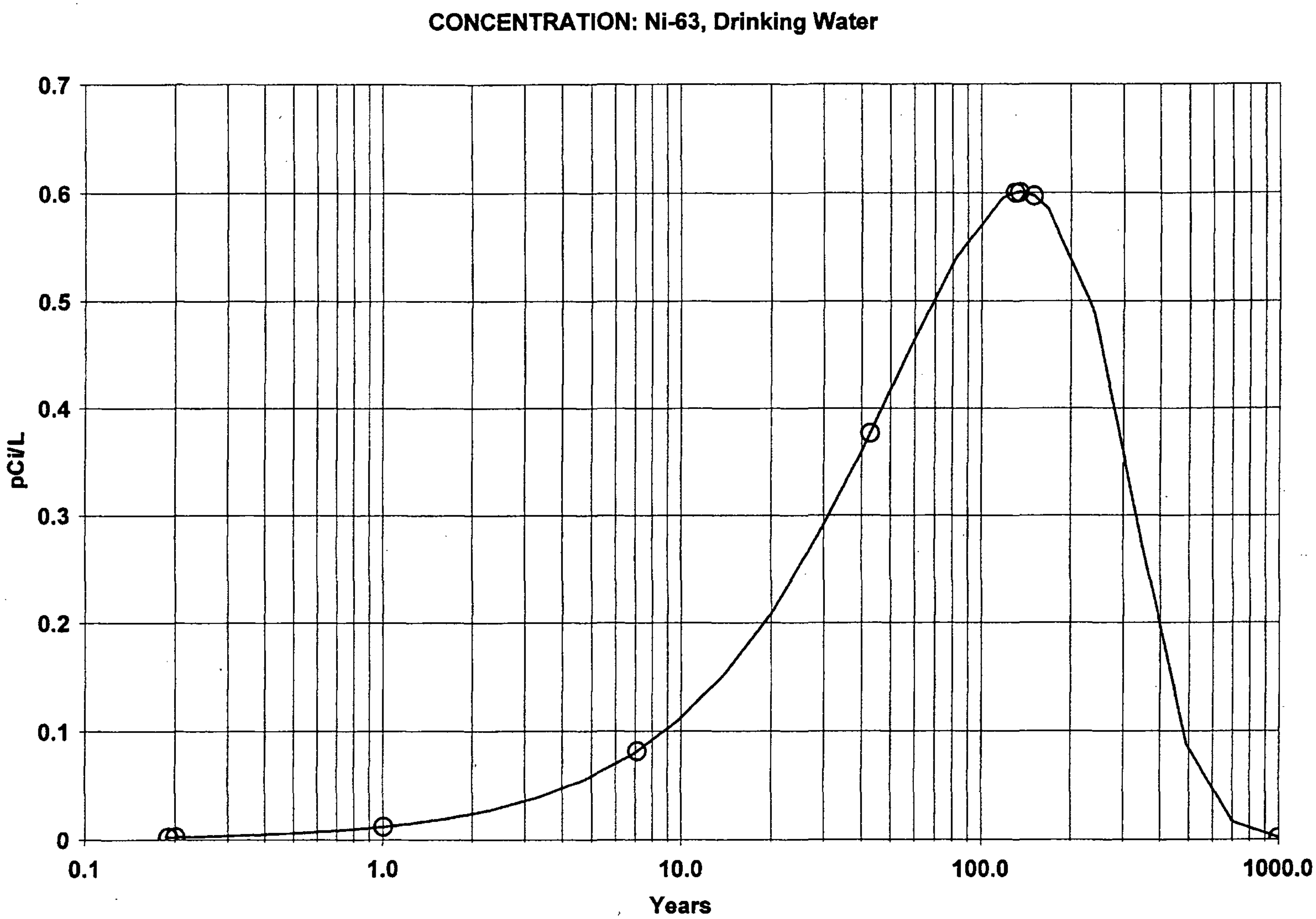
Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 1.344E+02 years*

Radio-Nuclide	Drinking Water pCi/L	Nonleafy Vegetable pCi/kg	Leafy Vegetable pCi/kg	Fodder Meat pCi/kg	Fodder Milk pCi/kg	Meat pCi/kg	Milk pCi/L	Fish pCi/kg	Crustacea pCi/kg
Ni-63	6.008E-01	1.785E+01	1.864E+01	1.874E+01	1.873E+01	7.323E+00	2.575E+01	6.007E-01	6.007E-01

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.



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Concentration of radionuclides in environmental media
at t = 2.000E-01 years

Radio-Nuclide	Contaminat-ted Zone	Surface Soil*	Air Par-ticulate	Well Water	Surface Water
	pCi/g	pCi/g	pCi/m**3	pCi/L	pCi/L
H-3	7.659E-01	1.021E-03	1.729E-08	6.770E+03	6.770E+01

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.
Concentration of H-3 in soil moisture = 0.000E+00 pCi/ml
Concentration of gaseous H-3 in air   = 0.000E+00 pCi/m**3

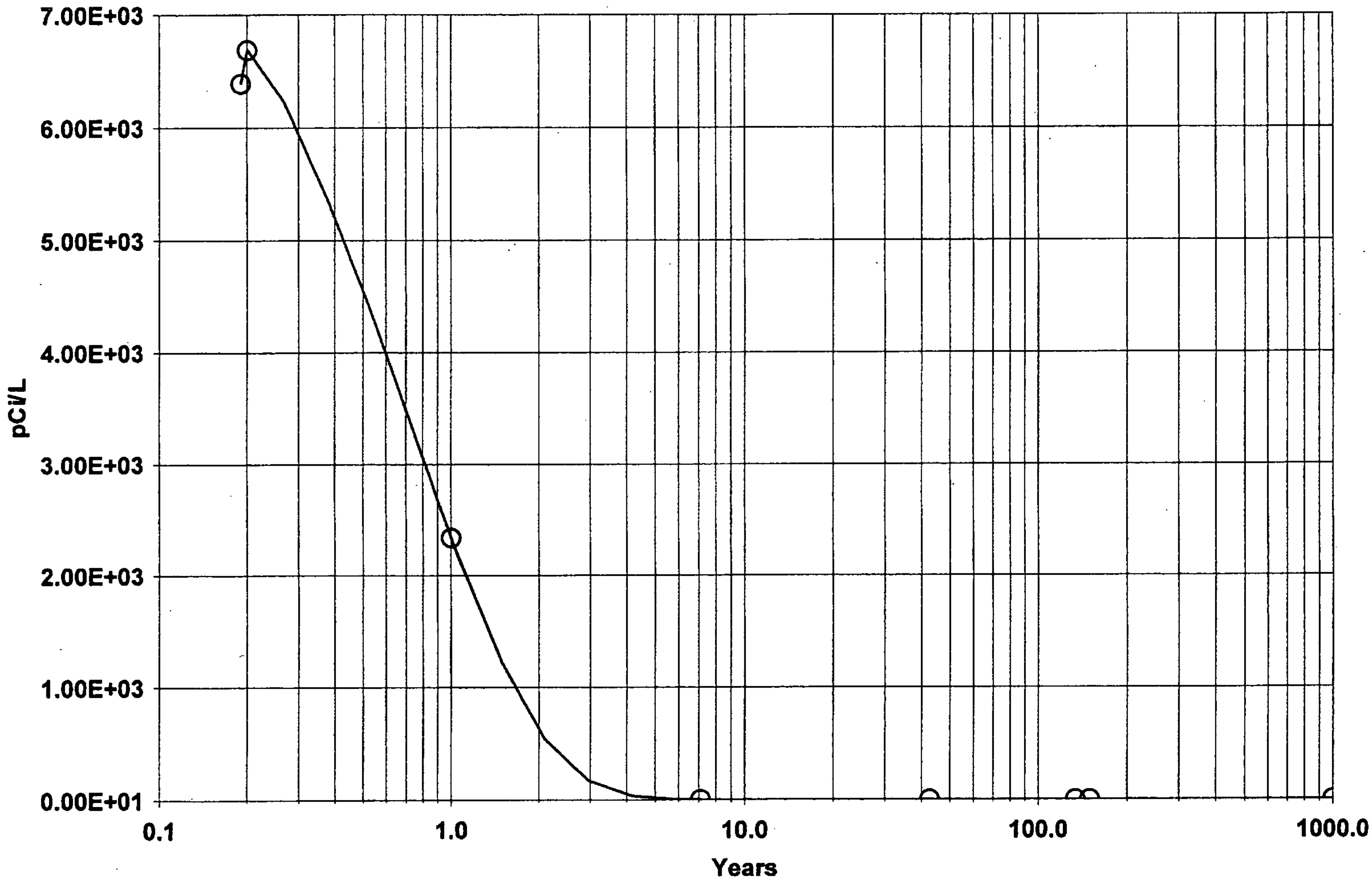
Concentration of radionuclides in foodstuff media
at t = 2.000E-01 years*

Radio-Nuclide	Drinking Water	Nonleafy Vegetable	Leafy Vegetable	Fodder Meat	Fodder Milk	Meat	Milk	Fish	Crustacea
	pCi/L	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/L	pCi/kg	pCi/kg
H-3	6.687E+03	8.966E+03	1.217E+04	5.295E+03	5.763E+03	3.497E+03	5.923E+03	6.186E+01	6.186E+01

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

CONCENTRATION: H-3, Drinking Water



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Attachment 6-10
Buried Piping List and Projected Concentration Calculation

Attachment 7, Table 1

Description of Buried Pipe/Conduit	Diam., in.	Length ft.	Surface Area, ft^2	Surface Area, cm^2	Surface Area, m^2	Vol. ft^3	Vol. cm^3	Vol. m^3
Turbine Hall to Sewage Treatment	12	206	647	6.0E+05	60	162	4.6E+06	4.6E+00
Staff Building to Basin West of MH 17	6	52	82	7.6E+04	8	10	2.9E+05	2.9E-01
Staff Building to Basin West of MH 17	10	43	113	1.0E+05	10	23	6.6E+05	6.6E-01
Staff Building to Basin West of MH 17	4	39	41	3.8E+04	4	3	9.6E+04	9.6E-02
Catch Basin West of MH 17 to MH 17	4	55	58	5.4E+04	5	5	1.4E+05	1.4E-01
MH 17 to MH 16	24	45	283	2.6E+05	26	141	4.0E+06	4.0E+00
MH 16 to MH 15	30	55	432	4.0E+05	40	270	7.6E+06	7.6E+00
MH 15 to Back River	30	170	1335	1.2E+06	124	834	2.4E+07	2.4E+01
Wart to Catch Basin East of Staff	4	60	63	5.8E+04	6	5	1.5E+05	1.5E-01
Catch Basin W. of MH 17 to Catch Basin E. of Tunnel	4	75	79	7.3E+04	7	7	1.9E+05	1.9E-01
Catch Basin E. of Staff Tunnel to Catch Basin N. of MH 20	4	26	27	2.5E+04	3	2	6.4E+04	6.4E-02
Warehouse to Catch Basin E. of Staff Tunnel	4	515	539	5.0E+05	50	45	1.3E+06	1.3E+00
Temp Sewer N. of MH 20 to Sewage Treatment Plant	6	340	534	5.0E+05	50	67	1.9E+06	1.9E+00
Temp Sewer N. of MH 20 to Sewage Treatment Plant	8	340	712	6.6E+05	66	119	3.4E+06	3.4E+00
Wart Roof Drain to MH 21	4	37	39	3.6E+04	4	3	9.1E+04	9.1E-02
MH21 to MH 22	15	103	404	3.8E+05	38	126	3.6E+06	3.6E+00
Roof Drain by Road TH to 15" Pipe	10	65	170	1.6E+05	16	35	1.0E+06	1.0E+00
MH 22 to MH 23	18	155	730	6.8E+05	68	274	7.8E+06	7.8E+00
Basin North of TK-16 to MH 23	24	52	327	3.0E+05	30	163	4.6E+06	4.6E+00
MH 23 to MH 24	18	56	264	2.5E+05	25	99	2.8E+06	2.8E+00
MH 24 to MH 27	24	110	691	6.4E+05	64	346	9.8E+06	9.8E+00
Fire Pumphouse Drain to MH 38	4	26	27	2.5E+04	3	2	6.4E+04	6.4E-02
MH 38 to MH 25b	10	86	225	2.1E+05	21	47	1.3E+06	1.3E+00
MH 25b to MH 25a	12	95	298	2.8E+05	28	75	2.1E+06	2.1E+00
MH 25a to MH 26	12	60	188	1.8E+05	18	47	1.3E+06	1.3E+00
MH 26 to MH 27	15	146	573	5.3E+05	53	179	5.1E+06	5.1E+00
WH Bldg. Drain, MH 26 to MH 27	4	404	423	3.9E+05	39	35	1.0E+06	1.0E+00
MH 27 to MH 28	24	190	1194	1.1E+06	111	597	1.7E+07	1.7E+01
MH 28 to MH 29	24	146	917	8.5E+05	85	459	1.3E+07	1.3E+01
MH 29 to West Shoreline	30	50	393	3.6E+05	36	245	6.9E+06	6.9E+00
MH 20 to MH 19	12	133	418	3.9E+05	39	104	3.0E+06	3.0E+00
Roof Drain, NW Corner to MH 19	6	10	16	1.5E+04	1	2	5.6E+04	5.6E-02
MH 19 to MH 18	12	60	188	1.8E+05	18	47	1.3E+06	1.3E+00
Roof Drain, NE Corner to MH 18	6	19	30	2.8E+04	3	4	1.1E+05	1.1E-01
MH 18 to MH5	12	110	346	3.2E+05	32	86	2.4E+06	2.4E+00
Roof Drain E Side of Turbine to MH 4	8	82	172	1.6E+05	16	29	8.1E+05	8.1E-01
MH 4 to MH 5	12	80	251	2.3E+05	23	63	1.8E+06	1.8E+00
MH 5 to Back River	15	200	785	7.3E+05	73	245	6.9E+06	6.9E+00
	Totals	4496	14015	1.3E+07	1302	5008	1.4E+08	1.4E+02

Table 1 is a compilation of the piping and conduit that will remain underground and at depths greater than three feet.

The piping dimensions were provided by Maine Yankee Engineering. The Surface Area was calculated for pipes assuming that they are cylinders using the equation $\pi*D*L$, where $\pi*D$ is the circumference of the cylinder and L is the length. The area was converted to metric units, cm^2 and m^2. The volume of the cylinder is determined using the equation $\pi*r^2*L$ and was also converted to metric units, cm^3 and m^3.

In its approach to model actual or potential residual radiological constituents Maine Yankee developed unitized dose factors for buried piping and conduit by assuming a unit inventory of 1 dpm/100 cm^2 gross beta radioactivity was present on the internal surfaces. This allows a calculation that ratios the total available gross beta radioactivity to the total volume of the piping. So, if the total surface area of the buried piping is 1.302 E7 cm^2, then the total gross beta radioactivity is 1.302E5 dpm:

$$1.302E7cm^2*1dpm/100cm^2=1.302E5dpm$$

If this gross beta radioactivity is divided by the total volume (1.42E+08 cm^3), this results in a concentration of 9.182E-4 dpm/cm^3. Using density of 1.6 g/cm3, and converting to pCi, we get a conversion factor of 2.59E-4 pCi/g per dpm/100cm^2. [9.182E-4dpm/cm^3*cm^3/1.6g*pCi/2.22 dpm]. This factor is used in the Buried Pipe and Conduit Worksheet.

Section 6.7.1.c discusses drinking water and irrigation model input parameters-porosity, bulk density, annual drinking water and irrigation rates used in this assessment.

Direct dose conversion factors were determined using the computer code Microshield. The dimension that equated to a volume of soil displaced by the pipes, 141.8 m^3, was calculated for input, assuming that the thickness was one meter (h=1)-

$$\pi*r^2*h = \text{Volume}$$
$$\pi*r^2*1 = 141.8\ m^3$$
$$\pi*r^2 = 141.8\ m^2$$
$$r^2 = 45.14\ m^2$$
$$r = 6.71836\ m$$

Source dimension having a radius of 671.8 cm and a thickness of 1 meter was used. The depth is assumed to be 1 meter below grade. Unit concentrations (e.g., 1 pCi/g ^{60}Co, ^{57}Co, etc.)of radionuclides were input to Microshield along with a density of 1.6 g/cm^3. The ICRP 51 Deep Dose Equivalent Rate-Rotational was determined by the code. The result was multiplied by the DandD default outdoor occupancy time of 0.1101 years or 964 hours. The direct dose factors are listed in the following tables.

Nuclide	Microshield Deep Dose Equivalent Rate-Rotational mSv/h	mrem/hr	Hours/year	mrem/year
Cs-134	2.291E-10	2.291E-8	964	2.21E-05
Cs-137	4.121E-11	4.121E-9	964	3.97E-06
Co-60	2.624E-9	2.624E-7	964	2.53E-04
Co-57	9.789E-14	9.789E-12	964	9.44E-09

Attachment 6-11
Buried Piping RESRAD Output

Table of Contents

Part I: Mixture Sums and Single Radionuclide Guidelines

Buried Piping, deep soil, water pathway only, site K_d values, Gerber adjustments –

Dose Conversion Factor (and Related) Parameter Summary
File: FGR 13 Morbidity

Menu	Parameter	Current Value	Default	Parameter Name
B-1	Dose conversion factors for inhalation, mrem/pCi:			
B-1	Ni-63	6.290E-06	6.290E-06	DCF2(1)
D-1	Dose conversion factors for ingestion, mrem/pCi:			
D-1	Ni-63	5.770E-07	5.770E-07	DCF3(1)
D-34	Food transfer factors:			
D-34	Ni-63 , plant/soil concentration ratio, dimensionless	5.000E-02	5.000E-02	RTF(1,1)
D-34	Ni-63 , beef/livestock-intake ratio, (pCi/kg)/(pCi/d)	5.000E-03	5.000E-03	RTF(1,2)
D-34	Ni-63 , milk/livestock-intake ratio, (pCi/L)/(pCi/d)	2.000E-02	2.000E-02	RTF(1,3)
D-5	Bioaccumulation factors, fresh water, L/kg:			
D-5	Ni-63 , fish	1.000E+02	1.000E+02	BIOFAC(1,1)
D-5	Ni-63 , crustacea and mollusks	1.000E+02	1.000E+02	BIOFAC(1,2)

Site-Specific Parameter Summary

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R011	Area of contaminated zone (m**2)	1.420E+02	1.000E+04	---	AREA
R011	Thickness of contaminated zone (m)	1.000E+00	2.000E+00	---	THICK0
R011	Length parallel to aquifer flow (m)	1.000E+02	1.000E+02	---	LCZPAQ
R011	Basic radiation dose limit (mrem/yr)	1.000E+01	2.500E+01	---	BRDL
R011	Time since placement of material (yr)	0.000E+00	0.000E+00	---	TI
R011	Times for calculations (yr)	1.900E-01	1.000E+00	---	T(2)
R011	Times for calculations (yr)	2.000E-01	3.000E+00	---	T(3)
R011	Times for calculations (yr)	1.000E+00	1.000E+01	---	T(4)
R011	Times for calculations (yr)	7.100E+00	3.000E+01	---	T(5)
R011	Times for calculations (yr)	4.270E+01	1.000E+02	---	T(6)
R011	Times for calculations (yr)	1.300E+02	3.000E+02	---	T(7)
R011	Times for calculations (yr)	1.344E+02	1.000E+03	---	T(8)
R011	Times for calculations (yr)	1.500E+02	0.000E+00	---	T(9)
R011	Times for calculations (yr)	1.000E+03	0.000E+00	---	T(10)
R012	Initial principal radionuclide (pCi/g): Ni-63	1.000E+00	0.000E+00	---	S1(1)
R012	Concentration in groundwater (pCi/L): Ni-63	not used	0.000E+00	---	W1(1)
R013	Cover depth (m)	1.500E-01	0.000E+00	---	COVER0
R013	Density of cover material (g/cm**3)	not used	1.500E+00	---	DENSCV
R013	Cover depth erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCV
R013	Density of contaminated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSCZ
R013	Contaminated zone erosion rate (m/yr)	1.000E-03	1.000E-03	---	VCZ
R013	Contaminated zone total porosity	3.000E-01	4.000E-01	---	TPCZ
R013	Contaminated zone field capacity	2.000E-01	2.000E-01	---	FCCZ
R013	Contaminated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+01	---	HCCZ
R013	Contaminated zone b parameter	4.050E+00	5.300E+00	---	BCZ
R013	Average annual wind speed (m/sec)	2.000E+00	2.000E+00	---	WIND
R013	Humidity in air (g/m**3)	not used	8.000E+00	---	HUMID
R013	Evapotranspiration coefficient	5.000E-01	5.000E-01	---	EVAPTR
R013	Precipitation (m/yr)	1.000E+00	1.000E+00	---	PRECIP
R013	Irrigation (m/yr)	2.000E-01	2.000E-01	---	RI
R013	Irrigation mode	overhead	overhead	---	IDITCH
R013	Runoff coefficient	2.000E-01	2.000E-01	---	RUNOFF
R013	Watershed area for nearby stream or pond (m**2)	1.000E+06	1.000E+06	---	WAREA
R013	Accuracy for water/soil computations	1.000E-03	1.000E-03	---	EPS
R014	Density of saturated zone (g/cm**3)	1.600E+00	1.500E+00	---	DENSAQ
R014	Saturated zone total porosity	3.000E-01	4.000E-01	---	TPSZ
R014	Saturated zone effective porosity	1.000E-02	2.000E-01	---	EPSZ
R014	Saturated zone field capacity	2.000E-01	2.000E-01	---	FCSZ
R014	Saturated zone hydraulic conductivity (m/yr)	3.200E+01	1.000E+02	---	HCSZ
R014	Saturated zone hydraulic gradient	2.000E-02	2.000E-02	---	HGWT
R014	Saturated zone b parameter	4.050E+00	5.300E+00	---	BSZ
R014	Water table drop rate (m/yr)	1.000E-03	1.000E-03	---	VWT
R014	Well pump intake depth (m below water table)	1.000E+01	1.000E+01	---	DWIBWT
R014	Model: Nondispersion (ND) or Mass-Balance (MB)	ND	ND	---	MODEL
R014	Well pumping rate (m**3/yr)	not used	2.500E+02	---	UW
R015	Number of unsaturated zone strata	1	1	---	NS

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R015	Unsat. zone 1, thickness (m)	0.000E+00	4.000E+00	---	H(1)
R015	Unsat. zone 1, soil density (g/cm**3)	1.600E+00	1.500E+00	---	DENSUZ(1)
R015	Unsat. zone 1, total porosity	3.000E-01	4.000E-01	---	TPUZ(1)
R015	Unsat. zone 1, effective porosity	2.000E-01	2.000E-01	---	EPUZ(1)
R015	Unsat. zone 1, field capacity	2.000E-01	2.000E-01	---	FCUZ(1)
R015	Unsat. zone 1, soil-specific b parameter	5.300E+00	5.300E+00	---	BUZ(1)
R015	Unsat. zone 1, hydraulic conductivity (m/yr)	1.000E+03	1.000E+01	---	HCUZ(1)
R016	Distribution coefficients for Ni-63				
R016	Contaminated zone (cm**3/g)	2.740E+02	1.000E+03	---	DCNUCC(1)
R016	Unsaturated zone 1 (cm**3/g)	2.740E+02	1.000E+03	---	DCNUCU(1,1)
R016	Saturated zone (cm**3/g)	2.740E+02	1.000E+03	---	DCNUCS(1)
R016	Leach rate (/yr)	0.000E+00	0.000E+00	1.140E-03	ALEACH(1)
R016	Solubility constant	0.000E+00	0.000E+00	not used	SOLUBK(1)
R017	Inhalation rate (m**3/yr)	not used	8.400E+03	---	INHALR
R017	Mass loading for inhalation (g/m**3)	not used	1.000E-04	---	MLINH
R017	Exposure duration	3.000E+01	3.000E+01	---	ED
R017	Shielding factor, inhalation	not used	4.000E-01	---	SHF3
R017	Shielding factor, external gamma	not used	7.000E-01	---	SHF1
R017	Fraction of time spent indoors	not used	5.000E-01	---	FIND
R017	Fraction of time spent outdoors (on site)	not used	2.500E-01	---	FOTD
R017	Shape factor flag, external gamma	not used	1.000E+00	>0 shows circular AREA.	FS
R017	Radii of shape factor array (used if FS = -1):				
R017	Outer annular radius (m), ring 1:	not used	5.000E+01	---	RAD_SHAPE(1)
R017	Outer annular radius (m), ring 2:	not used	7.071E+01	---	RAD_SHAPE(2)
R017	Outer annular radius (m), ring 3:	not used	0.000E+00	---	RAD_SHAPE(3)
R017	Outer annular radius (m), ring 4:	not used	0.000E+00	---	RAD_SHAPE(4)
R017	Outer annular radius (m), ring 5:	not used	0.000E+00	---	RAD_SHAPE(5)
R017	Outer annular radius (m), ring 6:	not used	0.000E+00	---	RAD_SHAPE(6)
R017	Outer annular radius (m), ring 7:	not used	0.000E+00	---	RAD_SHAPE(7)
R017	Outer annular radius (m), ring 8:	not used	0.000E+00	---	RAD_SHAPE(8)
R017	Outer annular radius (m), ring 9:	not used	0.000E+00	---	RAD_SHAPE(9)
R017	Outer annular radius (m), ring 10:	not used	0.000E+00	---	RAD_SHAPE(10)
R017	Outer annular radius (m), ring 11:	not used	0.000E+00	---	RAD_SHAPE(11)
R017	Outer annular radius (m), ring 12:	not used	0.000E+00	---	RAD_SHAPE(12)
R017	Fractions of annular areas within AREA:				
R017	Ring 1	not used	1.000E+00	---	FRACA(1)
R017	Ring 2	not used	2.732E-01	---	FRACA(2)
R017	Ring 3	not used	0.000E+00	---	FRACA(3)
R017	Ring 4	not used	0.000E+00	---	FRACA(4)
R017	Ring 5	not used	0.000E+00	---	FRACA(5)
R017	Ring 6	not used	0.000E+00	---	FRACA(6)
R017	Ring 7	not used	0.000E+00	---	FRACA(7)
R017	Ring 8	not used	0.000E+00	---	FRACA(8)
R017	Ring 9	not used	0.000E+00	---	FRACA(9)
R017	Ring 10	not used	0.000E+00	---	FRACA(10)
R017	Ring 11	not used	0.000E+00	---	FRACA(11)
R017	Ring 12	not used	0.000E+00	---	FRACA(12)

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
R018	Fruits, vegetables and grain consumption (kg/yr)	not used	1.600E+02	---	DIET(1)
R018	Leafy vegetable consumption (kg/yr)	not used	1.400E+01	---	DIET(2)
R018	Milk consumption (L/yr)	not used	9.200E+01	---	DIET(3)
R018	Meat and poultry consumption (kg/yr)	not used	6.300E+01	---	DIET(4)
R018	Fish consumption (kg/yr)	not used	5.400E+00	---	DIET(5)
R018	Other seafood consumption (kg/yr)	not used	9.000E-01	---	DIET(6)
R018	Soil ingestion rate (g/yr)	not used	3.650E+01	---	SOIL
R018	Drinking water intake (L/yr)	4.780E+02	5.100E+02	---	DWI
R018	Contamination fraction of drinking water	1.000E+00	1.000E+00	---	FDW
R018	Contamination fraction of household water	not used	1.000E+00	---	FHHW
R018	Contamination fraction of livestock water	not used	1.000E+00	---	FLW
R018	Contamination fraction of irrigation water	not used	1.000E+00	---	FIRW
R018	Contamination fraction of aquatic food	not used	5.000E-01	---	FR9
R018	Contamination fraction of plant food	not used	-1	---	FPLANT
R018	Contamination fraction of meat	not used	-1	---	FMEAT
R018	Contamination fraction of milk	not used	-1	---	FMILK
R019	Livestock fodder intake for meat (kg/day)	not used	6.800E+01	---	LFI5
R019	Livestock fodder intake for milk (kg/day)	not used	5.500E+01	---	LFI6
R019	Livestock water intake for meat (L/day)	not used	5.000E+01	---	LWI5
R019	Livestock water intake for milk (L/day)	not used	1.600E+02	---	LWI6
R019	Livestock soil intake (kg/day)	not used	5.000E-01	---	LSI
R019	Mass loading for foliar deposition (g/m**3)	not used	1.000E-04	---	MLFD
R019	Depth of soil mixing layer (m)	not used	1.500E-01	---	DM
R019	Depth of roots (m)	not used	9.000E-01	---	DROOT
R019	Drinking water fraction from ground water	1.000E+00	1.000E+00	---	FGWDW
R019	Household water fraction from ground water	not used	1.000E+00	---	FGWHH
R019	Livestock water fraction from ground water	not used	1.000E+00	---	FGWLW
R019	Irrigation fraction from ground water	not used	1.000E+00	---	FGWIR
R19B	Wet weight crop yield for Non-Leafy (kg/m**2)	not used	7.000E-01	---	YV(1)
R19B	Wet weight crop yield for Leafy (kg/m**2)	not used	1.500E+00	---	YV(2)
R19B	Wet weight crop yield for Fodder (kg/m**2)	not used	1.100E+00	---	YV(3)
R19B	Growing Season for Non-Leafy (years)	not used	1.700E-01	---	TE(1)
R19B	Growing Season for Leafy (years)	not used	2.500E-01	---	TE(2)
R19B	Growing Season for Fodder (years)	not used	8.000E-02	---	TE(3)
R19B	Translocation Factor for Non-Leafy	not used	1.000E-01	---	TIV(1)
R19B	Translocation Factor for Leafy	not used	1.000E+00	---	TIV(2)
R19B	Translocation Factor for Fodder	not used	1.000E+00	---	TIV(3)
R19B	Dry Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RDRY(1)
R19B	Dry Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RDRY(2)
R19B	Dry Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RDRY(3)
R19B	Wet Foliar Interception Fraction for Non-Leafy	not used	2.500E-01	---	RWET(1)
R19B	Wet Foliar Interception Fraction for Leafy	not used	2.500E-01	---	RWET(2)
R19B	Wet Foliar Interception Fraction for Fodder	not used	2.500E-01	---	RWET(3)
R19B	Weathering Removal Constant for Vegetation	not used	2.000E+01	---	WLAM
C14	C-12 concentration in water (g/cm**3)	not used	2.000E-05	---	C12WTR
C14	C-12 concentration in contaminated soil (g/g)	not used	3.000E-02	---	C12CZ
C14	Fraction of vegetation carbon from soil	not used	2.000E-02	---	CSOIL
C14	Fraction of vegetation carbon from air	not used	9.800E-01	---	CAIR

Site-Specific Parameter Summary (continued)

Menu	Parameter	User Input	Default	Used by RESRAD (If different from user input)	Parameter Name
C14	C-14 evasion layer thickness in soil (m)	not used	3.000E-01	---	DMC
C14	C-14 evasion flux rate from soil (1/sec)	not used	7.000E-07	---	EVSN
C14	C-12 evasion flux rate from soil (1/sec)	not used	1.000E-10	---	REVSN
C14	Fraction of grain in beef cattle feed	not used	8.000E-01	---	AVFG4
C14	Fraction of grain in milk cow feed	not used	2.000E-01	---	AVFG5
C14	DCF correction factor for gaseous forms of C14	not used	8.894E+01	---	CO2F
STOR	Storage times of contaminated foodstuffs (days):				
STOR	Fruits, non-leafy vegetables, and grain	1.400E+01	1.400E+01	---	STOR_T(1)
STOR	Leafy vegetables	1.000E+00	1.000E+00	---	STOR_T(2)
STOR	Milk	1.000E+00	1.000E+00	---	STOR_T(3)
STOR	Meat and poultry	2.000E+01	2.000E+01	---	STOR_T(4)
STOR	Fish	7.000E+00	7.000E+00	---	STOR_T(5)
STOR	Crustacea and mollusks	7.000E+00	7.000E+00	---	STOR_T(6)
STOR	Well water	1.000E+00	1.000E+00	---	STOR_T(7)
STOR	Surface water	1.000E+00	1.000E+00	---	STOR_T(8)
STOR	Livestock fodder	4.500E+01	4.500E+01	---	STOR_T(9)
R021	Thickness of building foundation (m)	not used	1.500E-01	---	FLOOR1
R021	Bulk density of building foundation (g/cm**3)	not used	2.400E+00	---	DENSFL
R021	Total porosity of the cover material	not used	4.000E-01	---	TPCV
R021	Total porosity of the building foundation	not used	1.000E-01	---	TPFL
R021	Volumetric water content of the cover material	not used	5.000E-02	---	PH2OCV
R021	Volumetric water content of the foundation	not used	3.000E-02	---	PH2OFL
R021	Diffusion coefficient for radon gas (m/sec):				
R021	in cover material	not used	2.000E-06	---	DIFCV
R021	in foundation material	not used	3.000E-07	---	DIFFL
R021	in contaminated zone soil	not used	2.000E-06	---	DIFCZ
R021	Radon vertical dimension of mixing (m)	not used	2.000E+00	---	HMIX
R021	Average building air exchange rate (1/hr)	not used	5.000E-01	---	REXG
R021	Height of the building (room) (m)	not used	2.500E+00	---	HRM
R021	Building interior area factor	not used	0.000E+00	---	FAI
R021	Building depth below ground surface (m)	not used	-1.000E+00	---	DMFL
R021	Emanating power of Rn-222 gas	not used	2.500E-01	---	EMANA(1)
R021	Emanating power of Rn-220 gas	not used	1.500E-01	---	EMANA(2)
TITL	Number of graphical time points	32	---	---	NPTS
TITL	Maximum number of integration points for dose	17	---	---	LYMAX
TITL	Maximum number of integration points for risk	257	---	---	KYMAX

Summary of Pathway Selections

Pathway	User Selection
1 -- external gamma	suppressed
2 -- inhalation (w/o radon)	suppressed
3 -- plant ingestion	suppressed
4 -- meat ingestion	suppressed
5 -- milk ingestion	suppressed
6 -- aquatic foods	suppressed
7 -- drinking water	active
8 -- soil ingestion	suppressed
9 -- radon	suppressed
Find peak pathway doses	active

Contaminated Zone Dimensions	
Area:	142.00 square meters
Thickness:	1.00 meters
Cover Depth:	0.15 meters

Initial Soil Concentrations, pCi/g	
Ni-63	1.000E+00

Total Dose TDOSE(t), mrem/yr
Basic Radiation Dose Limit = 1.000E+01 mrem/yr
Total Mixture Sum M(t) = Fraction of Basic Dose Limit Received at Time (t)

t (years):	0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.344E+02	1.500E+02	1.000E+03
TDOSE(t):	6.007E-08	8.290E-08	8.410E-08	1.796E-07	8.693E-07	3.749E-06	5.749E-06	5.744E-06	5.677E-06	3.484E-09
M(t):	6.007E-09	8.290E-09	8.410E-09	1.796E-08	8.693E-08	3.749E-07	5.749E-07	5.744E-07	5.677E-07	3.484E-10

Maximum TDOSE(t): 5.750E-06 *mrem/yr* at t = 128.3 ± 0.3 years

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.283E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio- Nuclide	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.283E+02 years

Water Dependent Pathways

Radio- Nuclide	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	5.750E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.750E-06	1.0000
Total	5.750E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.750E-06	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 0.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	6.007E-08	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.007E-08	1.0000
Total	6.007E-08	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	6.007E-08	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.900E-01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.900E-01 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	8.290E-08	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.290E-08	1.0000
Total	8.290E-08	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.290E-08	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 2.000E-01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 2.000E-01 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	8.410E-08	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.410E-08	1.0000
Total	8.410E-08	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.410E-08	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+00 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	1.796E-07	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.796E-07	1.0000
Total	1.796E-07	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	1.796E-07	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 7.100E+00 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground mrem/yr	Ground fract.	Inhalation mrem/yr	Inhalation fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	Soil mrem/yr	Soil fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 7.100E+00 years

Water Dependent Pathways

Radio-Nuclide	Water mrem/yr	Water fract.	Fish mrem/yr	Fish fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	All Pathways* mrem/yr	All Pathways* fract.
Ni-63	8.693E-07	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.693E-07	1.0000
Total	8.693E-07	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	8.693E-07	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 4.270E+01 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 4.270E+01 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	3.749E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.749E-06	1.0000
Total	3.749E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.749E-06	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.300E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio- Nuclide	Ground mrem/yr	Ground fract.	Inhalation mrem/yr	Inhalation fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	Soil mrem/yr	Soil fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.300E+02 years

Water Dependent Pathways

Radio- Nuclide	Water mrem/yr	Water fract.	Fish mrem/yr	Fish fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	All Pathways* mrem/yr	All Pathways* fract.
Ni-63	5.749E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.749E-06	1.0000
Total	5.749E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.749E-06	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.344E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-Nuclide	Ground mrem/yr	Ground fract.	Inhalation mrem/yr	Inhalation fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	Soil mrem/yr	Soil fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.344E+02 years

Water Dependent Pathways

Radio-Nuclide	Water mrem/yr	Water fract.	Fish mrem/yr	Fish fract.	Radon mrem/yr	Radon fract.	Plant mrem/yr	Plant fract.	Meat mrem/yr	Meat fract.	Milk mrem/yr	Milk fract.	All Pathways* mrem/yr	All Pathways* fract.
Ni-63	5.744E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.744E-06	1.0000
Total	5.744E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.744E-06	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.500E+02 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.500E+02 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	5.677E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.677E-06	1.0000
Total	5.677E-06	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	5.677E-06	1.0000

*Sum of all water independent and dependent pathways.

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years

Water Independent Pathways (Inhalation excludes radon)

Radio-	Ground		Inhalation		Radon		Plant		Meat		Milk		Soil	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000
Total	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000

Total Dose Contributions TDOSE(i,p,t) for Individual Radionuclides (i) and Pathways (p)
As mrem/yr and Fraction of Total Dose At t = 1.000E+03 years

Water Dependent Pathways

Radio-	Water		Fish		Radon		Plant		Meat		Milk		All Pathways*	
Nuclide	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.	mrem/yr	fract.
Ni-63	3.484E-09	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.484E-09	1.0000
Total	3.484E-09	1.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	0.000E+00	0.0000	3.484E-09	1.0000

*Sum of all water independent and dependent pathways.

Dose/Source Ratios Summed Over All Pathways
Parent and Progeny Principal Radionuclide Contributions Indicated

Parent (i)	Product (j)	Branch Fraction*	DSR(j,t) (mrem/yr)/(pCi/g) t= 0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.344E+02	1.500E+02	1.000E+03
Ni-63	Ni-63	1.000E+00	6.007E-08	8.290E-08	8.410E-08	1.796E-07	8.693E-07	3.749E-06	5.749E-06	5.744E-06	5.677E-06	3.484E-09

*Branch Fraction is the cumulative factor for the j't principal radionuclide daughter: CUMBRF(j) = BRF(1)*BRF(2)* ... BRF(j).
The DSR includes contributions from associated (half-life ≤ 0.5 yr) daughters.

Single Radionuclide Soil Guidelines G(i,t) in pCi/g
Basic Radiation Dose Limit = 1.000E+01 mrem/yr

Nuclide (i)	t= 0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.344E+02	1.500E+02	1.000E+03
Ni-63	1.665E+08	1.206E+08	1.189E+08	5.569E+07	1.150E+07	2.668E+06	1.739E+06	1.741E+06	1.761E+06	2.871E+09

Summed Dose/Source Ratios DSR(i,t) in (mrem/yr)/(pCi/g)
and Single Radionuclide Soil Guidelines G(i,t) in pCi/g
at tmin = time of minimum single radionuclide soil guideline
and at tmax = time of maximum total dose = 128.3 ± 0.3 years

Nuclide (i)	Initial (pCi/g)	tmin (years)	DSR(i,tmin)	G(i,tmin) (pCi/g)	DSR(i,tmax)	G(i,tmax) (pCi/g)
Ni-63	1.000E+00	128.3 ± 0.3	5.750E-06	1.739E+06	5.750E-06	1.739E+06

Individual Nuclide Dose Summed Over All Pathways
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	BRF(i)	DOSE(j,t), mrem/yr t= 0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.344E+02	1.500E+02	1.000E+03
Ni-63	Ni-63	1.000E+00	6.007E-08	8.290E-08	8.410E-08	1.796E-07	8.693E-07	3.749E-06	5.749E-06	5.744E-06	5.677E-06	3.484E-09

BRF(i) is the branch fraction of the parent nuclide.

Individual Nuclide Soil Concentration
Parent Nuclide and Branch Fraction Indicated

Nuclide (j)	Parent (i)	BRF(i)	S(j,t), pCi/g t= 0.000E+00	1.900E-01	2.000E-01	1.000E+00	7.100E+00	4.270E+01	1.300E+02	1.344E+02	1.500E+02	1.000E+03
Ni-63	Ni-63	1.000E+00	1.000E+00	9.984E-01	9.983E-01	9.917E-01	9.424E-01	6.998E-01	3.373E-01	3.251E-01	2.854E-01	2.340E-04

BRF(i) is the branch fraction of the parent nuclide.

RESCALC.EXE execution time = 0.55 seconds

Table of Contents

Part IV: Concentration of Radionuclides

Concentration of radionuclides in environmental media
at t = 1.300E+02 years

Radio-Nuclide	Contaminated Zone	Surface Soil*	Air Particulate	Well Water	Surface Water
	pCi/g	pCi/g	pCi/m**3	pCi/L	pCi/L
Ni-63	3.373E-01	2.923E-01	3.171E-06	2.085E-02	8.143E-05

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 1.300E+02 years*

Radio-Nuclide	Drinking Water	Nonleafy Vegetable	Leafy Vegetable	Fodder Meat	Fodder Milk	Meat	Milk	Fish	Crustacea
	pCi/L	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/L	pCi/kg	pCi/kg
Ni-63	2.085E-02	1.650E+01	1.652E+01	1.653E+01	1.653E+01	6.355E+00	2.117E+01	8.142E-03	8.142E-03

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 1.000E+00 years

Radio-Nuclide	Contaminated Zone pCi/g	Surface Soil* pCi/g	Air Particulate pCi/m**3	Well Water pCi/L	Surface Water pCi/L
Co-57	3.924E-01	2.616E-03	2.838E-08	1.148E-04	4.486E-07

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 1.000E+00 years*

Radio-Nuclide	Drinking Water pCi/L	Nonleafy Vegetable pCi/kg	Leafy Vegetable pCi/kg	Fodder Meat pCi/kg	Fodder Milk pCi/kg	Meat pCi/kg	Milk pCi/L	Fish pCi/kg	Crustacea pCi/kg
Co-57	1.145E-04	2.620E+01	2.620E+01	2.757E+01	2.626E+01	3.565E+01	2.884E+00	1.319E-04	8.793E-05

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 7.100E+00 years

Radio-Nuclide	Contaminated Zone	Surface Soil*	Air Particulate	Well Water	Surface Water
	pCi/g	pCi/g	pCi/m**3	pCi/L	pCi/L
Co-60	3.905E-01	1.848E-02	2.005E-07	8.139E-04	3.179E-06

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 7.100E+00 years*

Radio-Nuclide	Drinking Water	Nonleafy Vegetable	Leafy Vegetable	Fodder Meat	Fodder Milk	Meat	Milk	Fish	Crustacea
	pCi/L	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/L	pCi/kg	pCi/kg
Co-60	8.136E-04	2.628E+01	2.628E+01	2.647E+01	2.629E+01	3.593E+01	2.910E+00	9.511E-04	6.341E-04

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 1.000E+00 years

Radio- Nuclide	Contaminat- ted Zone pCi/g	Surface Soil* pCi/g	Air Par- ticulate pCi/m**3	Well Water pCi/L	Surface Water pCi/L
Cs-134	7.143E-01	4.762E-03	5.166E-08	1.474E-05	5.758E-08

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 1.000E+00 years*

Radio- Nuclide	Drinking Water pCi/L	Nonleafy Vegetable pCi/kg	Leafy Vegetable pCi/kg	Fodder Meat pCi/kg	Fodder Milk pCi/kg	Meat pCi/kg	Milk pCi/L	Fish pCi/kg	Crustacea pCi/kg
Cs-134	1.469E-05	2.384E+01	2.384E+01	2.428E+01	2.386E+01	4.870E+01	1.051E+01	1.126E-04	5.631E-06

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 4.270E+01 years

Radio-Nuclide	Contaminated Zone	Surface Soil*	Air Particulate	Well Water	Surface Water
	pCi/g	pCi/g	pCi/m**3	pCi/L	pCi/L
Cs-137	3.687E-01	1.050E-01	1.139E-06	3.269E-04	1.277E-06

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 4.270E+01 years*

Radio-Nuclide	Drinking Water	Nonleafy Vegetable	Leafy Vegetable	Fodder Meat	Fodder Milk	Meat	Milk	Fish	Crustacea
	pCi/L	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/kg	pCi/L	pCi/kg	pCi/kg
Cs-137	3.269E-04	1.299E+01	1.299E+01	1.301E+01	1.299E+01	2.807E+01	6.136E+00	2.552E-03	1.276E-04

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 1.000E+00 years

Radio-Nuclide	Contaminated Zone pCi/g	Surface Soil* pCi/g	Air Particulate pCi/m**3	Well Water pCi/L	Surface Water pCi/L
Fe-55	7.729E-01	5.152E-03	5.589E-08	2.262E-04	8.836E-07

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in foodstuff media
at t = 1.000E+00 years*

Radio-Nuclide	Drinking Water pCi/L	Nonleafy Vegetable pCi/kg	Leafy Vegetable pCi/kg	Fodder Meat pCi/kg	Fodder Milk pCi/kg	Meat pCi/kg	Milk pCi/L	Fish pCi/kg	Crustacea pCi/kg
Fe-55	2.256E-04	6.450E-01	6.453E-01	6.543E-01	6.457E-01	9.264E-01	1.143E-02	1.732E-04	2.771E-03

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 1.900E-01 years

Radio-Nuclide	Contaminated Zone pCi/g	Surface Soil* pCi/g	Air Particulate pCi/m**3	Well Water pCi/L	Surface Water pCi/L
H-3	4.955E-01	6.276E-04	6.808E-09	1.914E+02	7.476E-01

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.
Concentration of H-3 in soil moisture = 0.000E+00 pCi/ml
Concentration of gaseous H-3 in air   = 0.000E+00 pCi/m**3

Concentration of radionuclides in foodstuff media
at t = 1.900E-01 years*

Radio-Nuclide	Drinking Water pCi/L	Nonleafy Vegetable pCi/kg	Leafy Vegetable pCi/kg	Fodder Meat pCi/kg	Fodder Milk pCi/kg	Meat pCi/kg	Milk pCi/L	Fish pCi/kg	Crustacea pCi/kg
H-3	1.893E+02	3.072E+03	2.781E+03	4.916E+03	4.080E+03	1.956E+03	1.097E+03	6.915E-01	6.915E-01

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

Concentration of radionuclides in environmental media
at t = 4.000E+01 years

Radio-Nuclide	Contaminated Zone pCi/g	Surface Soil* pCi/g	Air Particulate pCi/m**3	Well Water pCi/L	Surface Water pCi/L
Sr-90	3.555E-01	9.479E-02	1.028E-06	2.147E-02	8.389E-05

*The Surface Soil is the top layer of soil within the user specified mixing zone/depth.

Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

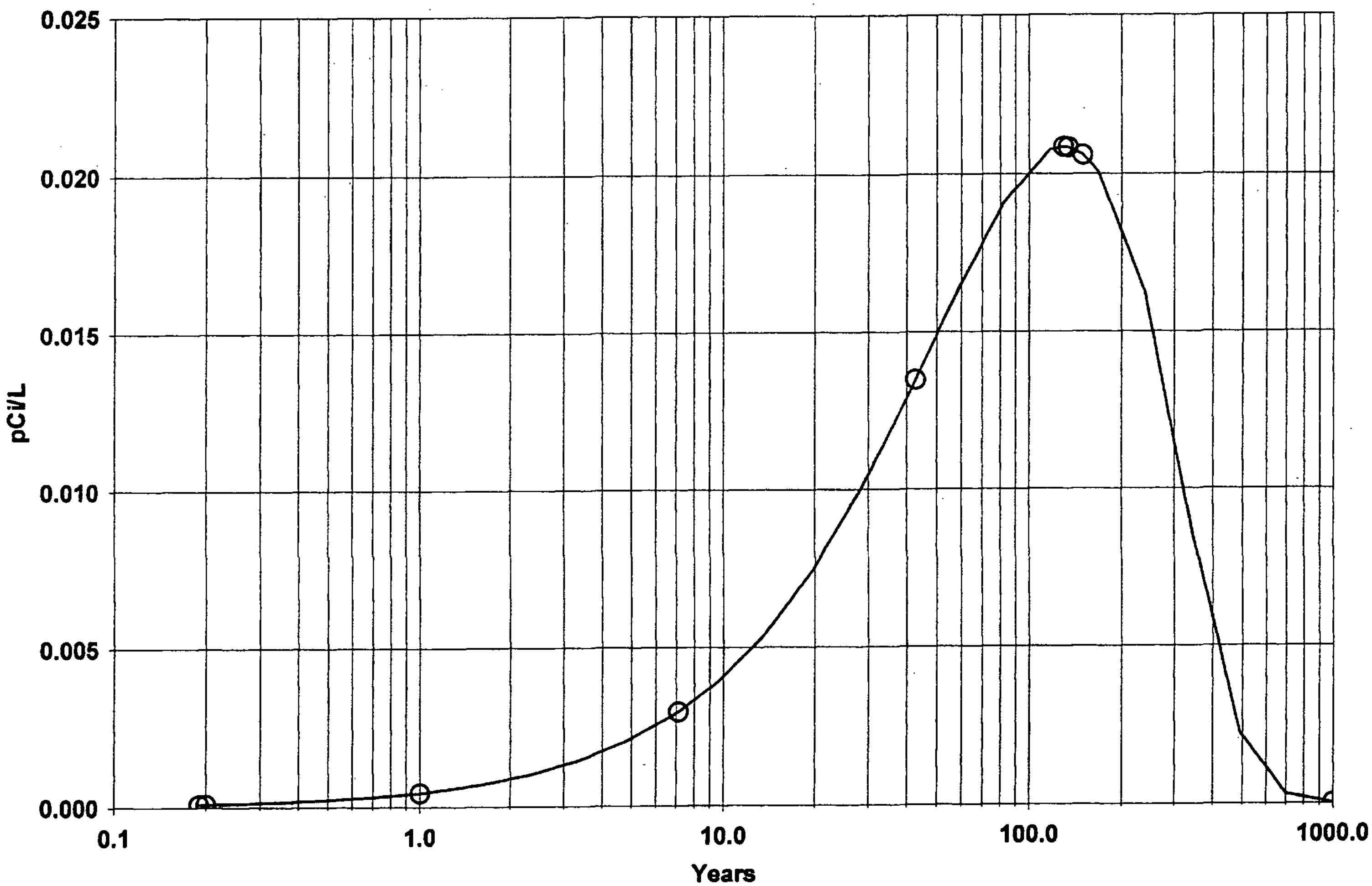
Concentration of radionuclides in foodstuff media
at t = 4.000E+01 years*

Radio-Nuclide	Drinking Water pCi/L	Nonleafy Vegetable pCi/kg	Leafy Vegetable pCi/kg	Fodder Meat pCi/kg	Fodder Milk pCi/kg	Meat pCi/kg	Milk pCi/L	Fish pCi/kg	Crustacea pCi/kg
Sr-90	2.147E-02	9.362E+01	9.365E+01	9.378E+01	9.366E+01	5.134E+01	1.040E+01	5.031E-03	8.385E-03

*Concentrations are at consumption time and include radioactive decay and ingrowth during storage time.
For livestock fodder, consumption time is t minus meat or milk storage time.

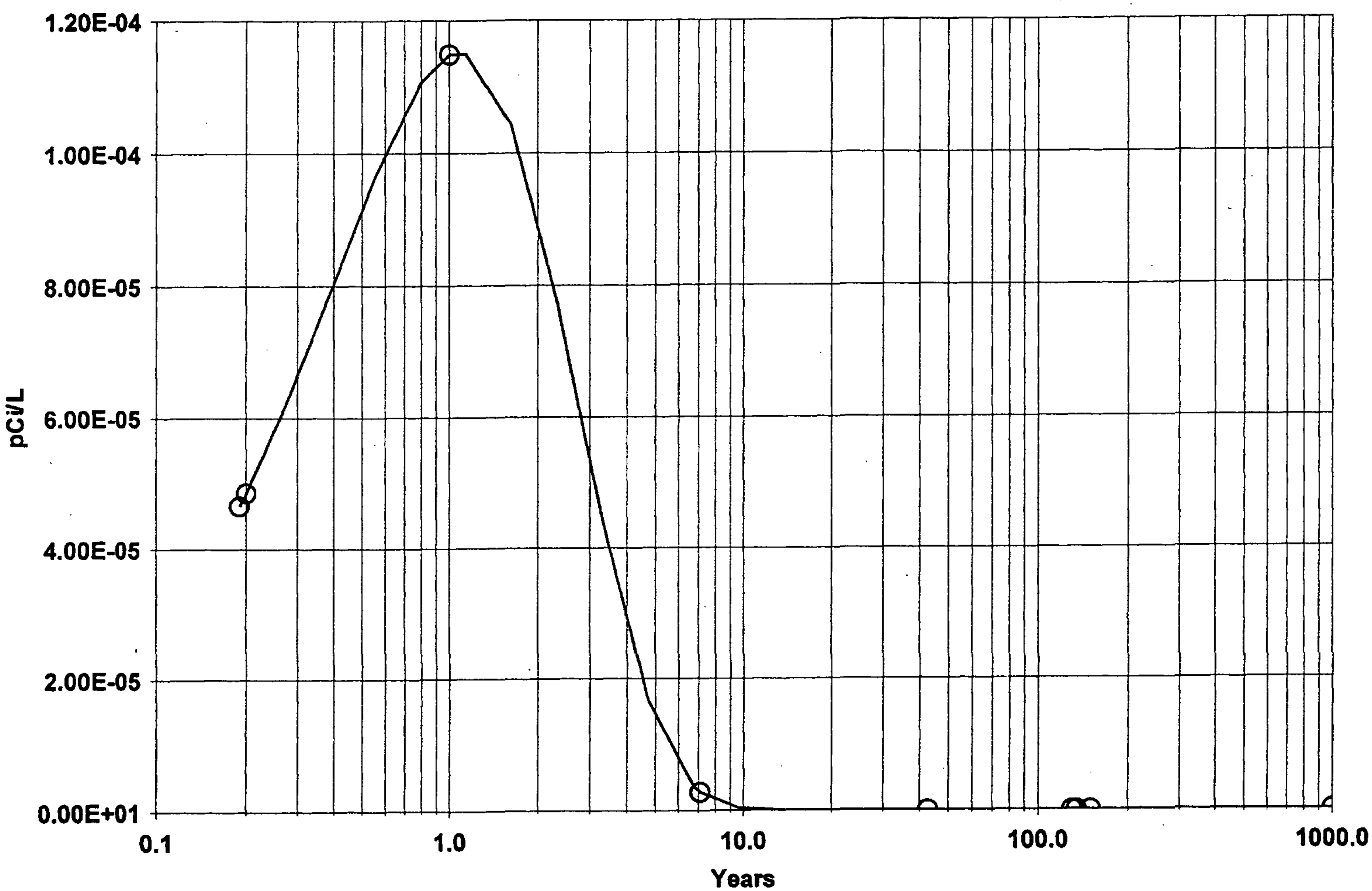
Concentrations in the media occurring in pathways that are suppressed are calculated using the current input parameters, i.e. using parameters appearing in the input screen when the pathways are active.

CONCENTRATION: Ni-63, Well Water



buriedpipe092402Ni63b405.RAD 09/25/2002 08:16

CONCENTRATION: Co-57, Well Water



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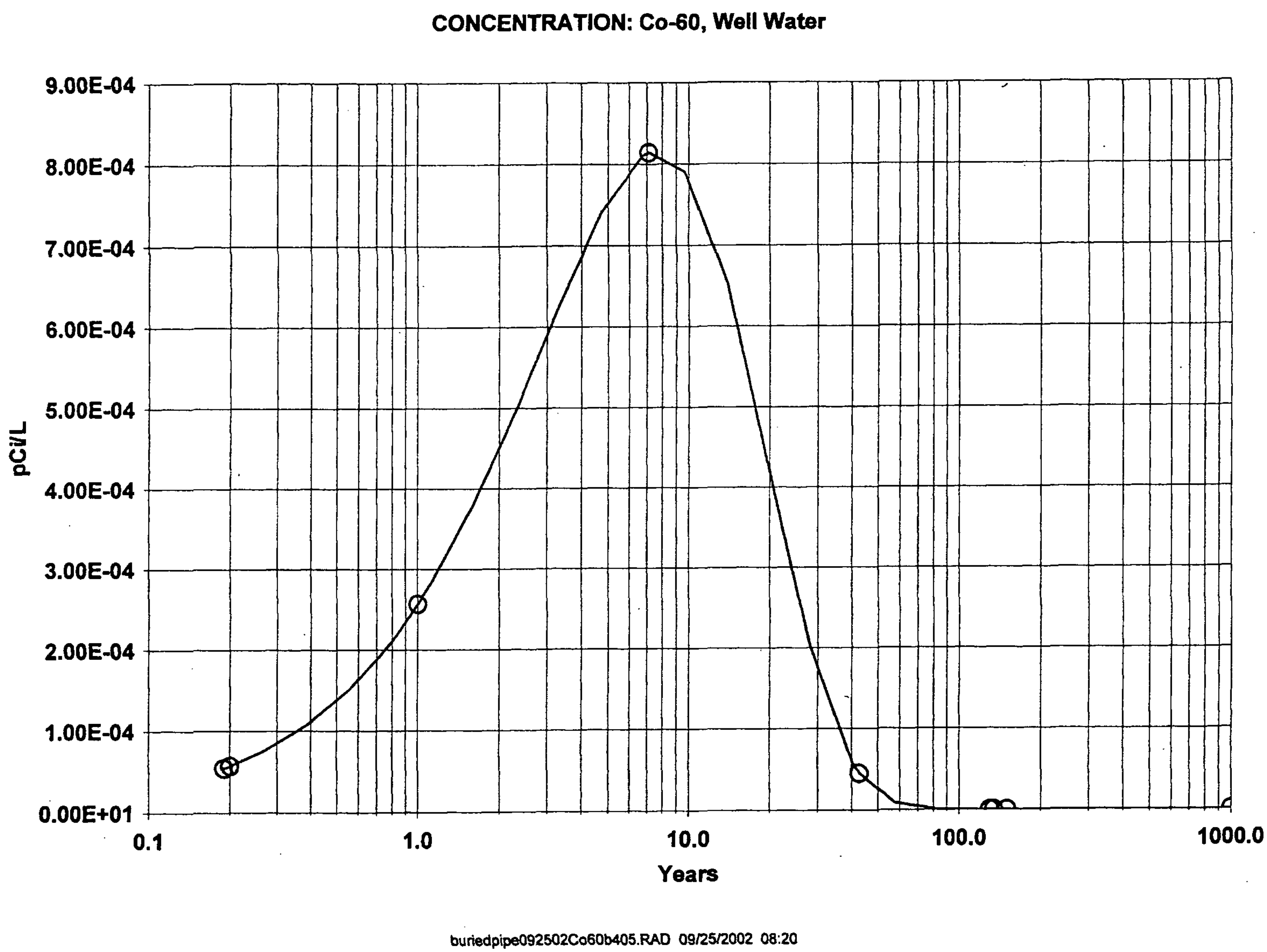
CONCENTRATION: Co-60, Well Water
pCi/L
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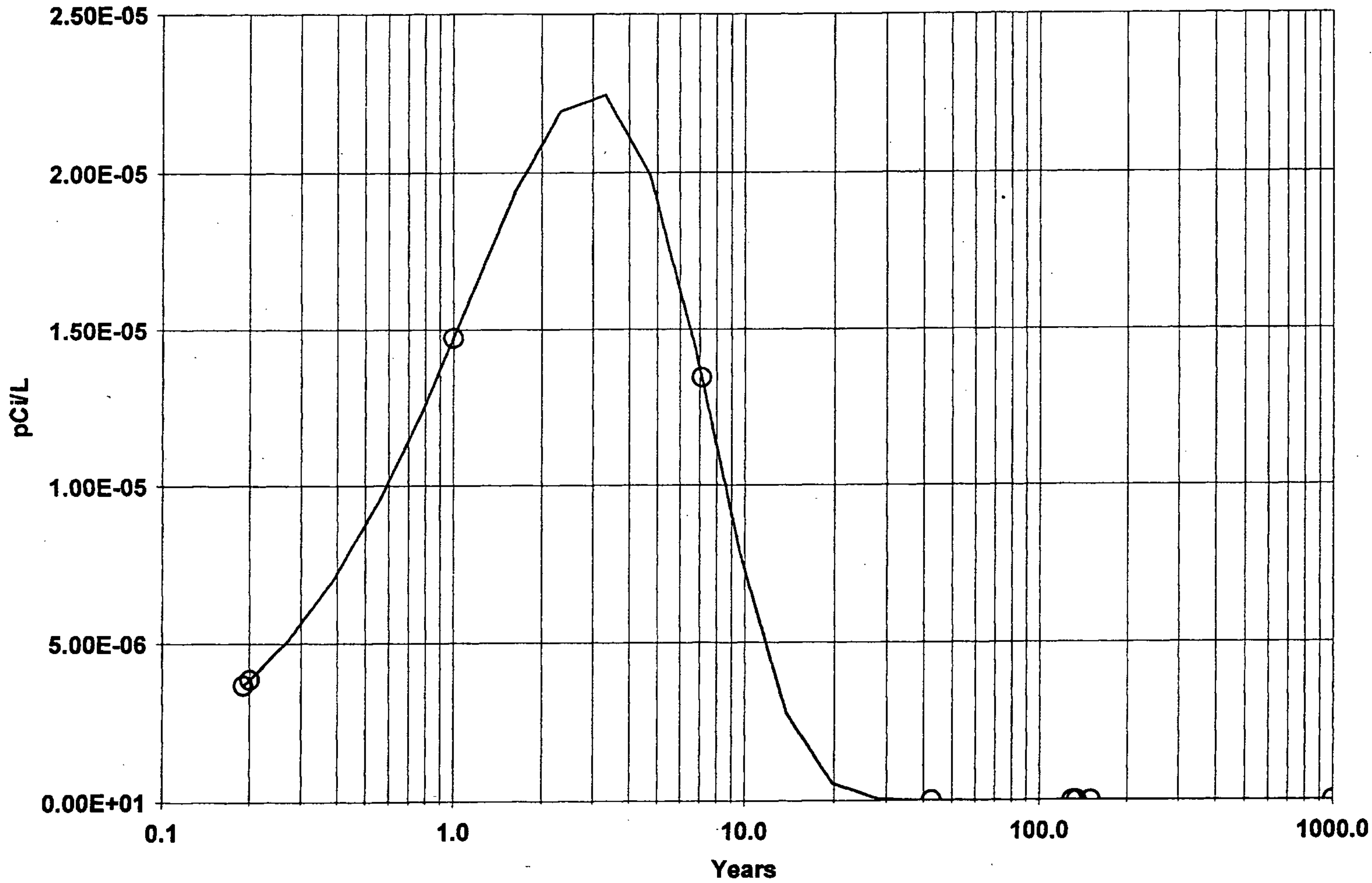
7.00E-04
6.00E-04
5.00E-04
4.00E-04
3.00E-04
2.00E-04
1.00E-04
0.00E+01
0.1
1.0
10.0
100.0
1000.0
Years
buriedpipe092502Co60b405.RAD 09/25/2002 08:20

CONCENTRATION: Cs-134, Well Water



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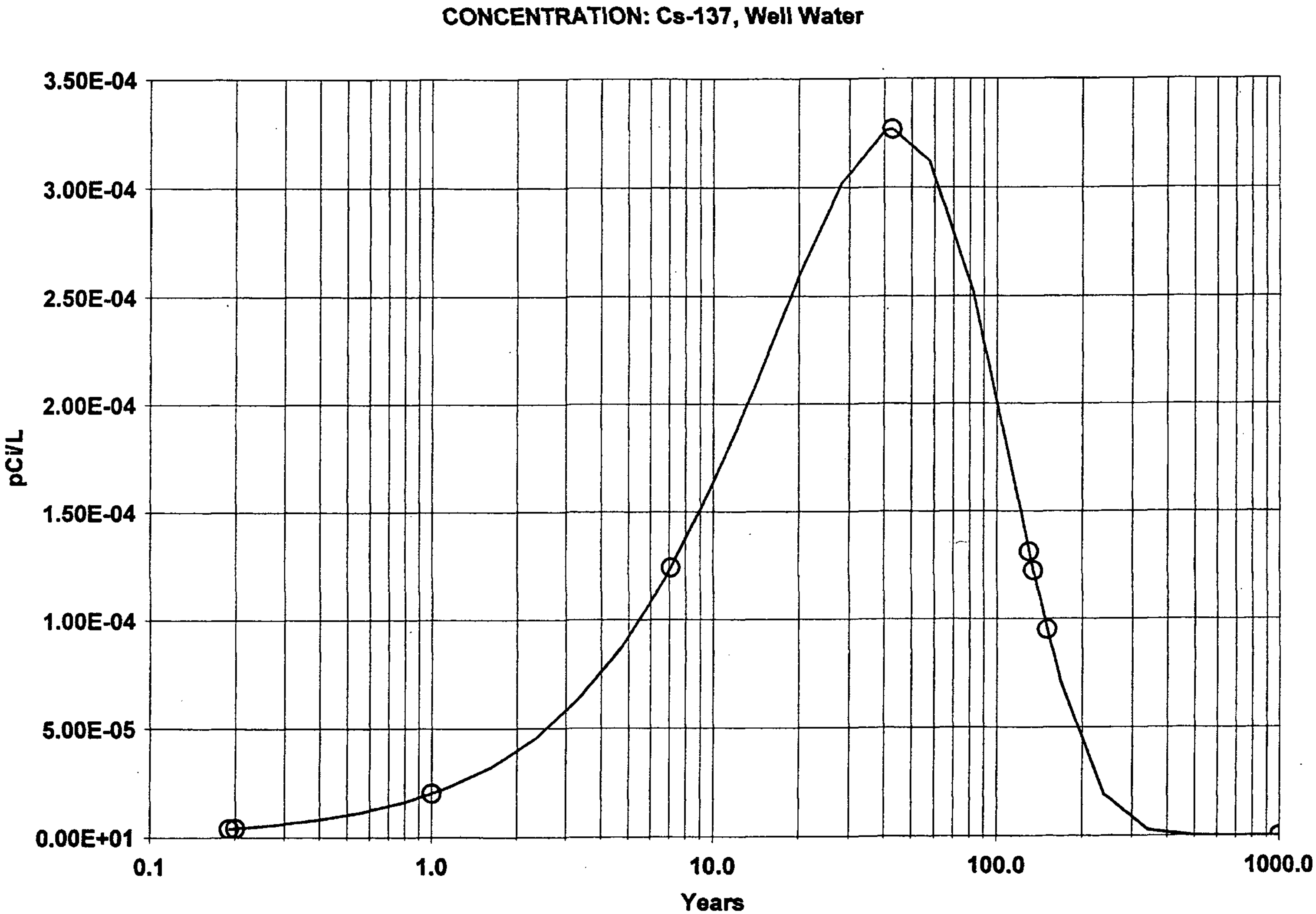
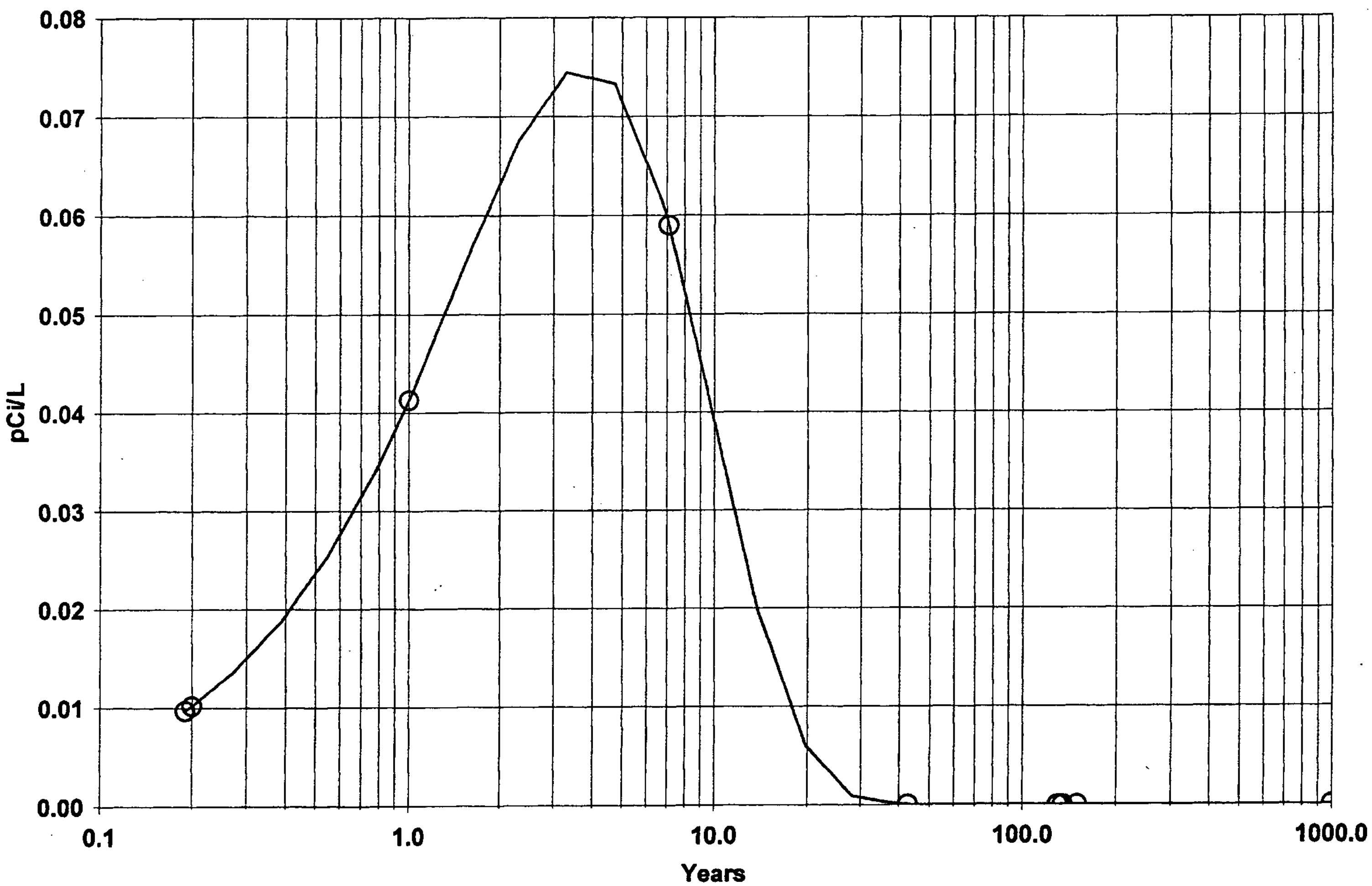
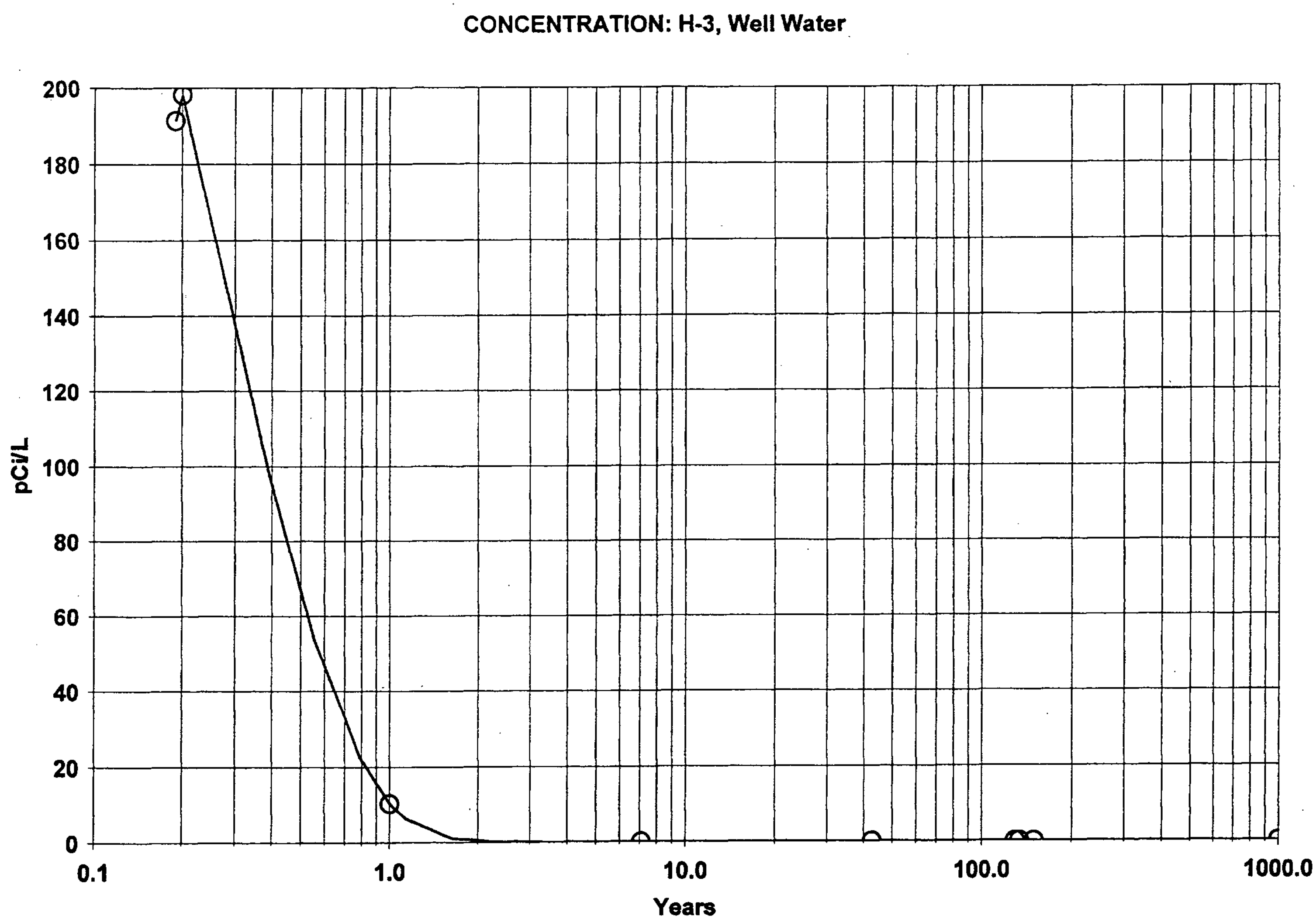


buriedpipe092502Cs137b405.RAD 09/25/2002 08:53

CONCENTRATION: Fe-55, Well Water





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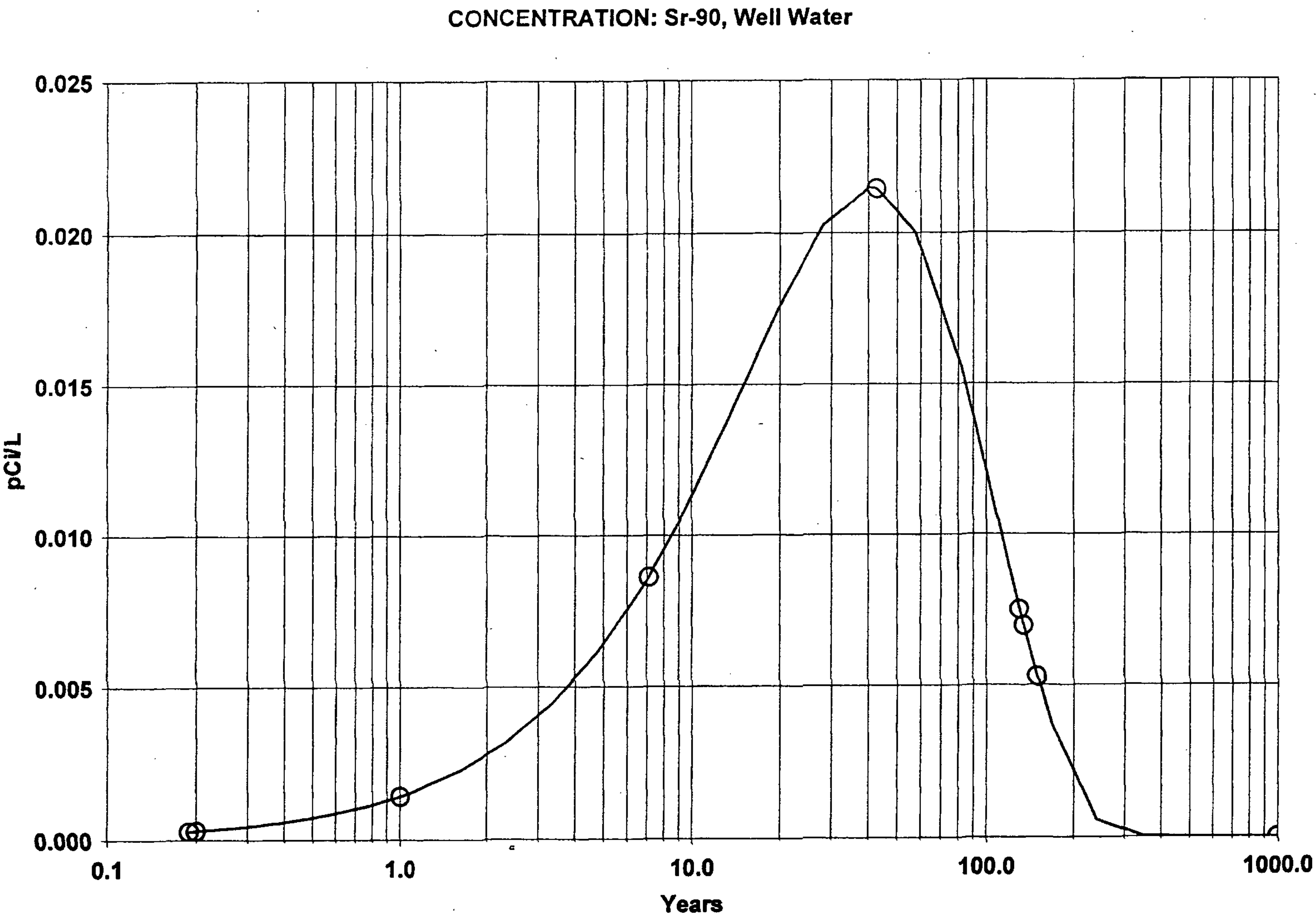


buriedpipe092502Sr90b405.RAD 09/25/2002 09:31

Attachment 6-12
Buried Piping Microshield Output

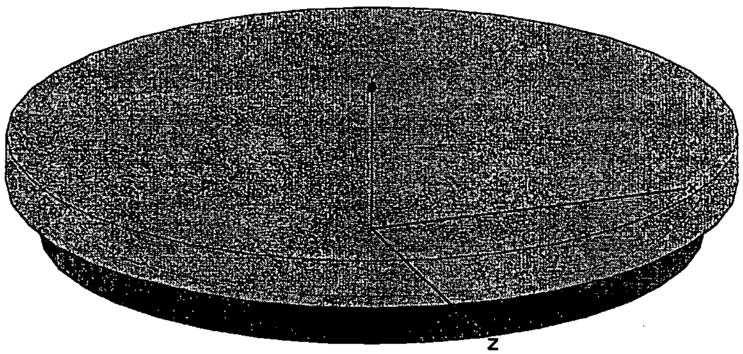
MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power

Page : 1
DOS File: Case1
Run Date: September 25, 2002
Run Time: 10:40:31 AM.
Duration: 00:00:00

File Ref: 017-01
Date: 9-25-02
By: RFD
Checked:

(Illustration of Geometry)

Case Title: Cs-137
Description: Case 1
Geometry: 8 - Cylinder Volume - End Shields



Source Dimensions

Height	100.0 cm	3 ft 3.4 in
Radius	671.0 cm	22 ft 0.2 in

Dose Points

	X	Y	Z
# 1	0 cm	300 cm	0 cm
	0.0 in	9 ft 10.1 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	141.447 m³	SiO2	1.6
Shield 1	1.0 m	SiO2	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	2.2632e-004	8.3738e+006	1.6000e-006	5.9201e-002
Cs-137	2.2632e-004	8.3738e+006	1.6000e-006	5.9201e-002

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.734e+05	1.881e-64	5.623e-29	1.567e-66	4.684e-31

Page : 2
DOS File: Case1
Run Date: September 25, 2002
Run Time: 10:40:31 AM
Duration: 00:00:00

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0322	3.199e+05	1.593e-62	1.082e-28	1.282e-64	8.710e-31
0.0364	1.164e+05	9.643e-49	6.663e-29	5.479e-51	3.786e-31
0.6616	7.535e+06	4.403e-08	2.569e-06	8.535e-11	4.980e-09
TOTALS:	8.144e+06	4.403e-08	2.569e-06	8.535e-11	4.980e-09

MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power
Conversion of calculated exposure in air to dose
FILE: Case1
Case Title: Cs-134
This case was run on Wednesday, September 25, 2002 at 10:33:42 AM
Dose Point # 1 - (0,3,0) m

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	4.865e-007	1.673e-005
Photon Energy Fluence Rate	MeV/cm²/sec	4.899e-007	1.472e-005
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	8.871e-010	2.727e-008
Absorbed Dose Rate in Air	mGy/hr	7.744e-012	2.380e-010
"	mrad/hr	7.744e-010	2.380e-008
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	8.905e-012	2.759e-010
o Opposed	"	7.502e-012	2.291e-010
o Rotational	"	7.502e-012	2.291e-010
o Isotropic	"	6.675e-012	2.033e-010
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	9.488e-012	2.941e-010
o Opposed	"	9.101e-012	2.813e-010
o Rotational	"	9.101e-012	2.813e-010
o Isotropic	"	7.097e-012	2.167e-010
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	7.976e-012	2.463e-010
o Posterior/Anterior	"	7.253e-012	2.221e-010
o Lateral	"	5.651e-012	1.705e-010
o Rotational	"	6.515e-012	1.991e-010
o Isotropic	"	5.695e-012	1.727e-010

MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power
Conversion of calculated exposure in air to dose
FILE: Case1
Case Title: Cs-137
This case was run on Wednesday, September 25, 2002 at 10:40:31 AM
Dose Point # 1 - (0,3,0) m

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	6.654e-008	3.882e-006
Photon Energy Fluence Rate	MeV/cm²/sec	4.403e-008	2.569e-006
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	8.535e-011	4.980e-009
Absorbed Dose Rate in Air	mGy/hr	7.451e-013	4.347e-011
"	mrad/hr	7.451e-011	4.347e-009
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	8.822e-013	5.147e-011
o Opposed	"	7.063e-013	4.121e-011
o Rotational	"	7.063e-013	4.121e-011
o Isotropic	"	6.246e-013	3.644e-011
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	9.376e-013	5.470e-011
o Opposed	"	8.906e-013	5.196e-011
o Rotational	"	8.906e-013	5.196e-011
o Isotropic	"	6.677e-013	3.896e-011
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	7.801e-013	4.551e-011
o Posterior/Anterior	"	6.885e-013	4.017e-011
o Lateral	"	5.106e-013	2.979e-011
o Rotational	"	6.153e-013	3.590e-011
o Isotropic	"	5.238e-013	3.056e-011

MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power
Conversion of calculated exposure in air to dose
FILE: Case1
Case Title: Co-60
This case was run on Wednesday, September 25, 2002 at 10:37:19 AM
Dose Point # 1 - (0,3,0) m

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	7.087e-006	1.378e-004
Photon Energy Fluence Rate	MeV/cm²/sec	9.071e-006	1.753e-004
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	1.588e-008	3.074e-007
Absorbed Dose Rate in Air	mGy/hr	1.386e-010	2.683e-009
"	mrad/hr	1.386e-008	2.683e-007
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.573e-010	3.046e-009
o Opposed	"	1.356e-010	2.624e-009
o Rotational	"	1.356e-010	2.624e-009
o Isotropic	"	1.212e-010	2.345e-009
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	1.674e-010	3.242e-009
o Opposed	"	1.613e-010	3.122e-009
o Rotational	"	1.613e-010	3.122e-009
o Isotropic	"	1.284e-010	2.484e-009
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	1.417e-010	2.743e-009
o Posterior/Anterior	"	1.306e-010	2.527e-009
o Lateral	"	1.040e-010	2.011e-009
o Rotational	"	1.176e-010	2.276e-009
o Isotropic	"	1.042e-010	2.015e-009

MicroShield v5.01 (5.01-00010)
Maine Yankee Atomic Power
Conversion of calculated exposure in air to dose
FILE: Case1
Case Title: Co-57
This case was run on Wednesday, September 25, 2002 at 10:42:32 AM
Dose Point # 1 - (0,3,0) m

Results (Summed over energies)	Units	Without Buildup	With Buildup
Photon Fluence Rate (flux)	Photons/cm²/sec	1.628e-010	9.016e-009
Photon Energy Fluence Rate	MeV/cm²/sec	1.117e-010	6.114e-009
Exposure and Dose Rates:			
Exposure Rate in Air	mR/hr	2.157e-013	1.181e-011
Absorbed Dose Rate in Air	mGy/hr	1.883e-015	1.031e-013
"	mrad/hr	1.883e-013	1.031e-011
Deep Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.223e-015	1.218e-013
o Opposed	"	1.788e-015	9.789e-014
o Rotational	"	1.788e-015	9.789e-014
o Isotropic	"	1.581e-015	8.655e-014
Shallow Dose Equivalent Rate	(ICRP 51 - 1987)		
o Parallel Geometry	mSv/hr	2.366e-015	1.296e-013
o Opposed	"	2.250e-015	1.232e-013
o Rotational	"	2.250e-015	1.232e-013
o Isotropic	"	1.691e-015	9.256e-014
Effective Dose Equivalent Rate	(ICRP 51 - 1987)		
o Anterior/Posterior Geometry	mSv/hr	1.968e-015	1.078e-013
o Posterior/Anterior	"	1.742e-015	9.541e-014
o Lateral	"	1.298e-015	7.102e-014
o Rotational	"	1.557e-015	8.527e-014
o Isotropic	"	1.329e-015	7.272e-014

Attachment 6-13
DCGL/Total Dose Spreadsheets

Table 6-11
Contaminated Material DCGL

Refer to Section 6 for Table 6-11 (Table 6-11a for Containment, Table 6-11b for Non-Containment)

(Containment)
Contaminated Concrete

CONTAMINATED CONCRETE

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	1131.6	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	Gross Beta DCGL	1.80E+04	dpm/100 cm^2
Surface Area/Open Volume	0.46	m^2/m^3	Gross Beta Nuclide Fraction	0.6160	
Concrete Volume	1.13	m^3	Total Inventory	2.92E+04	dpm/100 cm^2

DOSE CALCULATION FACTORS				SOURCE TERM			Kd		WATER, FILL, CONCRETE CONCENTRATION				CONTAMINATED CONCRETE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	2.80E-03	8.19E+01	4.17E+06	6.02E+01	1.00E+00	3.02E+02	1.87E-02	1.13E-03	1.87E-05	Sr-90	1.27E-03	1.22E-04	0.00E+00	1.39E-03
Cs-134	4.39E+00	7.33E-05	6.09E-05	4.55E-03	1.33E+02	6.77E+06	7.91E+01	3.00E+00	3.96E+02	2.31E-02	1.83E-03	6.94E-05	Cs-134	8.11E-04	4.52E-05	1.12E-07	8.56E-04
Cs-137	2.27E+00	5.00E-05	1.20E-05	5.50E-01	1.61E+04	8.20E+08	7.91E+01	3.00E+00	3.96E+02	2.80E+00	2.22E-01	8.40E-03	Cs-137	6.69E-02	2.83E-03	2.66E-06	6.98E-02
Co-60	6.58E+00	2.69E-05	6.30E-04	5.84E-02	1.71E+03	8.70E+07	1.28E+02	1.00E+02	6.40E+02	1.84E-01	2.35E-02	1.84E-02	Co-60	2.37E-03	5.39E-04	1.48E-05	2.92E-03
Co-57	1.67E-01	1.18E-06	2.80E-08	3.06E-04	8.95E+00	4.56E+05	1.28E+02	1.00E+02	6.40E+02	9.66E-04	1.23E-04	9.66E-05	Co-57	5.45E-07	7.17E-08	3.46E-12	6.17E-07
Fe-55	2.50E-03	6.07E-07	0.00E+00	4.81E-03	1.41E+02	7.17E+06	2.50E+01	1.00E+02	1.26E+02	7.69E-02	1.92E-03	7.69E-03	Fe-55	2.23E-05	8.55E-08	0.00E+00	2.24E-05
H-3	2.27E-01	6.40E-08	0.00E+00	2.36E-02	6.88E+02	3.51E+07	0.00E+00	0.00E+00	1.00E+00	4.75E+01	0.00E+00	0.00E+00	H-3	1.45E-03	4.79E-03	0.00E+00	6.25E-03
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.55E-01	1.04E+04	5.29E+08	1.28E+02	1.00E+02	6.40E+02	1.12E+00	1.43E-01	1.12E-01	Ni-63	3.09E-04	5.92E-06	0.00E+00	3.15E-04
													SUM	7.32E-02	8.33E-03	1.76E-05	8.15E-02

SURFACE SOIL

Key Parameters:

Soil Depth	0.15	m
Surface Soil (Cs-137) Concentration	2.39	pCi/g
Surface Soil Total Concentration	2.69	pCi/g

DOSE CALCULATION FACTORS		SOURCE TERM		SURFACE SOIL ANNUAL DOSE	
Nuclide	**NUREG-1727 mrem/y per pCi/g**	**Nuclide Fraction**	**Soil pCi/g**		**Total Dose mrem/y**
Cs-137	2.27E+00	8.90E-01	2.39E+00		5.43E+00
Co-60	6.58E+00	9.00E-03	2.42E-02		1.59E-01
H-3	2.27E-01	5.30E-02	1.43E-01		3.24E-02
Ni-63	1.19E-02	4.80E-02	1.29E-01		1.54E-03
				SUM	5.63E+00

(Containment)
Activated Concrete

ACTIVATED CONCRETE (50 yr Liner Breach and Diffusion)

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	1131.6	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	Activated Concrete Total Inventory	4.88E+08	pCi
Surface Area/Open Volume	0.46	m^2/m^3			
Concrete Volume	1.13	m^3			

DOSE CALCULATION FACTORS				SOURCE TERM		Kd		WATER, FILL, CONCRETE CONCENTRATION				ACTIVATED CONCRETE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Cs-134	4.39E+00	7.33E-05	6.09E-05	5.72E-03	2.79E+06	7.91E+01	3.00E+00	3.96E+02	9.54E-03	7.55E-04	2.86E-05	Cs-134	3.34E-04	1.86E-05	4.60E-08	3.53E-04
Co-60	6.58E+00	2.69E-05	6.30E-04	2.73E-02	1.33E+07	1.28E+02	1.00E+02	6.40E+02	2.82E-02	3.60E-03	2.82E-03	Co-60	3.63E-04	8.25E-05	2.27E-06	4.47E-04
C-14	2.08E+00	2.09E-06	0.00E+00	1.48E-01	7.22E+07	5.00E+00	1.00E+02	2.63E+01	3.72E+00	1.86E-02	3.72E-01	C-14	3.71E-03	3.44E-03	0.00E+00	7.15E-03
Eu-154	3.13E+00	9.55E-06	3.10E-04	6.10E-03	2.98E+06	4.00E+02	5.00E+03	2.02E+03	2.00E-03	8.00E-04	9.99E-03	Eu-154	9.12E-06	2.78E-06	2.48E-07	1.21E-05
Fe-55	2.50E-03	6.07E-07	0.00E+00	8.44E-02	4.12E+07	2.50E+01	1.00E+02	1.26E+02	4.42E-01	1.10E-02	4.42E-02	Fe-55	1.28E-04	4.91E-07	0.00E+00	1.29E-04
H-3	2.27E-01	6.40E-08	0.00E+00	4.41E-01	2.15E+08	0.00E+00	0.00E+00	1.00E+00	2.91E+02	0.00E+00	0.00E+00	H-3	8.92E-03	2.94E-02	0.00E+00	3.83E-02
Eu-152	2.87E+00	6.48E-06	2.09E-04	7.56E-02	3.69E+07	4.00E+02	5.00E+03	2.02E+03	2.48E-02	9.91E-03	1.24E-01	Eu-152	7.67E-05	3.16E-05	2.07E-06	1.10E-04
Ni-63	1.19E-02	5.77E-07	0.00E+00	2.11E-01	1.03E+08	1.28E+02	1.00E+02	6.40E+02	2.18E-01	2.78E-02	2.18E-02	Ni-63	6.01E-05	1.15E-06	0.00E+00	6.13E-05
				9.99E-01	4.88E+08											
												SUM	1.36E-02	3.30E-02	4.63E-06	4.66E-02

Activated Rebar

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	1131.6	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	Activated Concrete Total Inventory	9.01E+08	pCi (1.9*Activated Concrete Inventory)
Surface Area/Open Volume	0.46	m^2/m^3			
Concrete Volume	1.13	m^3			

	DOSE CALCULATION FACTORS			SOURCE TERM		Kd		WATER, FILL, CONCRETE CONCENTRATION				ACTIVATED REBAR ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Cs-134	4.39E+00	7.33E-05	6.09E-05	0.00E+00	0.00E+00	7.91E+01	3.00E+00	3.96E+02	0.00E+00	0.00E+00	0.00E+00	Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	6.58E+00	2.69E-05	6.30E-04	3.00E-02	2.70E+07	1.28E+02	1.00E+02	6.40E+02	5.72E-02	7.31E-03	5.72E-03	Co-60	7.36E-04	1.67E-04	4.61E-06	9.08E-04
C-14	2.08E+00	2.09E-06	0.00E+00	0.00E+00	0.00E+00	5.00E+00	1.00E+02	2.63E+01	0.00E+00	0.00E+00	0.00E+00	C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	3.13E+00	9.55E-06	3.10E-04	0.00E+00	0.00E+00	4.00E+02	5.00E+03	2.02E+03	0.00E+00	0.00E+00	0.00E+00	Eu-154	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	2.50E-03	6.07E-07	0.00E+00	8.18E-04	7.37E+05	2.50E+01	1.00E+02	1.26E+02	7.90E-03	1.98E-04	7.90E-04	Fe-55	2.29E-06	8.78E-09	0.00E+00	2.30E-06
H-3	2.27E-01	6.40E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-152	2.87E+00	6.48E-06	2.09E-04	0.00E+00	0.00E+00	4.00E+02	5.00E+03	2.02E+03	0.00E+00	0.00E+00	0.00E+00	Eu-152	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	9.69E-01	8.73E+08	1.28E+02	1.00E+02	6.40E+02	1.85E+00	2.36E-01	1.85E-01	Ni-63	5.10E-04	9.78E-06	0.00E+00	5.19E-04
												SUM	1.25E-03	1.77E-04	4.61E-06	1.43E-03

DEEP SOIL

Key Parameters:

Porosity	0.3		Surface Soil Depth		0.15	m
Bulk Density	1.6	g/cm^3	Deep Soil (Cs-137) Concentration	(Limited to 2.39)	4.31	pCi/g
Yearly Drinking Water	478	L/y	Deep Soil Total Concentration		4.83	pCi/g
Irrigation Rate	0.274	L/m^2-d				

Using ResRad results for pCi/L per pCi/g conversion Table 1 of EC-018-01 half sand half gravel(.44) 6/18/02

DOSE CALCULATION FACTORS				SOURCE TERM				DEEP SOIL ANNUAL DOSE			
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Deep Soil Inventory pCi/g	Derived Water Conversion Units pCi/L per pCi/g	Water Inventory pCi/L	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Cs-137	2.27E+00	5.00E-05	4.00E-01	8.90E-01	4.30E+00	9.02E-03	3.87E-02	9.26E-04	3.66E-05	1.72E+00	1.72E+00
Co-60	6.58E+00	2.69E-05	2.40E+00	9.00E-03	4.34E-02	2.24E-02	9.74E-04	1.25E-05	2.67E-06	1.04E-01	1.04E-01
H-3	2.27E-01	6.40E-08	0.00E+00	5.30E-02	2.56E-01	6.69E+03	1.71E+03	5.23E-02	1.62E-01	0.00E+00	2.14E-01
Ni-63	1.19E-02	5.77E-07	0.00E+00	4.80E-02	2.32E-01	6.01E-01	1.39E-01	3.84E-05	6.90E-07	0.00E+00	3.91E-05
								5.33E-02	1.62E-01	1.82E+00	2.04E+00

GROUND WATER

Key Parameters:
Annual Water Intake 478 L/y

Dose Calculation Factors		Source Term		Ground Water Annual Dose
Nuclide	FGR 11 mrem/pCi	Nuclide Fraction	Inventory pCi/L	Drinking Water Dose mrem/y
H-3	6.40E-08	1.00E+00	6,812	2.08E-01
				SUM 2.08E-01

SURFACE WATER

Key Parameters:

Annual Water Intake	478	L/y
Annual Fish Consumption	20.6	Kg/y

Dose Calculation factors			Source Term		Surface Water Annual Dose		
Nuclide	FGR 11 mrem/pCi	Bioaccumulation Factor for Fish pCi/Kg per pCi/L	Nuclide Fraction	Water Inventory pCi/L	Drinking Water Dose mrem/y	Fish Ingestion Dose mrem/y	Total Dose mrem/y
H-3	6.40E-08	1.00E+00	1.00E+00	960	2.94E-02	1.27E-03	3.06E-02
				SUM	2.94E-02	1.27E-03	3.06E-02

BURIED PIPING

Key Parameters:

Porosity	0.3	
Bulk Density	1.6	g/cm^3
Yearly Drinking Water	478	L/y
Irrigation Rate	0.274	L/m^2-d
Surface Soil Depth	0.15	m

Buried Pipe Conversion Factor	2.59E-04	pCi/g per dpm/100 cm^2
Gross Beta DCGL	9.80E+03	dpm/100 cm^2
Gross Beta Nuclide Fraction	0.616	
Total Inventory	1.59E+04	dpm/100 cm^2

Dose Calculation Factors				Source Term				Buried Piping Annual Dose			
Nuclide	FGR 11 mrem/pCi	NUREG-1727 mrem/y per pCi/g	Microshield mrem/y per pCi/g	Nuclide Fraction	Water Inventory pCi/L per pCi/g	Pipe Surface Inventory dpm/100cm^2	Soil Inventory pCi/g	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.42E-04	1.47E+01	0.00E+00	2.80E-03	2.15E-02	4.46E+01	1.15E-02	1.68E-05	1.52E-06	0.00E+00	1.83E-05
Cs-134	7.33E-05	4.39E+00	2.21E-05	4.55E-03	2.25E-05	7.23E+01	1.87E-02	1.48E-08	7.71E-10	4.14E-07	4.30E-07
Cs-137	5.00E-05	2.27E+00	3.97E-06	5.50E-01	3.27E-04	8.75E+03	2.27E+00	1.77E-05	7.01E-07	9.01E-06	2.74E-05
Co-60	2.69E-05	6.58E+00	2.53E-04	5.84E-02	8.14E-04	9.29E+02	2.41E-01	2.52E-06	5.37E-07	6.09E-05	6.40E-05
Co-57	1.18E-06	1.67E-01	9.44E-09	3.06E-04	1.15E-04	4.88E+00	1.26E-03	8.18E-11	1.01E-11	1.19E-11	1.04E-10
Fe-55	6.07E-07	2.50E-03	0.00E+00	4.81E-03	4.30E-05	7.66E+01	1.98E-02	2.47E-10	8.89E-13	0.00E+00	2.48E-10
H-3	6.40E-08	2.27E-01	0.00E+00	2.36E-02	1.98E+02	3.75E+02	9.70E-02	5.88E-04	1.82E-03	0.00E+00	2.41E-03
Ni-63	5.77E-07	1.19E-02	0.00E+00	3.55E-01	2.09E-02	5.65E+03	1.46E+00	8.42E-06	1.51E-07	0.00E+00	8.57E-06
							SUM	6.33E-04	1.82E-03	7.03E-05	2.52E-03

BOP EMBEDDED PIPE

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Surface Soil Depth	0.15	m
Yearly Drinking Water	478.0	l/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	1131.6	m^2	Annual Total Well Water Vol	738	m^3
Fill Volume	2460.0	m^3	Embedded Pipe Conversion Factor	5754.5	pCi per dpm/100 cm^2
Surface Area/Open Volume	0.46	m^2/m^3	Gross Beta DCGL	1.00E+05	dpm/100 cm^2
Concrete Volume	1.13	m^3	Gross Beta Nuclide Fraction	0.616	
			Total Inventory	1.62E+05	dpm/100 cm^2

	DOSE CALCULATION FACTORS			SOURCE TERM			Kd	
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm
Sr-90	1.47E+01	1.42E-04	0.00E+00	2.80E-03	4.55E+02	2.62E+06	6.02E+01	1.00E+00
Cs-134	4.39E+00	7.33E-05	6.09E-05	4.55E-03	7.38E+02	4.25E+06	7.91E+01	3.00E+00
Cs-137	2.27E+00	5.00E-05	1.20E-05	5.50E-01	8.93E+04	5.14E+08	7.91E+01	3.00E+00
Co-60	6.58E+00	2.69E-05	6.30E-04	5.84E-02	9.48E+03	5.46E+07	1.28E+02	1.00E+02
Co-57	1.67E-01	1.18E-06	2.80E-08	3.06E-04	4.97E+01	2.86E+05	1.28E+02	1.00E+02
Fe-55	2.50E-03	6.07E-07	0.00E+00	4.81E-03	7.82E+02	4.50E+06	2.50E+01	1.00E+02
H-3	2.27E-01	6.40E-08	0.00E+00	2.36E-02	3.82E+03	2.20E+07	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.55E-01	5.77E+04	3.32E+08	1.28E+02	1.00E+02

	WATER, FILL, CONCRETE CONCENTRATION				EMBEDDED PIPE ANNUAL DOSE				
Nuclide	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	3.02E+02	1.18E-02	7.07E-04	1.18E-05	Sr-90	7.98E-04	7.68E-05	0.00E+00	8.75E-04
Cs-134	3.96E+02	1.45E-02	1.15E-03	4.36E-05	Cs-134	5.09E-04	2.83E-05	6.99E-08	5.37E-04
Cs-137	3.96E+02	1.76E+00	1.39E-01	5.27E-03	Cs-137	4.20E-02	1.77E-03	1.67E-06	4.38E-02
Co-60	6.40E+02	1.16E-01	1.48E-02	1.16E-02	Co-60	1.49E-03	3.38E-04	9.30E-06	1.83E-03
Co-57	6.40E+02	6.06E-04	7.74E-05	6.06E-05	Co-57	3.42E-07	4.50E-08	2.17E-12	3.87E-07
Fe-55	1.26E+02	4.82E-02	1.21E-03	4.82E-03	Fe-55	1.40E-05	5.36E-08	0.00E+00	1.40E-05
H-3	1.00E+00	2.98E+01	0.00E+00	0.00E+00	H-3	9.12E-04	3.01E-03	0.00E+00	3.92E-03
Ni-63	6.40E+02	7.02E-01	8.97E-02	7.02E-02	Ni-63	1.94E-04	3.72E-06	0.00E+00	1.97E-04
					SUM	4.59E-02	5.23E-03	1.10E-05	5.11E-02

(Containment)
Spray Building Embedded Pump Piping

EMBEDDED SPRAY PUMP PIPING

Key Parameters:

Porosity	0.30	
Bulk Density	1.50	g/cm^3
Yearly Drinking Water	478.0	l/yr
Wall Surface Area	1131.6	m^2
Fill Volume	2460.0	m^3
Surface Area/Open Volume	0.46	m^2/m^3
Concrete Volume	1.13	m^3

Concrete Density	2.20	g/cm^3
Surface Soil Depth	0.15	m
Irrigation Rate	0.274	L/m^2-d
Annual Total Well Water Vol	738	m^3
Embedded Pipe Conversion Factor	1191.7	pCi per dpm/100 cm^2
Gross Beta DCGL	8.00E+05	dpm/100 cm^2
Gross Beta Nuclide Fraction	0.616	
Total Inventory	1.30E+06	dpm/100 cm^2

	DOSE CALCULATION FACTORS			SOURCE TERM			Kd		WATER, FILL, CONCRETE CONCENTRATION				EMBEDDED PIPE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	2.80E-03	3.64E+03	4.34E+06	6.02E+01	1.00E+00	3.02E+02	1.95E-02	1.17E-03	1.95E-05	Sr-90	1.32E-03	1.27E-04	0.00E+00	1.45E-03
Cs-134	4.39E+00	7.33E-05	6.09E-05	4.55E-03	5.91E+03	7.04E+06	7.91E+01	3.00E+00	3.96E+02	2.41E-02	1.90E-03	7.22E-05	Cs-134	8.43E-04	4.69E-05	1.16E-07	8.90E-04
Cs-137	2.27E+00	5.00E-05	1.20E-05	5.50E-01	7.15E+05	8.52E+08	7.91E+01	3.00E+00	3.96E+02	2.91E+00	2.30E-01	8.73E-03	Cs-137	6.96E-02	2.94E-03	2.76E-06	7.25E-02
Co-60	6.58E+00	2.69E-05	6.30E-04	5.84E-02	7.59E+04	9.04E+07	1.28E+02	1.00E+02	6.40E+02	1.91E-01	2.45E-02	1.91E-02	Co-60	2.46E-03	5.60E-04	1.54E-05	3.04E-03
Co-57	1.67E-01	1.18E-06	2.80E-08	3.06E-04	3.98E+02	4.74E+05	1.28E+02	1.00E+02	6.40E+02	1.00E-03	1.28E-04	1.00E-04	Co-57	5.66E-07	7.45E-08	3.59E-12	6.41E-07
Fe-55	2.50E-03	6.07E-07	0.00E+00	4.81E-03	6.25E+03	7.45E+06	2.50E+01	1.00E+02	1.26E+02	7.99E-02	2.00E-03	7.99E-03	Fe-55	2.32E-05	8.88E-08	0.00E+00	2.33E-05
H-3	2.27E-01	6.40E-08	0.00E+00	2.36E-02	3.06E+04	3.64E+07	0.00E+00	0.00E+00	1.00E+00	4.94E+01	0.00E+00	0.00E+00	H-3	1.51E-03	4.98E-03	0.00E+00	6.49E-03
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.55E-01	4.61E+05	5.50E+08	1.28E+02	1.00E+02	6.40E+02	1.16E+00	1.49E-01	1.16E-01	Ni-63	3.21E-04	6.16E-06	0.00E+00	3.27E-04
					1.30E+06	1.55E+09											
													SUM	7.60E-02	8.66E-03	1.83E-05	8.47E-02

CONTAMINATED CONCRETE SPECIAL AREAS

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	1131.6	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	**Special Areas Gross Beta DCGL**	**9.50E+03**	dpm/100 cm^2
Surface Area/Open Volume	0.46	m^2/m^3	Gross Beta Nuclide Fraction	0.6672	
Concrete Volume	1.13	m^3	Total Inventory	1.42E+04	dpm/100 cm^2

8.15E-02 **<== Total Dose Contaminated Concrete**

2.92E-02 **<== Total Dose Special Areas**

DOSE CALCULATION FACTORS				SOURCE TERM			Kd		WATER, FILL, CONCRETE CONCENTRATION				CONTAMINATED CONCRETE ANNUAL DOSE				
Nuclide	**NUREG-1727 mrem/y per pCi/g**	**FGR 11 mrem/pCi**	**Microshield mrem/y per pCi/g**	**Nuclide Fraction**	**Inventory dpm/100 cm^2**	**Inventory pCi**	**Kd Fill cm^3/gm**	**Kd Concrete cm^3/gm**	**Adsorption Factor**	**Water pCi/L**	**Fill pCi/g**	**Concrete pCi/g**	**Nuclide**	**Drinking Water Dose mrem/y**	**Irrigation Dose mrem/y**	**Direct Dose mrem/y**	**Total Dose mrem/y**
Sr-90	1.47E+01	1.42E-04	0.00E+00	6.874E-03	9.79E+01	4.99E+06	6.12E+01	1.00E+00	3.07E+02	2.20E-02	1.35E-03	2.20E-05	Sr-90	1.50E-03	1.44E-04	0.00E+00	1.64E-03
Sb-125	9.77E-01	2.81E-06	3.83E-06	4.523E-03	6.44E+01	3.28E+06	4.50E+01	0.00E+00	2.26E+02	1.97E-02	8.86E-04	0.00E+00	Sb-125	2.64E-05	8.55E-06	3.39E-09	3.50E-05
Cs-134	4.39E+00	7.33E-05	6.09E-05	2.815E-03	4.01E+01	2.04E+06	7.91E+01	3.00E+00	3.96E+02	6.98E-03	5.52E-04	2.09E-05	Cs-134	2.45E-04	1.36E-05	3.36E-08	2.58E-04
Cs-137	2.27E+00	5.00E-05	1.20E-05	2.890E-01	4.12E+03	2.10E+08	7.91E+01	3.00E+00	3.96E+02	7.17E-01	5.67E-02	2.15E-03	Cs-137	1.71E-02	7.23E-04	6.80E-07	1.79E-02
Pu-238	1.00E+01	3.20E-03	2.45E-25	1.165E-04	1.66E+00	8.46E+04	5.50E+02	5.00E+03	2.77E+03	4.14E-05	2.28E-05	2.07E-04	Pu-238	6.33E-05	1.84E-07	5.58E-30	6.35E-05
Pu-239	1.09E+01	3.54E-03	6.10E-15	8.752E-05	1.25E+00	6.35E+04	5.50E+02	5.00E+03	2.77E+03	3.11E-05	1.71E-05	1.55E-04	Pu-239	5.26E-05	1.51E-07	1.04E-19	5.28E-05
Pu-240	1.09E+01	3.54E-03	7.52E-26	8.750E-05	1.25E+00	6.35E+04	5.50E+02	5.00E+03	2.77E+03	3.11E-05	1.71E-05	1.55E-04	Pu-240	5.26E-05	1.51E-07	1.29E-30	5.28E-05
Pu-241	3.47E-01	6.85E-05	0.00E+00	6.705E-03	9.55E+01	4.87E+06	5.50E+02	5.00E+03	2.77E+03	2.38E-03	1.31E-03	1.19E-02	Pu-241	7.80E-05	3.67E-07	0.00E+00	7.84E-05
Am-241	1.19E+01	3.64E-03	1.65E-19	5.929E-04	8.44E+00	4.30E+05	1.90E+03	5.00E+03	9.51E+03	6.13E-05	1.16E-04	3.06E-04	Am-241	1.07E-04	3.24E-07	1.93E-23	1.07E-04
Cm-243	7.81E+00	2.51E-03	1.27E-08	4.649E-05	6.62E-01	3.37E+04	4.00E+03	5.00E+03	2.00E+04	2.28E-06	9.14E-06	1.14E-05	Cm-243	2.74E-06	7.93E-09	1.16E-13	2.75E-06
Cm-244	6.00E+00	2.02E-03	9.81E-25	4.454E-05	6.34E-01	3.23E+04	4.00E+03	5.00E+03	2.00E+04	2.19E-06	8.75E-06	1.09E-05	Cm-244	2.11E-06	5.84E-09	8.59E-30	2.12E-06
Co-60	6.58E+00	2.69E-05	6.30E-04	3.639E-01	5.18E+03	2.64E+08	1.28E+02	1.00E+02	6.40E+02	5.59E-01	7.14E-02	5.59E-02	Co-60	7.19E-03	1.64E-03	4.50E-05	8.87E-03
Co-57	1.67E-01	1.18E-06	2.80E-08		0.00E+00	0.00E+00	1.28E+02	1.00E+02	6.40E+02	0.00E+00	0.00E+00	0.00E+00	Co-57	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	1.67E+00	2.77E-06	4.40E-05	4.028E-04	5.74E+00	2.92E+05	5.00E+01	0.00E+00	2.51E+02	1.58E-03	7.89E-05	0.00E+00	Mn-54	2.09E-06	1.17E-06	3.47E-09	3.26E-06
Fe-55	2.50E-03	6.07E-07	0.00E+00	2.235E-02	3.18E+02	1.62E+07	2.50E+01	1.00E+02	1.26E+02	1.74E-01	4.35E-03	1.74E-02	Fe-55	5.05E-05	1.93E-07	0.00E+00	5.07E-05
H-3	2.27E-01	6.40E-08	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.024E-01	4.31E+03	2.19E+08	1.28E+02	1.00E+02	6.40E+02	4.65E-01	5.94E-02	4.65E-02	Ni-63	1.28E-04	2.46E-06	0.00E+00	1.31E-04
													SUM	2.66E-02	2.53E-03	4.57E-05	2.92E-02

(Non-Containment)
Contaminated Concrete

CONTAMINATED CONCRETE

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	4182.0	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	Gross Beta DCGL	1.80E+04	dpm/100 cm^2
Surface Area/Open Volume	1.70	m^2/m^3	Gross Beta Nuclide Fraction	0.6160	
Concrete Volume	4.18	m^3	Total Inventory	2.92E+04	dpm/100 cm^2

DOSE CALCULATION FACTORS				SOURCE TERM			Kd		WATER, FILL, CONCRETE CONCENTRATION				CONTAMINATED CONCRETE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	2.80E-03	8.19E+01	1.54E+07	6.02E+01	1.00E+00	3.01E+02	6.93E-02	4.17E-03	6.93E-05	Sr-90	4.70E-03	4.52E-04	0.00E+00	5.15E-03
Cs-134	4.39E+00	7.33E-05	6.09E-05	4.55E-03	1.33E+02	2.50E+07	7.91E+01	3.00E+00	3.96E+02	8.55E-02	6.77E-03	2.57E-04	Cs-134	3.00E-03	1.67E-04	4.12E-07	3.16E-03
Cs-137	2.27E+00	5.00E-05	1.20E-05	5.50E-01	1.61E+04	3.03E+09	7.91E+01	3.00E+00	3.96E+02	1.03E+01	8.19E-01	3.10E-02	Cs-137	2.47E-01	1.04E-02	9.82E-06	2.58E-01
Co-60	6.58E+00	2.69E-05	6.30E-04	5.84E-02	1.71E+03	3.22E+08	1.28E+02	1.00E+02	6.40E+02	6.80E-01	8.68E-02	6.80E-02	Co-60	8.74E-03	1.99E-03	5.47E-05	1.08E-02
Co-57	1.67E-01	1.18E-06	2.80E-08	3.06E-04	8.95E+00	1.69E+06	1.28E+02	1.00E+02	6.40E+02	3.57E-03	4.56E-04	3.57E-04	Co-57	2.01E-06	2.65E-07	1.28E-11	2.28E-06
Fe-55	2.50E-03	6.07E-07	0.00E+00	4.81E-03	1.41E+02	2.65E+07	2.50E+01	1.00E+02	1.27E+02	2.82E-01	7.05E-03	2.82E-02	Fe-55	8.19E-05	3.14E-07	0.00E+00	8.22E-05
H-3	2.27E-01	6.40E-08	0.00E+00	2.36E-02	6.88E+02	1.30E+08	0.00E+00	0.00E+00	1.00E+00	1.75E+02	0.00E+00	0.00E+00	H-3	5.36E-03	1.77E-02	0.00E+00	2.31E-02
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.55E-01	1.04E+04	1.96E+09	1.28E+02	1.00E+02	6.40E+02	4.13E+00	5.28E-01	4.13E-01	Ni-63	1.14E-03	2.19E-05	0.00E+00	1.16E-03
													SUM	2.70E-01	3.08E-02	6.49E-05	3.01E-01

SURFACE SOIL

Key Parameters:

Soil Depth	0.15	m
Surface Soil (Cs-137) Concentration	2.39	pCi/g
Surface Soil Total Concentration	2.69	pCi/g

DOSE CALCULATION FACTORS		SOURCE TERM		SURFACE SOIL ANNUAL DOSE
Nuclide	**NUREG-1727 mrem/y per pCi/g**	**Nuclide Fraction**	**Soil pCi/g**	**Total Dose mrem/y**
Cs-137	2.27E+00	8.90E-01	2.39E+00	5.43E+00
Co-60	6.58E+00	9.00E-03	2.42E-02	1.59E-01
H-3	2.27E-01	5.30E-02	1.43E-01	3.24E-02
Ni-63	1.19E-02	4.80E-02	1.29E-01	1.54E-03
				SUM 5.63E+00

ACTIVATED CONCRETE (50 yr Liner Breach and Diffusion)

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	4182.0	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	Activated Concrete Total Inventory	0.00E+00	pCi
Surface Area/Open Volume	1.70	m^2/m^3			
Concrete Volume	4.18	m^3			

DOSE CALCULATION FACTORS				SOURCE TERM		Kd		WATER, FILL, CONCRETE CONCENTRATION				ACTIVATED CONCRETE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Cs-134	4.39E+00	7.33E-05	6.09E-05	5.72E-03	0.00E+00	7.91E+01	3.00E+00	3.96E+02	0.00E+00	0.00E+00	0.00E+00	Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	6.58E+00	2.69E-05	6.30E-04	2.73E-02	0.00E+00	1.28E+02	1.00E+02	6.40E+02	0.00E+00	0.00E+00	0.00E+00	Co-60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	2.08E+00	2.09E-06	0.00E+00	1.48E-01	0.00E+00	5.00E+00	1.00E+02	2.72E+01	0.00E+00	0.00E+00	0.00E+00	C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	3.13E+00	9.55E-06	3.10E-04	6.10E-03	0.00E+00	4.00E+02	5.00E+03	2.06E+03	0.00E+00	0.00E+00	0.00E+00	Eu-154	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	2.50E-03	6.07E-07	0.00E+00	8.44E-02	0.00E+00	2.50E+01	1.00E+02	1.27E+02	0.00E+00	0.00E+00	0.00E+00	Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00
H-3	2.27E-01	6.40E-08	0.00E+00	4.41E-01	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-152	2.87E+00	6.48E-06	2.09E-04	7.56E-02	0.00E+00	4.00E+02	5.00E+03	2.06E+03	0.00E+00	0.00E+00	0.00E+00	Eu-152	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	2.11E-01	0.00E+00	1.28E+02	1.00E+02	6.40E+02	0.00E+00	0.00E+00	0.00E+00	Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
				9.99E-01	0.00E+00											
												SUM	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Activated Rebar

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	4182.0	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	Activated Concrete Total Inventory	0.00E+00	pCi (1.9*Activated Concrete Inventory)
Surface Area/Open Volume	1.70	m^2/m^3			
Concrete Volume	4.18	m^3			

DOSE CALCULATION FACTORS				SOURCE TERM		Kd		WATER, FILL, CONCRETE CONCENTRATION				ACTIVATED REBAR ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Cs-134	4.39E+00	7.33E-05	6.09E-05	0.00E+00	0.00E+00	7.91E+01	3.00E+00	3.96E+02	0.00E+00	0.00E+00	0.00E+00	Cs-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	6.58E+00	2.69E-05	6.30E-04	3.00E-02	0.00E+00	1.28E+02	1.00E+02	6.40E+02	0.00E+00	0.00E+00	0.00E+00	Co-60	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C-14	2.08E+00	2.09E-06	0.00E+00	0.00E+00	0.00E+00	5.00E+00	1.00E+02	2.72E+01	0.00E+00	0.00E+00	0.00E+00	C-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	3.13E+00	9.55E-06	3.10E-04	0.00E+00	0.00E+00	4.00E+02	5.00E+03	2.06E+03	0.00E+00	0.00E+00	0.00E+00	Eu-154	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	2.50E-03	6.07E-07	0.00E+00	8.18E-04	0.00E+00	2.50E+01	1.00E+02	1.27E+02	0.00E+00	0.00E+00	0.00E+00	Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00
H-3	2.27E-01	6.40E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-152	2.87E+00	6.48E-06	2.09E-04	0.00E+00	0.00E+00	4.00E+02	5.00E+03	2.06E+03	0.00E+00	0.00E+00	0.00E+00	Eu-152	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	9.69E-01	0.00E+00	1.28E+02	1.00E+02	6.40E+02	0.00E+00	0.00E+00	0.00E+00	Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00
												SUM	0.00E+00	0.00E+00	0.00E+00	0.00E+00

DEEP SOIL

Key Parameters:

Porosity	0.3		Surface Soil Depth		0.15	m
Bulk Density	1.6	g/cm^3	Deep Soil (Cs-137) Concentration	(Limited to 2.39)	4.31	pCi/g
Yearly Drinking Water	478	L/y	Deep Soil Total Concentration		4.83	pCi/g
Irrigation Rate	0.274	L/m^2-d				

Using ResRad results for pCi/L per pCi/g conversion Table 1 of EC-018-01 half sand half gravel(.44) 6/18/02

DOSE CALCULATION FACTORS				SOURCE TERM				DEEP SOIL ANNUAL DOSE			
Nuclide	**NUREG-1727 mrem/y per pCi/g**	**FGR 11 mrem/pCi**	**Microshield mrem/y per pCi/g**	**Nuclide Fraction**	**Deep Soil Inventory pCi/g**	**Derived Water Conversion Units pCi/L per pCi/g**	**Water Inventory pCi/L**	**Drinking Water Dose mrem/y**	**Irrigation Dose mrem/y**	**Direct Dose mrem/y**	**Total Dose mrem/y**
Cs-137	2.27E+00	5.00E-05	4.00E-01	8.90E-01	4.30E+00	9.02E-03	3.87E-02	9.26E-04	3.66E-05	1.72E+00	1.72E+00
Co-60	6.58E+00	2.69E-05	2.40E+00	9.00E-03	4.34E-02	2.24E-02	9.74E-04	1.25E-05	2.67E-06	1.04E-01	1.04E-01
H-3	2.27E-01	6.40E-08	0.00E+00	5.30E-02	2.56E-01	6.69E+03	1.71E+03	5.23E-02	1.62E-01	0.00E+00	2.14E-01
Ni-63	1.19E-02	5.77E-07	0.00E+00	4.80E-02	2.32E-01	6.01E-01	1.39E-01	3.84E-05	6.90E-07	0.00E+00	3.91E-05
								5.33E-02	1.62E-01	1.82E+00	2.04E+00

GROUND WATER

Key Parameters:
Annual Water Intake 478 L/y

Dose Calculation Factors		Source Term		Ground Water Annual Dose	
Nuclide	FGR 11 mrem/pCi	Nuclide Fraction	Inventory pCi/L		Drinking Water Dose mrem/y
H-3	6.40E-08	1.00E+00	6,812		2.08E-01
				SUM	2.08E-01

SURFACE WATER

Key Parameters:

Annual Water Intake	478	L/y
Annual Fish Consumption	20.6	Kg/y

Dose Calculation factors			Source Term		Surface Water Annual Dose		
Nuclide	FGR 11 mrem/pCi	Bioaccumulation Factor for Fish pCi/Kg per pCi/L	Nuclide Fraction	Water Inventory pCi/L	Drinking Water Dose mrem/y	Fish Ingestion Dose mrem/y	Total Dose mrem/y
H-3	6.40E-08	1.00E+00	1.00E+00	960	2.94E-02	1.27E-03	3.06E-02
				SUM	2.94E-02	1.27E-03	3.06E-02

BURIED PIPING

Key Parameters:

Porosity	0.3	
Bulk Density	1.6	g/cm^3
Yearly Drinking Water	478	L/y
Irrigation Rate	0.274	L/m^2-d
Surface Soil Depth	0.15	m

Buried Pipe Conversion Factor	2.59E-04	pCi/g per dpm/100 cm^2
Gross Beta DCGL	9.80E+03	dpm/100 cm^2
Gross Beta Nuclide Fraction	0.616	
Total Inventory	1.59E+04	dpm/100 cm^2

Dose Calculation Factors				Source Term				Buried Piping Annual Dose			
Nuclide	FGR 11 mrem/pCi	NUREG-1727 mrem/y per pCi/g	Microshield mrem/y per pCi/g	Nuclide Fraction	Water Inventory pCi/L per pCi/g	Pipe Surface Inventory dpm/100cm^2	Soil Inventory pCi/g	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.42E-04	1.47E+01	0.00E+00	2.80E-03	2.15E-02	4.46E+01	1.15E-02	1.68E-05	1.52E-06	0.00E+00	1.83E-05
Cs-134	7.33E-05	4.39E+00	2.21E-05	4.55E-03	2.25E-05	7.23E+01	1.87E-02	1.48E-08	7.71E-10	4.14E-07	4.30E-07
Cs-137	5.00E-05	2.27E+00	3.97E-06	5.50E-01	3.27E-04	8.75E+03	2.27E+00	1.77E-05	7.01E-07	9.01E-06	2.74E-05
Co-60	2.69E-05	6.58E+00	2.53E-04	5.84E-02	8.14E-04	9.29E+02	2.41E-01	2.52E-06	5.37E-07	6.09E-05	6.40E-05
Co-57	1.18E-06	1.67E-01	9.44E-09	3.06E-04	1.15E-04	4.88E+00	1.26E-03	8.18E-11	1.01E-11	1.19E-11	1.04E-10
Fe-55	6.07E-07	2.50E-03	0.00E+00	4.81E-03	4.30E-05	7.66E+01	1.98E-02	2.47E-10	8.89E-13	0.00E+00	2.48E-10
H-3	6.40E-08	2.27E-01	0.00E+00	2.36E-02	1.98E+02	3.75E+02	9.70E-02	5.88E-04	1.82E-03	0.00E+00	2.41E-03
Ni-63	5.77E-07	1.19E-02	0.00E+00	3.55E-01	2.09E-02	5.65E+03	1.46E+00	8.42E-06	1.51E-07	0.00E+00	8.57E-06
							SUM	6.33E-04	1.82E-03	7.03E-05	2.52E-03

BOP EMBEDDED PIPE

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Surface Soil Depth	0.15	m
Yearly Drinking Water	478.0	l/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	4182.0	m^2	Annual Total Well Water Vol	738	m^3
Fill Volume	2460.0	m^3	Embedded Pipe Conversion Factor	5754.5	pCi per dpm/100 cm^2
Surface Area/Open Volume	1.70	m^2/m^3	Gross Beta DCGL	1.00E+05	dpm/100 cm^2
Concrete Volume	4.18	m^3	Gross Beta Nuclide Fraction	0.616	
			Total Inventory	1.62E+05	dpm/100 cm^2

	DOSE CALCULATION FACTORS			SOURCE TERM			Kd	
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm
Sr-90	1.47E+01	1.42E-04	0.00E+00	2.80E-03	4.55E+02	2.62E+06	6.02E+01	1.00E+00
Cs-134	4.39E+00	7.33E-05	6.09E-05	4.55E-03	7.38E+02	4.25E+06	7.91E+01	3.00E+00
Cs-137	2.27E+00	5.00E-05	1.20E-05	5.50E-01	8.93E+04	5.14E+08	7.91E+01	3.00E+00
Co-60	6.58E+00	2.69E-05	6.30E-04	5.84E-02	9.48E+03	5.46E+07	1.28E+02	1.00E+02
Co-57	1.67E-01	1.18E-06	2.80E-08	3.06E-04	4.97E+01	2.86E+05	1.28E+02	1.00E+02
Fe-55	2.50E-03	6.07E-07	0.00E+00	4.81E-03	7.82E+02	4.50E+06	2.50E+01	1.00E+02
H-3	2.27E-01	6.40E-08	0.00E+00	2.36E-02	3.82E+03	2.20E+07	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.55E-01	5.77E+04	3.32E+08	1.28E+02	1.00E+02

WATER, FILL, CONCRETE CONCENTRATION				EMBEDDED PIPE ANNUAL DOSE				
Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
3.01E+02	1.18E-02	7.07E-04	1.18E-05	Sr-90	7.98E-04	7.68E-05	0.00E+00	8.75E-04
3.96E+02	1.45E-02	1.15E-03	4.36E-05	Cs-134	5.09E-04	2.83E-05	6.99E-08	5.37E-04
3.96E+02	1.76E+00	1.39E-01	5.27E-03	Cs-137	4.20E-02	1.77E-03	1.67E-06	4.38E-02
6.40E+02	1.15E-01	1.47E-02	1.15E-02	Co-60	1.48E-03	3.37E-04	9.29E-06	1.83E-03
6.40E+02	6.05E-04	7.73E-05	6.05E-05	Co-57	3.41E-07	4.49E-08	2.16E-12	3.86E-07
1.27E+02	4.79E-02	1.20E-03	4.79E-03	Fe-55	1.39E-05	5.32E-08	0.00E+00	1.39E-05
1.00E+00	2.98E+01	0.00E+00	0.00E+00	H-3	9.10E-04	3.00E-03	0.00E+00	3.91E-03
6.40E+02	7.01E-01	8.96E-02	7.01E-02	Ni-63	1.93E-04	3.71E-06	0.00E+00	1.97E-04
				SUM	4.59E-02	5.22E-03	1.10E-05	5.11E-02

(Non-Containment)
Spray Building Embedded Pump Piping

EMBEDDED SPRAY PUMP PIPING

Key Parameters:

Porosity	0.30	
Bulk Density	1.50	g/cm^3
Yearly Drinking Water	478.0	l/yr
Wall Surface Area	4182.0	m^2
Fill Volume	2460.0	m^3
Surface Area/Open Volume	1.70	m^2/m^3
Concrete Volume	4.18	m^3

Concrete Density	2.20	g/cm^3
Surface Soil Depth	0.15	m
Irrigation Rate	0.274	L/m^2-d
Annual Total Well Water Vol	738	m^3
Embedded Pipe Conversion Factor	1191.7	pCi per dpm/100 cm^2
Gross Beta DCGL	8.00E+05	dpm/100 cm^2
Gross Beta Nuclide Fraction	0.616	
Total Inventory	1.30E+06	dpm/100 cm^2

	DOSE CALCULATION FACTORS			SOURCE TERM			Kd		WATER, FILL, CONCRETE CONCENTRATION				EMBEDDED PIPE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	2.80E-03	3.64E+03	4.34E+06	6.02E+01	1.00E+00	3.01E+02	1.95E-02	1.17E-03	1.95E-05	Sr-90	1.32E-03	1.27E-04	0.00E+00	1.45E-03
Cs-134	4.39E+00	7.33E-05	6.09E-05	4.55E-03	5.91E+03	7.04E+06	7.91E+01	3.00E+00	3.96E+02	2.41E-02	1.90E-03	7.22E-05	Cs-134	8.43E-04	4.69E-05	1.16E-07	8.90E-04
Cs-137	2.27E+00	5.00E-05	1.20E-05	5.50E-01	7.15E+05	8.52E+08	7.91E+01	3.00E+00	3.96E+02	2.91E+00	2.30E-01	8.73E-03	Cs-137	6.95E-02	2.94E-03	2.76E-06	7.25E-02
Co-60	6.58E+00	2.69E-05	6.30E-04	5.84E-02	7.59E+04	9.04E+07	1.28E+02	1.00E+02	6.40E+02	1.91E-01	2.44E-02	1.91E-02	Co-60	2.46E-03	5.59E-04	1.54E-05	3.03E-03
Co-57	1.67E-01	1.18E-06	2.80E-08	3.06E-04	3.98E+02	4.74E+05	1.28E+02	1.00E+02	6.40E+02	1.00E-03	1.28E-04	1.00E-04	Co-57	5.65E-07	7.44E-08	3.59E-12	6.40E-07
Fe-55	2.50E-03	6.07E-07	0.00E+00	4.81E-03	6.25E+03	7.45E+06	2.50E+01	1.00E+02	1.27E+02	7.93E-02	1.98E-03	7.93E-03	Fe-55	2.30E-05	8.82E-08	0.00E+00	2.31E-05
H-3	2.27E-01	6.40E-08	0.00E+00	2.36E-02	3.06E+04	3.64E+07	0.00E+00	0.00E+00	1.00E+00	4.93E+01	0.00E+00	0.00E+00	H-3	1.51E-03	4.97E-03	0.00E+00	6.48E-03
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.55E-01	4.61E+05	5.50E+08	1.28E+02	1.00E+02	6.40E+02	1.16E+00	1.48E-01	1.16E-01	Ni-63	3.21E-04	6.15E-06	0.00E+00	3.27E-04
					1.30E+06	1.55E+09											
													SUM	7.60E-02	8.65E-03	1.83E-05	8.47E-02

CONTAMINATED CONCRETE SPECIAL AREAS

Key Parameters:

Porosity	0.30		Concrete Density	2.20	g/cm^3
Bulk Density	1.50	g/cm^3	Annual Total Well Water Vol	738.0	m^3
Yearly Drinking Water	478.0	L/yr	Irrigation Rate	0.274	L/m^2-d
Wall Surface Area	4182.0	m^2	Surface Soil Depth	0.15	m
Fill Volume	2460.0	m^3	**Special Areas Gross Beta DCGL**	**9.50E+03**	dpm/100 cm^2
Surface Area/Open Volume	1.70	m^2/m^3	Gross Beta Nuclide Fraction	0.6672	
Concrete Volume	4.18	m^3	Total Inventory	1.42E+04	dpm/100 cm^2

3.01E-01 **<== Total Dose Contaminated Concrete**

1.08E-01 **<== Total Dose Special Areas**

DOSE CALCULATION FACTORS				SOURCE TERM			Kd		WATER, FILL, CONCRETE CONCENTRATION				CONTAMINATED CONCRETE ANNUAL DOSE				
Nuclide	NUREG-1727 mrem/y per pCi/g	FGR 11 mrem/pCi	Microshield mrem/y per pCi/g	Nuclide Fraction	Inventory dpm/100 cm^2	Inventory pCi	Kd Fill cm^3/gm	Kd Concrete cm^3/gm	Adsorption Factor	Water pCi/L	Fill pCi/g	Concrete pCi/g	Nuclide	Drinking Water Dose mrem/y	Irrigation Dose mrem/y	Direct Dose mrem/y	Total Dose mrem/y
Sr-90	1.47E+01	1.42E-04	0.00E+00	6.874E-03	9.79E+01	1.84E+07	6.12E+01	1.00E+00	3.06E+02	8.14E-02	4.98E-03	8.14E-05	Sr-90	5.53E-03	5.32E-04	0.00E+00	6.06E-03
Sb-125	9.77E-01	2.81E-06	3.83E-06	4.523E-03	6.44E+01	1.21E+07	4.50E+01	0.00E+00	2.26E+02	7.27E-02	3.27E-03	0.00E+00	Sb-125	9.77E-05	3.16E-05	1.25E-08	1.29E-04
Cs-134	4.39E+00	7.33E-05	6.09E-05	2.815E-03	4.01E+01	7.55E+06	7.91E+01	3.00E+00	3.96E+02	2.58E-02	2.04E-03	7.74E-05	Cs-134	9.04E-04	5.03E-05	1.24E-07	9.54E-04
Cs-137	2.27E+00	5.00E-05	1.20E-05	2.890E-01	4.12E+03	7.75E+08	7.91E+01	3.00E+00	3.96E+02	2.65E+00	2.10E-01	7.95E-03	Cs-137	6.33E-02	2.67E-03	2.51E-06	6.60E-02
Pu-238	1.00E+01	3.20E-03	2.45E-25	1.165E-04	1.66E+00	3.13E+05	5.50E+02	5.00E+03	2.81E+03	1.51E-04	8.28E-05	7.53E-04	Pu-238	2.30E-04	6.69E-07	2.03E-29	2.31E-04
Pu-239	1.09E+01	3.54E-03	6.10E-15	8.752E-05	1.25E+00	2.35E+05	5.50E+02	5.00E+03	2.81E+03	1.13E-04	6.22E-05	5.65E-04	Pu-239	1.91E-04	5.48E-07	3.79E-19	1.92E-04
Pu-240	1.09E+01	3.54E-03	7.52E-26	8.750E-05	1.25E+00	2.35E+05	5.50E+02	5.00E+03	2.81E+03	1.13E-04	6.22E-05	5.65E-04	Pu-240	1.91E-04	5.48E-07	4.68E-30	1.92E-04
Pu-241	3.47E-01	6.85E-05	0.00E+00	6.705E-03	9.55E+01	1.80E+07	5.50E+02	5.00E+03	2.81E+03	8.66E-03	4.76E-03	4.33E-02	Pu-241	2.84E-04	1.34E-06	0.00E+00	2.85E-04
Am-241	1.19E+01	3.64E-03	1.65E-19	5.929E-04	8.44E+00	1.59E+06	1.90E+03	5.00E+03	9.55E+03	2.25E-04	4.28E-04	1.13E-03	Am-241	3.92E-04	1.19E-06	7.08E-23	3.93E-04
Cm-243	7.81E+00	2.51E-03	1.27E-08	4.649E-05	6.62E-01	1.25E+05	4.00E+03	5.00E+03	2.00E+04	8.42E-06	3.37E-05	4.21E-05	Cm-243	1.01E-05	2.92E-08	4.28E-13	1.01E-05
Cm-244	6.00E+00	2.02E-03	9.81E-25	4.454E-05	6.34E-01	1.19E+05	4.00E+03	5.00E+03	2.00E+04	8.07E-06	3.23E-05	4.03E-05	Cm-244	7.79E-06	2.15E-08	3.17E-29	7.81E-06
Co-60	6.58E+00	2.69E-05	6.30E-04	3.639E-01	5.18E+03	9.76E+08	1.28E+02	1.00E+02	6.40E+02	2.06E+00	2.64E-01	2.06E-01	Co-60	2.65E-02	6.04E-03	1.66E-04	3.27E-02
Co-57	1.67E-01	1.18E-06	2.80E-08		0.00E+00	0.00E+00	1.28E+02	1.00E+02	6.40E+02	0.00E+00	0.00E+00	0.00E+00	Co-57	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	1.67E+00	2.77E-06	4.40E-05	4.028E-04	5.74E+00	1.08E+06	5.00E+01	0.00E+00	2.51E+02	5.83E-03	2.92E-04	0.00E+00	Mn-54	7.72E-06	4.33E-06	1.28E-08	1.21E-05
Fe-55	2.50E-03	6.07E-07	0.00E+00	2.235E-02	3.18E+02	6.00E+07	2.50E+01	1.00E+02	1.27E+02	6.38E-01	1.60E-02	6.38E-02	Fe-55	1.85E-04	7.09E-07	0.00E+00	1.86E-04
H-3	2.27E-01	6.40E-08	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.00E+00	0.00E+00	0.00E+00	0.00E+00	H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ni-63	1.19E-02	5.77E-07	0.00E+00	3.024E-01	4.31E+03	8.11E+08	1.28E+02	1.00E+02	6.40E+02	1.71E+00	2.19E-01	1.71E-01	Ni-63	4.73E-04	9.07E-06	0.00E+00	4.82E-04
													SUM	9.83E-02	9.34E-03	1.69E-04	1.08E-01

Attachment 6-14
Soil Area Factor Microshield Output

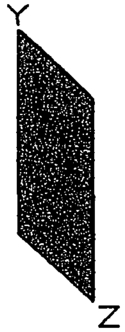
Attachment 6-14 illustrates the Microshield runs for determination of soil area factors. The associated Engineering Calculation for soil area factors provides all the Microshield runs used to derive the area factors for the Maine Yankee nuclide mixture, and mixtures containing 100 percent Co-60 and 100 percent Cs-137. These are presented in Section 6, Table 6-12 of the LTP. The runs illustrated in this attachment are for 100 percent Cs-137. These runs are the most conservative of the three area factor groups.

MicroShield v5.05 (5.05-00461)
Maine Yankee

File Ref: ____
Date: ____
By: DBR
Checked: ____

Page : 1
DOS File : S10000CS.MS5
Run Date: May 17, 2002
Run Time: 7:06:32 AM
Duration : 00:00:02

Case Title: Soil AF 10,000 m^2
Description: Soil AF for CS-137 only, 1e4 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	1.0e+4 cm	328 ft 1.0 in
Height	1.0e+4 cm	328 ft 1.0 in

Dose Points

	X	Y	Z
# 1	115 cm	5000 cm	5000 cm
	3 ft 9.3 in	164 ft 0.5 in	164 ft 0.5 in

Shields

Shield Name	Dimension	Material	Density
Source	1.50e+09 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	2.4000e-003	8.8800e+007	1.6000e-006	5.9200e-002
Cs-137	2.4000e-003	8.8800e+007	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.838e+06	1.604e-05	2.373e-05	1.336e-07	1.977e-07
0.0322	3.392e+06	3.089e-05	4.611e-05	2.486e-07	3.711e-07
0.0364	1.234e+06	1.724e-05	2.857e-05	9.796e-08	1.623e-07
0.6616	7.990e+07	1.414e-01	3.078e-01	2.740e-04	5.968e-04

OS File : S10000CS.MS5
un Date: May 17, 2002
un Time: 7:06:32 AM
uration : 00:00:02

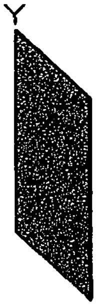
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	8.637e+07	1.414e-01	3.079e-01	2.745e-04	5.975e-04

MicroShield v5.05 (5.05-00461)
Maine Yankee

Page : 1
DOS File : S5000CS.MS5
Run Date: May 17, 2002
Run Time: 7:26:18 AM
Duration : 00:00:02

File Ref: ________
Date: ________
By: DDR
Checked: ________

Case Title: Soil AF 5,000 m^2
Description: Soil AF for Cs-137 only, 5e3 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	7.1e+3 cm	231 ft 11.9 in
Height	7.1e+3 cm	231 ft 11.9 in

Dose Points

	X	Y	Z
# 1	115 cm	3536 cm	3536 cm
	3 ft 9.3 in	116 ft 0.1 in	116 ft 0.1 in

Shields

Shield Name	Dimension	Material	Density
Source	7.50e+08 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	1.2000e-003	4.4399e+007	1.6000e-006	5.9200e-002
Cs-137	1.2000e-003	4.4399e+007	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.192e+05	1.517e-05	2.248e-05	1.264e-07	1.873e-07
0.0322	1.696e+06	2.922e-05	4.368e-05	2.352e-07	3.516e-07
0.0364	6.171e+05	1.632e-05	2.711e-05	9.273e-08	1.540e-07
0.6616	3.995e+07	1.329e-01	2.914e-01	2.576e-04	5.648e-04

JOS File : S5000CS.MS5
Run Date: May 17, 2002
Run Time: 7:26:18 AM
Duration : 00:00:02

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	4.318e+07	1.329e-01	2.914e-01	2.580e-04	5.655e-04

MicroShield v5.05 (5.05-00461)
Maine Yankee

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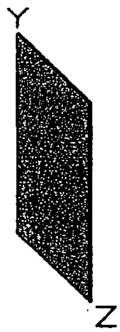
File Ref: ________
Date: ________
By: DBR
Checked: ________

Page : 1
DOS File : S2500CS.MS5
Run Date: May 17, 2002
Run Time: 7:33:30 AM
Duration : 00:00:02

Case Title: Soil AF 2,500 m^2
Description: Soil AF for Cs-137 only, 2.5e3 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	5.0e+3 cm	164 ft 0.5 in
Height	5.0e+3 cm	164 ft 0.5 in

Dose Points

	X	Y	Z
# 1	115 cm	2500 cm	2500 cm
	3 ft 9.3 in	82 ft 0.3 in	82 ft 0.3 in

Shields

Shield Name	Dimension	Material	Density
Source	3.75e+08 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	6.0000e-004	2.2200e+007	1.6000e-006	5.9200e-002
Cs-137	6.0000e-004	2.2200e+007	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	4.596e+05	1.440e-05	2.144e-05	1.199e-07	1.786e-07
0.0322	8.480e+05	2.774e-05	4.166e-05	2.233e-07	3.353e-07
0.0364	3.086e+05	1.551e-05	2.582e-05	8.815e-08	1.467e-07
0.6616	1.998e+07	1.272e-01	2.818e-01	2.465e-04	5.464e-04

DOS File : S2500CS.MS5
Run Date: May 17, 2002
Run Time: 7:33:30 AM
Duration : 00:00:02

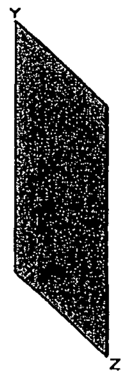
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	2.159e+07	1.272e-01	2.819e-01	2.469e-04	5.470e-04

MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: S2000CS.MS5
Run Date: June 6, 2002
Run Time: 8:48:53 AM
Duration: 00:00:09

File Ref:
Date:
By: DR
Checked:

Case Title: Soil AF 2,000 m^2
Description: Soil AF for Cs-137 only, 2e3 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	4.5e+3 cm	146 ft 8.6 in
Height	4.5e+3 cm	146 ft 8.6 in

Dose Points

	X	Y	Z
# 1	115 cm	2236 cm	2236 cm
	3 ft 9.3 in	73 ft 4.3 in	73 ft 4.3 in

Shields

Shield Name	Dimension	Material	Density
Source	3.00e+08 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	4.7997e-004	1.7759e+007	1.6000e-006	5.9200e-002
Cs-137	4.7997e-004	1.7759e+007	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	3.677e+05	1.428e-05	2.126e-05	1.189e-07	1.771e-07
0.0322	6.783e+05	2.751e-05	4.130e-05	2.214e-07	3.324e-07
0.0364	2.468e+05	1.538e-05	2.560e-05	8.740e-08	1.454e-07
0.6616	1.598e+07	1.262e-01	2.800e-01	2.446e-04	5.428e-04
TOTALS:	1.727e+07	1.262e-01	2.801e-01	2.451e-04	5.435e-04

MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: S1000CS.MS5
Run Date: June 6, 2002
Run Time: 8:50:07 AM
Duration: 00:00:09

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: Soil AF 1,000 m^2
Description: Soil AF for Cs-137 only, 1e3 m^2
Geometry: 13 - Rectangular Volume

Source Dimensions

Length	15.0 cm	5.9 in
Width	3.2e+3 cm	103 ft 8.9 in
Height	3.2e+3 cm	103 ft 8.9 in

Dose Points

	X	Y	Z
# 1	115 cm	1581 cm	1581 cm
	3 ft 9.3 in	51 ft 10.4 in	51 ft 10.4 in

Shields

Shield Name	Dimension	Material	Density
Source	1.50e+08 cm^3	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	$\mu Ci/cm^3$	Bq/cm^3
Ba-137m	2.3996e-004	8.8784e+006	1.6000e-006	5.9200e-002
Cs-137	2.3996e-004	8.8784e+006	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate $MeV/cm^2/sec$ No Buildup	Fluence Rate $MeV/cm^2/sec$ With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.838e+05	1.417e-05	2.108e-05	1.181e-07	1.756e-07
0.0322	3.391e+05	2.731e-05	4.095e-05	2.198e-07	3.295e-07
0.0364	1.234e+05	1.525e-05	2.534e-05	8.665e-08	1.439e-07
0.6616	7.989e+06	1.242e-01	2.749e-01	2.408e-04	5.329e-04
TOTALS:	8.635e+06	1.243e-01	2.750e-01	2.412e-04	5.336e-04

MicroShield v5.05 (5.05-00461)
Maine Yankee

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File Ref: ________
Date: ________
By: DR
Checked: ________

'age : 1
)OS File : S500CS.MS5
₹un Date: May 17, 2002
₹un Time: 8:02:12 AM
)uration : 00:00:05

Case Title: Soil AF 500 m^2
Description: Soil AF for Cs-137 only, 5e2 m^2
Geometry: 13 - Rectangular Volume

Source Dimensions

Length	15.0 cm	5.9 in
Width	2.2e+3 cm	73 ft 4.3 in
Height	2.2e+3 cm	73 ft 4.3 in

Dose Points

	X	Y	Z
# 1	115 cm	1118 cm	1118 cm
	3 ft 9.3 in	36 ft 8.2 in	36 ft 8.2 in

Shields

Shield Name	Dimension	Material	Density
Source	7.50e+07 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	1.1999e-004	4.4397e+006	1.6000e-006	5.9200e-002
Cs-137	1.1999e-004	4.4397e+006	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.192e+04	1.414e-05	2.098e-05	1.178e-07	1.748e-07
0.0322	1.696e+05	2.723e-05	4.075e-05	2.192e-07	3.280e-07
0.0364	6.171e+04	1.515e-05	2.505e-05	8.607e-08	1.423e-07
0.6616	3.995e+06	1.216e-01	2.675e-01	2.357e-04	5.186e-04

)OS File : S500CS.MS5
Run Date: May 17, 2002
Run Time: 8:02:12 AM
)uration : 00:00:05

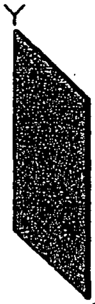
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	4.318e+06	1.216e-01	2.676e-01	2.361e-04	5.192e-04

MicroShield v5.05 (5.05-00461)
Maine Yankee

File Ref: ______
Date: ______
By: DR
Checked: ______

Page : 1
DOS File : S300CS.MS5
Run Date: May 17, 2002
Run Time: 9:22:35 AM
Duration : 00:00:02

Case Title: Soil AF 300 m^2
Description: Soil AF for Cs-137 only, 3e2 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	1.7e+3 cm	56 ft 9.9 in
Height	1.7e+3 cm	56 ft 9.9 in

Dose Points

	X	Y	Z
# 1	115 cm	866 cm	866 cm
	3 ft 9.3 in	28 ft 4.9 in	28 ft 4.9 in

Shields

Shield Name	Dimension	Material	Density
Source	4.50e+07 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	7.1996e-005	2.6638e+006	1.6000e-006	5.9200e-002
Cs-137	7.1996e-005	2.6638e+006	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	5.515e+04	1.400e-05	2.071e-05	1.166e-07	1.725e-07
0.0322	1.018e+05	2.695e-05	4.018e-05	2.169e-07	3.233e-07
0.0364	3.703e+04	1.493e-05	2.456e-05	8.482e-08	1.396e-07
0.6616	2.397e+06	1.186e-01	2.596e-01	2.299e-04	5.033e-04

DOS File : S300CS.MS5
Run Date: May 17, 2002
Run Time: 9:22:35 AM
Duration : 00:00:02

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	2.591e+06	1.186e-01	2.597e-01	2.303e-04	5.039e-04

MicroShield v5.05 (5.05-00105)
Stone & Webster

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Page : 1
DOS File: S100CS.MS5
Run Date: June 6, 2002
Run Time: 8:52:09 AM
Duration: 00:00:08

File Ref:
Date:
By: DR
Checked:

Case Title: Soil AF 100 m^2
Description: Soil AF for Cs-137 only, 1e2 m^2
Geometry: 13 - Rectangular Volume

Source Dimensions

Length	15.0 cm	5.9 in
Width	1.0e+3 cm	32 ft 9.7 in
Height	1.0e+3 cm	32 ft 9.7 in

Dose Points

	X	Y	Z
# 1	115 cm	500 cm	500 cm
	3 ft 9.3 in	16 ft 4.9 in	16 ft 4.9 in

Shields

Shield Name	Dimension	Material	Density
Source	1.50e+07 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	2.4000e-005	8.8800e+005	1.6000e-006	5.9200e-002
Cs-137	2.4000e-005	8.8800e+005	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.838e+04	1.318e-05	1.930e-05	1.098e-07	1.607e-07
0.0322	3.392e+04	2.535e-05	3.739e-05	2.040e-07	3.009e-07
0.0364	1.234e+04	1.393e-05	2.266e-05	7.913e-08	1.287e-07
0.6616	7.990e+05	1.085e-01	2.337e-01	2.103e-04	4.530e-04
TOTALS:	8.637e+05	1.085e-01	2.338e-01	2.107e-04	4.536e-04

MicroShield v5.05 (5.05-00105)
Stone & Webster

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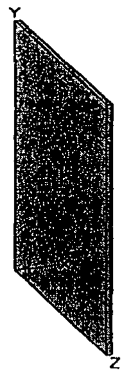
Page : 1
DOS File: S50CS.MS5
Run Date: June 6, 2002
Run Time: 8:53:37 AM
Duration: 00:00:08

File Ref:
Date:
By: DR
Checked:

Case Title: Soil AF 50 m^2
Description: Soil AF for Cs-137 only, 50 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	707.1 cm	23 ft 2.4 in
Height	707.1 cm	23 ft 2.4 in

Dose Points

	X	Y	Z
# 1	115 cm	353.6 cm	353.6 cm
	3 ft 9.3 in	11 ft 7.2 in	11 ft 7.2 in

Shields

Shield Name	Dimension	Material	Density
Source	7.50e+06 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	1.2000e-005	4.4399e+005	1.6000e-006	5.9200e-002
Cs-137	1.2000e-005	4.4399e+005	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

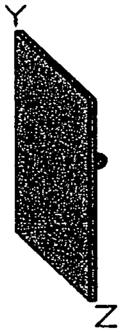
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.192e+03	1.221e-05	1.775e-05	1.017e-07	1.478e-07
0.0322	1.696e+04	2.346e-05	3.438e-05	1.888e-07	2.767e-07
0.0364	6.171e+03	1.285e-05	2.080e-05	7.300e-08	1.182e-07
0.6616	3.995e+05	9.868e-02	2.088e-01	1.913e-04	4.047e-04
TOTALS:	4.318e+05	9.873e-02	2.088e-01	1.917e-04	4.053e-04

MicroShield v5.05 (5.05-00461)
Maine Yankee

ʾage : 1
)OS File : S25CS.MS5
ʾun Date: May 17, 2002
ʾun Time: 8:24:07 AM
)uration : 00:00:07

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: Soil AF 25 m^2
Description: Soil AF for Cs-137 only, 25 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	500.0 cm	16 ft 4.9 in
Height	500.0 cm	16 ft 4.9 in

Dose Points

	X	Y	Z
# 1	115 cm	250 cm	250 cm
	3 ft 9.3 in	8 ft 2.4 in	8 ft 2.4 in

Shields

Shield Name	Dimension	Material	Density
Source	3.75e+06 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	6.0000e-006	2.2200e+005	1.6000e-006	5.9200e-002
Cs-137	6.0000e-006	2.2200e+005	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	4.596e+03	1.080e-05	1.563e-05	9.000e-08	1.302e-07
0.0322	8.480e+03	2.076e-05	3.028e-05	1.671e-07	2.437e-07
0.0364	3.086e+03	1.135e-05	1.831e-05	6.448e-08	1.041e-07
0.6616	1.998e+05	8.554e-02	1.762e-01	1.658e-04	3.415e-04

)OS File : S25CS.MS5
Run Date: May 17, 2002
Run Time: 8:24:07 AM
Duration : 00:00:07

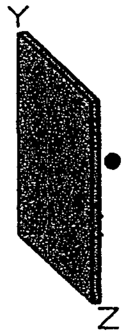
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	2.159e+05	8.558e-02	1.762e-01	1.661e-04	3.420e-04

MicroShield v5.05 (5.05-00461)
Maine Yankee

File Ref: ________
Date: ________
By: DR
Checked: ________

Page : 1
DOS File : S16CS.MS5
Run Date: May 17, 2002
Run Time: 8:27:32 AM
Duration : 00:00:09

Case Title: Soil AF 16 m^2
Description: Soil AF for Cs-137 only, 16 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	400.0 cm	13 ft 1.5 in
Height	400.0 cm	13 ft 1.5 in

Dose Points

	X	Y	Z
# 1	115 cm	200 cm	200 cm
	3 ft 9.3 in	6 ft 6.7 in	6 ft 6.7 in

Shields

Shield Name	Dimension	Material	Density
Source	2.40e+06 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	3.8400e-006	1.4208e+005	1.6000e-006	5.9200e-002
Cs-137	3.8400e-006	1.4208e+005	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	2.941e+03	9.678e-06	1.397e-05	8.061e-08	1.163e-07
0.0322	5.427e+03	1.859e-05	2.705e-05	1.496e-07	2.177e-07
0.0364	1.975e+03	1.015e-05	1.635e-05	5.770e-08	9.291e-08
0.6616	1.278e+05	7.531e-02	1.518e-01	1.460e-04	2.943e-04

)OS File : S16CS.MS5
₹un Date: May 17, 2002
₹un Time: 8:27:32 AM
)uration : 00:00:09

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	1.382e+05	7.535e-02	1.519e-01	1.463e-04	2.947e-04

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Maine Yankee

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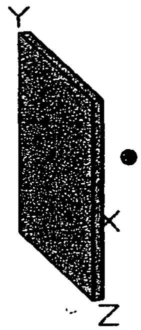
File Ref: ________
Date: ________
By: DR
Checked: ________

Page : 1
DOS File : S10CS.MS5
Run Date: May 17, 2002
Run Time: 8:35:28 AM
Duration : 00:00:07

Case Title: Soil AF 10 m^2
Description: Soil AF for Cs-137 only, 10 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	316.2 cm	10 ft 4.5 in
Height	316.2 cm	10 ft 4.5 in

Dose Points

	X	Y	Z
# 1	115 cm	158.1 cm	158.1 cm
	3 ft 9.3 in	5 ft 2.2 in	5 ft 2.2 in

Shields

Shield Name	Dimension	Material	Density
Source	1.50e+06 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	2.3996e-006	8.8784e+004	1.6000e-006	5.9200e-002
Cs-137	2.3996e-006	8.8784e+004	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.838e+03	8.329e-06	1.199e-05	6.938e-08	9.983e-08
0.0322	3.391e+03	1.600e-05	2.321e-05	1.288e-07	1.868e-07
0.0364	1.234e+03	8.730e-06	1.403e-05	4.960e-08	7.970e-08
0.6616	7.989e+04	6.344e-02	1.247e-01	1.230e-04	2.417e-04

JOS File : S10CS.MS5
Run Date: May 17, 2002
Run Time: 8:35:28 AM
Duration : 00:00:07

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	8.635e+04	6.347e-02	1.247e-01	1.232e-04	2.421e-04

MicroShield v5.05 (5.05-00461)
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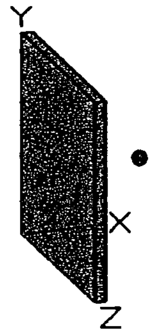
'age : 1
)OS File : S8CS.MS5
₹un Date: May 17, 2002
₹un Time: 8:37:33 AM
)uration : 00:00:09

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: Soil AF 8 m^2
Description: Soil AF for Cs-137 only, 8 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	282.8 cm	9 ft 3.3 in
Height	282.8 cm	9 ft 3.3 in

Dose Points

	X	Y	Z
# 1	115 cm	141.4 cm	141.4 cm
	3 ft 9.3 in	4 ft 7.7 in	4 ft 7.7 in

Shields

Shield Name	Dimension	Material	Density
Source	1.20e+06 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	1.9194e-006	7.1019e+004	1.6000e-006	5.9200e-002
Cs-137	1.9194e-006	7.1019e+004	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.470e+03	7.648e-06	1.099e-05	6.370e-08	9.153e-08
0.0322	2.713e+03	1.469e-05	2.128e-05	1.182e-07	1.713e-07
0.0364	9.872e+02	8.012e-06	1.286e-05	4.552e-08	7.307e-08
0.6616	6.390e+04	5.760e-02	1.118e-01	1.117e-04	2.168e-04

)OS File : S8CS.MS5
Run Date: May 17, 2002
Run Time: 8:37:33 AM
)uration : 00:00:09

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	6.907e+04	5.763e-02	1.119e-01	1.119e-04	2.172e-04

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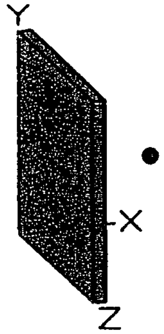
'age : 1
)OS File : S6CS.MS5
Run Date: May 17, 2002
Run Time: 8:44:26 AM
Duration : 00:00:09

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: Soil AF 6 m^2
Description: Soil AF for Cs-137 only, 6 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	244.9 cm	8 ft 0.4 in
Height	244.9 cm	8 ft 0.4 in

Dose Points

	X	Y	Z
# 1	115 cm	122.5 cm	122.5 cm
	3 ft 9.3 in	4 ft 0.2 in	4 ft 0.2 in

Shields

Shield Name	Dimension	Material	Density
Source	9.00e+05 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	1.4394e-006	5.3259e+004	1.6000e-006	5.9200e-002
Cs-137	1.4394e-006	5.3259e+004	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.103e+03	6.751e-06	9.683e-06	5.623e-08	8.066e-08
0.0322	2.034e+03	1.297e-05	1.875e-05	1.044e-07	1.509e-07
0.0364	7.403e+02	7.068e-06	1.133e-05	4.016e-08	6.438e-08
0.6616	4.792e+04	5.008e-02	9.576e-02	9.708e-05	1.856e-04

)OS File : S6CS.MS5
Run Date: May 17, 2002
Run Time: 8:44:26 AM
Duration : 00:00:09

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	5.180e+04	5.011e-02	9.580e-02	9.728e-05	1.859e-04

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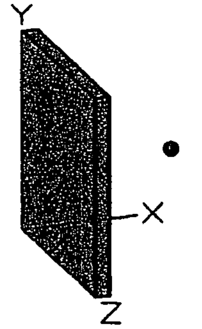
Maine Yankee

File Ref: ________
Date: ________
By: DR
Checked: ________

'age : 1
)OS File : S4CS.MS5
२un Date: May 17, 2002
२un Time: 8:46:52 AM
)uration : 00:00:09

Case Title: Soil AF 4 m^2
Description: Soil AF for Cs-137 only, 4 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	200.0 cm	6 ft 6.7 in
Height	200.0 cm	6 ft 6.7 in

Dose Points

	X	Y	Z
# 1	115 cm	100 cm	100 cm
	3 ft 9.3 in	3 ft 3.4 in	3 ft 3.4 in

Shields

Shield Name	Dimension	Material	Density
Source	6.00e+05 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	9.6000e-007	3.5520e+004	1.6000e-006	5.9200e-002
Cs-137	9.6000e-007	3.5520e+004	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	7.354e+02	5.499e-06	7.872e-06	4.581e-08	6.557e-08
0.0322	1.357e+03	1.056e-05	1.525e-05	8.500e-08	1.227e-07
0.0364	4.937e+02	5.752e-06	9.208e-06	3.268e-08	5.232e-08
0.6616	3.196e+04	3.992e-02	7.480e-02	7.739e-05	1.450e-04

)OS File : S4CS.MS5
{un Date: May 17, 2002
{un Time: 8:46:52 AM
)uration : 00:00:09

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	3.455e+04	3.994e-02	7.484e-02	7.755e-05	1.453e-04

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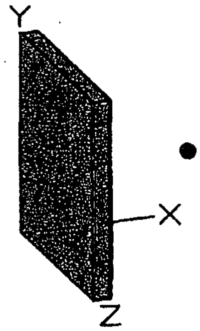
File Ref: ____
Date: ____
By: DR
Checked: ____

Page : 1
DOS File : S3CS.MS5
Run Date : May 17, 2002
Run Time : 8:53:12 AM
Duration : 00:00:05

Case Title: Soil AF 3 m^2
Description: Soil AF for Cs-137 only, 3 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	173.2 cm	5 ft 8.2 in
Height	173.2 cm	5 ft 8.2 in

Dose Points

	X	Y	Z
# 1	115 cm	86.6 cm	86.6 cm
	3 ft 9.3 in	2 ft 10.1 in	2 ft 10.1 in

Shields

Shield Name	Dimension	Material	Density
Source	4.50e+05 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	7.1996e-007	2.6638e+004	1.6000e-006	5.9200e-002
Cs-137	7.1996e-007	2.6638e+004	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	5.515e+02	4.653e-06	6.653e-06	3.876e-08	5.542e-08
0.0322	1.018e+03	8.936e-06	1.289e-05	7.192e-08	1.037e-07
0.0364	3.703e+02	4.863e-06	7.778e-06	2.763e-08	4.419e-08
0.6616	2.397e+04	3.327e-02	6.156e-02	6.450e-05	1.193e-04

JOS File : S3CS.MS5
Run Date: May 17, 2002
Run Time: 8:53:12 AM
Duration : 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	2.591e+04	3.329e-02	6.158e-02	6.464e-05	1.195e-04

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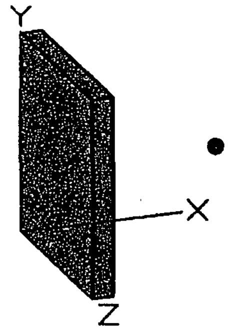
'age : 1
)OS File : S2CS.MS5
ìun Date: May 17, 2002
ìun Time: 8:55:48 AM
)uration : 00:00:02

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: Soil AF 2 m^2
Description: Soil AF for Cs-137 only, 2 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	141.4 cm	4 ft 7.7 in
Height	141.4 cm	4 ft 7.7 in

Dose Points

	X	Y	Z
# 1	115 cm	70.71 cm	70.71 cm
	3 ft 9.3 in	2 ft 3.8 in	2 ft 3.8 in

Shields

Shield Name	Dimension	Material	Density
Source	3.00e+05 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	4.7986e-007	1.7755e+004	1.6000e-006	5.9200e-002
Cs-137	4.7986e-007	1.7755e+004	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	3.676e+02	3.569e-06	5.097e-06	2.973e-08	4.246e-08
0.0322	6.782e+02	6.854e-06	9.872e-06	5.516e-08	7.945e-08
0.0364	2.468e+02	3.727e-06	5.954e-06	2.118e-08	3.383e-08
0.6616	1.598e+04	2.503e-02	4.560e-02	4.853e-05	8.840e-05

)OS File : S2CS.MS5
Run Date: May 17, 2002
Run Time: 8:55:48 AM
)uration : 00:00:02

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	1.727e+04	2.505e-02	4.562e-02	4.863e-05	8.856e-05

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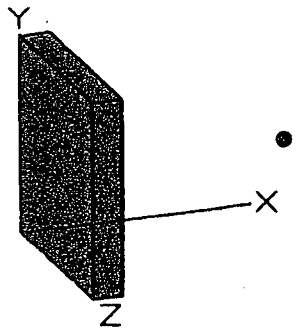
File Ref: ___
Date: ___
By: DR
Checked: ___

Page : 1
DOS File : S1CS.MS5
Run Date: May 17, 2002
Run Time: 9:16:15 AM
Duration : 00:00:07

Case Title: Soil AF 1 m^2
Description: Soil AF for Cs-137 only, 1 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	100.0 cm	3 ft 3.4 in
Height	100.0 cm	3 ft 3.4 in

Dose Points

	X	Y	Z
# 1	115 cm	50 cm	50 cm
	3 ft 9.3 in	1 ft 7.7 in	1 ft 7.7 in

Shields

Shield Name	Dimension	Material	Density
Source	1.50e+05 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	2.4000e-007	8.8800e+003	1.6000e-006	5.9200e-002
Cs-137	2.4000e-007	8.8800e+003	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.838e+02	2.114e-06	3.014e-06	1.761e-08	2.511e-08
0.0322	3.392e+02	4.059e-06	5.837e-06	3.266e-08	4.698e-08
0.0364	1.234e+02	2.204e-06	3.515e-06	1.252e-08	1.997e-08
0.6616	7.990e+03	1.444e-02	2.580e-02	2.799e-05	5.001e-05

)OS File : S1CS.MS5
Run Date: May 17, 2002
Run Time: 9:16:15 AM
Duration : 00:00:07

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	8.637e+03	1.444e-02	2.581e-02	2.805e-05	5.010e-05

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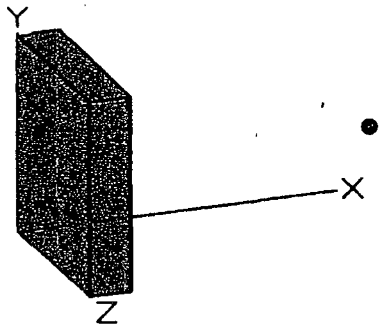
DOS File	: S_5CS.MS5
Run Date	: May 17, 2002
Run Time	: 9:05:35 AM
Duration	: 00:00:02

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: Soil AF 0.5 m^2
Description: Soil AF for Cs-137 only, 0.5 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	70.71 cm	2 ft 3.8 in
Height	70.71 cm	2 ft 3.8 in

Dose Points

	X	Y	Z
# 1	115 cm	35.36 cm	35.36 cm
	3 ft 9.3 in	1 ft 1.9 in	1 ft 1.9 in

Shields

Shield Name	Dimension	Material	Density
Source	7.50e+04 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	1.2000e-007	4.4399e+003	1.6000e-006	5.9200e-002
Cs-137	1.2000e-007	4.4399e+003	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.192e+01	1.168e-06	1.664e-06	9.727e-09	1.386e-08
0.0322	1.696e+02	2.242e-06	3.222e-06	1.804e-08	2.593e-08
0.0364	6.171e+01	1.217e-06	1.938e-06	6.913e-09	1.101e-08
0.6616	3.995e+03	7.837e-03	1.384e-02	1.519e-05	2.683e-05

)OS File : S_5CS.MS5
Run Date: May 17, 2002
Run Time: 9:05:35 AM
Duration : 00:00:02

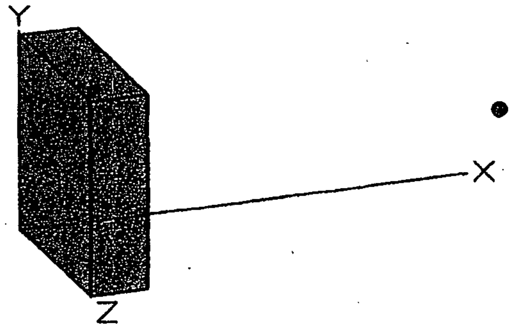
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	4.318e+03	7.842e-03	1.385e-02	1.523e-05	2.689e-05

MicroShield v5.05 (5.05-00481)
Maine Yankee

Page : 1
DOS File : S_25CS.MS5
Run Date: May 17, 2002
Run Time: 9:14:52 AM
Duration : 00:00:09

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: Soil AF 0.25 m^2
Description: Soil AF for Cs-137 only, 0.25 m^2
Geometry: 13 - Rectangular Volume



Source Dimensions

Length	15.0 cm	5.9 in
Width	50.0 cm	1 ft 7.7 in
Height	50.0 cm	1 ft 7.7 in

Dose Points

	X	Y	Z
# 1	115 cm	25 cm	25 cm
	3 ft 9.3 in	9.8 in	9.8 in

Shields

Shield Name	Dimension	Material	Density
Source	3.75e+04 cm³	soil (SiO2)	1.6
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	6.0000e-008	2.2200e+003	1.6000e-006	5.9200e-002
Cs-137	6.0000e-008	2.2200e+003	1.6000e-006	5.9200e-002

Buildup
The material reference is : Source

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	4.596e+01	6.169e-07	8.784e-07	5.138e-09	7.317e-09
0.0322	8.480e+01	1.184e-06	1.701e-06	9.532e-09	1.369e-08
0.0364	3.086e+01	6.424e-07	1.023e-06	3.650e-09	5.809e-09
0.6616	1.998e+03	4.098e-03	7.191e-03	7.945e-06	1.394e-05

)US File : S_Z5US.MS5
Run Date: May 17, 2002
Run Time: 9:14:52 AM
Duration : 00:00:09

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
TOTALS:	2.159e+03	4.100e-03	7.195e-03	7.963e-06	1.397e-05

Attachment 6-15
Standing Building Area Factor Microshield Output

Attachment 6-15 illustrates the Microshield runs for determination of standing building area factors. The associated Engineering Calculation for standing building area factors provides all the Microshield runs used to derive the area factors for the Maine Yankee nuclide mixture and mixtures containing100 percent Co-60 and 100 percent Cs-137. These are presented in Section 6, Table 6-14 of the LTP. The runs illustrated in this attachment are for the Maine Yankee nuclide mixture.

Calc 016-01 (MY)
Rev 2, Att 2
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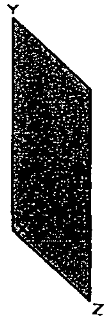
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC10000.MS5
Run Date: June 12, 2002
Run Time: 7:48:19 AM
Duration: 00:00:06

File Ref:
Date:
By: DR
Checked:

Case Title: AF 10,000 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	1.0e+4 cm	328 ft 1.0 in
Height	1.0e+4 cm	328 ft 1.0 in

Dose Points

	X	Y	Z
# 1	100 cm	5000 cm	5000 cm
	3 ft 3.4 in	164 ft 0.5 in	164 ft 0.5 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	2.4800e-007	9.1760e+003	2.4800e-009	9.1760e-005
Co-57	1.3800e-010	5.1060e+000	1.3800e-012	5.1060e-008
Co-60	2.6300e-008	9.7310e+002	2.6300e-010	9.7310e-006
Cs-134	2.0500e-009	7.5850e+001	2.0500e-011	7.5850e-007
Cs-137	2.4800e-007	9.1760e+003	2.4800e-009	9.1760e-005
Fe-55	2.1700e-009	8.0290e+001	2.1700e-011	8.0290e-007
H-3	1.0600e-008	3.9220e+002	1.0600e-010	3.9220e-006
Ni-63	1.6000e-007	5.9200e+003	1.6000e-009	5.9200e-005
Sr-90	1.2600e-009	4.6620e+001	1.2600e-011	4.6620e-007
Y-90	1.2600e-009	4.6620e+001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.900e+02	8.502e-08	1.222e-07	7.082e-10	1.018e-09
0.0318	1.626e-01	7.277e-11	1.046e-10	6.062e-13	8.712e-13
0.0322	3.505e+02	1.595e-07	2.302e-07	1.283e-09	1.853e-09
0.0322	3.000e-01	1.365e-10	1.971e-10	1.099e-12	1.586e-12
0.0364	1.275e+02	6.839e-08	1.031e-07	3.886e-10	5.857e-10
0.0364	1.092e-01	5.854e-11	8.824e-11	3.326e-13	5.013e-13
0.1221	4.366e+00	8.929e-09	1.264e-08	1.400e-11	1.982e-11

Calc 916-01 (MY)
Rev 2, Att 2
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Page : 2
DOS File: CC10000.MS5
Run Date: June 12, 2002
Run Time: 7:48:19 AM
Duration: 00:00:06

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	5.414e-01	1.245e-09	1.728e-09	2.002e-12	2.779e-12
0.2769	2.685e-02	1.300e-10	1.567e-10	2.439e-13	2.939e-13
0.4753	1.107e+00	9.470e-09	1.076e-08	1.858e-11	2.110e-11
0.536	1.464e-03	1.420e-11	1.596e-11	2.785e-14	3.130e-14
0.5632	6.356e+00	6.496e-08	7.272e-08	1.272e-10	1.424e-10
0.5693	1.170e+01	1.210e-07	1.353e-07	2.368e-10	2.648e-10
0.6047	7.403e+01	8.152e-07	9.073e-07	1.590e-09	1.770e-09
0.6616	8.257e+03	9.992e-05	1.104e-04	1.937e-07	2.141e-07
0.692	8.164e-03	1.036e-10	1.141e-10	2.000e-13	2.203e-13
0.6938	1.587e-01	2.019e-09	2.224e-09	3.898e-12	4.293e-12
0.7958	6.478e+01	9.513e-07	1.038e-06	1.811e-09	1.975e-09
0.8019	6.622e+00	9.802e-08	1.069e-07	1.864e-10	2.033e-10
1.0386	7.585e-01	1.472e-08	1.582e-08	2.695e-11	2.898e-11
1.1679	1.365e+00	2.994e-08	3.202e-08	5.356e-11	5.727e-11
1.1732	9.731e+02	2.144e-05	2.292e-05	3.832e-08	4.096e-08
1.3325	9.731e+02	2.449e-05	2.603e-05	4.248e-08	4.515e-08
1.3652	2.306e+00	5.951e-08	6.318e-08	1.026e-10	1.089e-10
TOTALS:	1.105e+04	1.483e-04	1.622e-04	2.811e-07	3.083e-07

C91C 016-02 (MY)
Rev 2, Att 2
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MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC5000.MS5
Run Date: June 12, 2002
Run Time: 8:34:10 AM
Duration: 00:00:06

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: AF 5,000 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	7.1e+3 cm	231 ft 11.9 in
Height	7.1e+3 cm	231 ft 11.9 in

Dose Points

	X	Y	Z
# 1	100 cm	3536 cm	3536 cm
	3 ft 3.4 in	116 ft 0.1 in	116 ft 0.1 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm²	Bq/cm²
Ba-137m	1.2400e-007	4.5879e+003	2.4800e-009	9.1760e-005
Co-57	6.8999e-011	2.5530e+000	1.3800e-012	5.1060e-008
Co-60	1.3150e-008	4.8654e+002	2.6300e-010	9.7310e-006
Cs-134	1.0250e-009	3.7924e+001	2.0500e-011	7.5850e-007
Cs-137	1.2400e-007	4.5879e+003	2.4800e-009	9.1760e-005
Fe-55	1.0850e-009	4.0144e+001	2.1700e-011	8.0290e-007
H-3	5.2999e-009	1.9610e+002	1.0600e-010	3.9220e-006
Ni-63	7.9998e-008	2.9599e+003	1.6000e-009	5.9200e-005
Sr-90	6.2999e-010	2.3310e+001	1.2600e-011	4.6620e-007
Y-90	6.2999e-010	2.3310e+001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.498e+01	8.134e-08	1.141e-07	6.775e-10	9.503e-10
0.0318	8.130e-02	6.962e-11	9.765e-11	5.799e-13	8.134e-13
0.0322	1.752e+02	1.525e-07	2.146e-07	1.227e-09	1.727e-09
0.0322	1.500e-01	1.305e-10	1.837e-10	1.050e-12	1.478e-12
0.0364	6.377e+01	6.513e-08	9.446e-08	3.701e-10	5.367e-10
0.0364	5.459e-02	5.575e-11	8.086e-11	3.168e-13	4.594e-13
0.1221	2.183e+00	8.364e-09	1.111e-08	1.311e-11	1.742e-11

Calc 016-01 CMY,
Rev 2, Att 2
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Page   : 2
DOS File: CC5000.MS5
Run Date: June 12, 2002
Run Time: 8:34:10 AM
Duration: 00:00:06

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	2.707e-01	1.165e-09	1.524e-09	1.874e-12	2.450e-12
0.2769	1.343e-02	1.210e-10	1.406e-10	2.269e-13	2.637e-13
0.4753	5.537e-01	8.767e-09	9.709e-09	1.720e-11	1.905e-11
0.536	7.318e-04	1.313e-11	1.442e-11	2.575e-14	2.828e-14
0.5632	3.178e+00	6.005e-08	6.572e-08	1.176e-10	1.287e-10
0.5693	5.852e+00	1.118e-07	1.223e-07	2.188e-10	2.393e-10
0.6047	3.701e+01	7.530e-07	8.204e-07	1.469e-09	1.601e-09
0.6616	4.128e+03	9.222e-05	9.991e-05	1.788e-07	1.937e-07
0.692	4.082e-03	9.555e-11	1.032e-10	1.845e-13	1.994e-13
0.6938	7.936e-02	1.863e-09	2.012e-09	3.596e-12	3.885e-12
0.7958	3.239e+01	8.766e-07	9.397e-07	1.668e-09	1.789e-09
0.8019	3.311e+00	9.032e-08	9.679e-08	1.717e-10	1.841e-10
1.0386	3.792e-01	1.353e-08	1.434e-08	2.477e-11	2.625e-11
1.1679	6.826e-01	2.750e-08	2.901e-08	4.919e-11	5.189e-11
1.1732	4.865e+02	1.969e-05	2.077e-05	3.519e-08	3.711e-08
1.3325	4.865e+02	2.247e-05	2.358e-05	3.898e-08	4.091e-08
1.3652	1.153e+00	5.458e-08	5.725e-08	9.413e-11	9.872e-11
TOTALS:	5.523e+03	1.367e-04	1.469e-04	2.591e-07	2.791e-07

CALC 016-01 (MY)
Rev 2, Att 2
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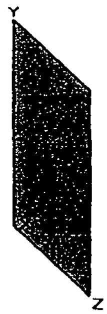
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC2500.MS5
Run Date: June 12, 2002
Run Time: 8:35:46 AM
Duration: 00:00:05

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: AF 2,500 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	5.0e+3 cm	164 ft 0.5 in
Height	5.0e+3 cm	164 ft 0.5 in

Dose Points

	X	Y	Z
# 1	100 cm	2500 cm	2500 cm
	3 ft 3.4 in	82 ft 0.3 in	82 ft 0.3 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	6.2000e-008	2.2940e+003	2.4800e-009	9.1760e-005
Co-57	3.4500e-011	1.2765e+000	1.3800e-012	5.1060e-008
Co-60	6.5750e-009	2.4328e+002	2.6300e-010	9.7310e-006
Cs-134	5.1250e-010	1.8963e+001	2.0500e-011	7.5850e-007
Cs-137	6.2000e-008	2.2940e+003	2.4800e-009	9.1760e-005
Fe-55	5.4250e-010	2.0073e+001	2.1700e-011	8.0290e-007
H-3	2.6500e-009	9.8050e+001	1.0600e-010	3.9220e-006
Ni-63	4.0000e-008	1.4800e+003	1.6000e-009	5.9200e-005
Sr-90	3.1500e-010	1.1655e+001	1.2600e-011	4.6620e-007
Y-90	3.1500e-010	1.1655e+001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	4.749e+01	7.729e-08	1.046e-07	6.438e-10	8.712e-10
0.0318	4.065e-02	6.616e-11	8.952e-11	5.511e-13	7.457e-13
0.0322	8.762e+01	1.448e-07	1.964e-07	1.166e-09	1.581e-09
0.0322	7.500e-02	1.240e-10	1.681e-10	9.977e-13	1.353e-12
0.0364	3.189e+01	6.155e-08	8.501e-08	3.497e-10	4.830e-10
0.0364	2.729e-02	5.269e-11	7.277e-11	2.994e-13	4.134e-13
0.1221	1.092e+00	7.766e-09	9.738e-09	1.218e-11	1.527e-11

Calc 016-01 (MY)
Rev 2, Att 2
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Page : 2
DOS File: CC2500.MS5
Run Date: June 12, 2002
Run Time: 8:35:46 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.353e-01	1.081e-09	1.338e-09	1.738e-12	2.152e-12
0.2769	6.713e-03	1.116e-10	1.257e-10	2.093e-13	2.359e-13
0.4753	2.769e-01	8.052e-09	8.732e-09	1.580e-11	1.713e-11
0.536	3.659e-04	1.205e-11	1.298e-11	2.363e-14	2.545e-14
0.5632	1.589e+00	5.508e-08	5.917e-08	1.078e-10	1.158e-10
0.5693	2.926e+00	1.025e-07	1.101e-07	2.007e-10	2.155e-10
0.6047	1.851e+01	6.903e-07	7.389e-07	1.347e-09	1.442e-09
0.6616	2.064e+03	8.448e-05	9.002e-05	1.638e-07	1.745e-07
0.692	2.041e-03	8.750e-11	9.303e-11	1.690e-13	1.797e-13
0.6938	3.968e-02	1.706e-09	1.813e-09	3.293e-12	3.501e-12
0.7958	1.619e+01	8.019e-07	8.473e-07	1.526e-09	1.613e-09
0.8019	1.655e+00	8.262e-08	8.727e-08	1.571e-10	1.660e-10
1.0386	1.896e-01	1.235e-08	1.293e-08	2.262e-11	2.368e-11
1.1679	3.413e-01	2.509e-08	2.617e-08	4.487e-11	4.681e-11
1.1732	2.433e+02	1.796e-05	1.873e-05	3.210e-08	3.348e-08
1.3325	2.433e+02	2.047e-05	2.127e-05	3.552e-08	3.691e-08
1.3652	5.765e-01	4.974e-08	5.164e-08	8.577e-11	8.906e-11
TOTALS:	2.761e+03	1.250e-04	1.324e-04	2.371e-07	2.516e-07

Calc 016-01 (MY)
Rev 2, Att 2
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MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC2000.MS5
Run Date: June 12, 2002
Run Time: 8:37:23 AM
Duration: 00:00:05

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: AF 2,000 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	4.5e+3 cm	146 ft 8.6 in
Height	4.5e+3 cm	146 ft 8.6 in

Dose Points

	X	Y	Z
# 1	100 cm	2236 cm	2236 cm
	3 ft 3.4 in	73 ft 4.3 in	73 ft 4.3 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	4.9597e-008	1.8351e+003	2.4800e-009	9.1760e-005
Co-57	2.7598e-011	1.0211e+000	1.3800e-012	5.1060e-008
Co-60	5.2597e-009	1.9461e+002	2.6300e-010	9.7310e-006
Cs-134	4.0998e-010	1.5169e+001	2.0500e-011	7.5850e-007
Cs-137	4.9597e-008	1.8351e+003	2.4800e-009	9.1760e-005
Fe-55	4.3397e-010	1.6057e+001	2.1700e-011	8.0290e-007
H-3	2.1199e-009	7.8435e+001	1.0600e-010	3.9220e-006
Ni-63	3.1998e-008	1.1839e+003	1.6000e-009	5.9200e-005
Sr-90	2.5198e-010	9.3234e+000	1.2600e-011	4.6620e-007
Y-90	2.5198e-010	9.3234e+000	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	3.799e+01	7.591e-08	1.014e-07	6.323e-10	8.443e-10
0.0318	3.252e-02	6.498e-11	8.676e-11	5.412e-13	7.227e-13
0.0322	7.009e+01	1.422e-07	1.902e-07	1.145e-09	1.531e-09
0.0322	6.000e-02	1.217e-10	1.628e-10	9.797e-13	1.310e-12
0.0364	2.551e+01	6.034e-08	8.194e-08	3.428e-10	4.655e-10
0.0364	2.183e-02	5.165e-11	7.014e-11	2.934e-13	3.985e-13
0.1221	8.732e-01	7.570e-09	9.333e-09	1.187e-11	1.463e-11

Calc 016-01 (MT)
Rev 2, Att 2
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Page : 2
DOS File: CC2000.MS5
Run Date: June 12, 2002
Run Time: 8:37:23 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.083e-01	1.053e-09	1.283e-09	1.694e-12	2.064e-12
0.2769	5.370e-03	1.086e-10	1.213e-10	2.036e-13	2.275e-13
0.4753	2.215e-01	7.824e-09	8.434e-09	1.535e-11	1.655e-11
0.536	2.927e-04	1.170e-11	1.254e-11	2.295e-14	2.459e-14
0.5632	1.271e+00	5.349e-08	5.717e-08	1.047e-10	1.119e-10
0.5693	2.341e+00	9.959e-08	1.064e-07	1.949e-10	2.082e-10
0.6047	1.481e+01	6.703e-07	7.139e-07	1.308e-09	1.393e-09
0.6616	1.651e+03	8.202e-05	8.700e-05	1.590e-07	1.687e-07
0.692	1.633e-03	8.494e-11	8.991e-11	1.640e-13	1.736e-13
0.6938	3.174e-02	1.656e-09	1.753e-09	3.197e-12	3.384e-12
0.7958	1.295e+01	7.782e-07	8.190e-07	1.481e-09	1.559e-09
0.8019	1.324e+00	8.018e-08	8.436e-08	1.525e-10	1.604e-10
1.0386	1.517e-01	1.198e-08	1.250e-08	2.194e-11	2.289e-11
1.1679	2.730e-01	2.433e-08	2.529e-08	4.351e-11	4.525e-11
1.1732	1.946e+02	1.742e-05	1.811e-05	3.113e-08	3.236e-08
1.3325	1.946e+02	1.985e-05	2.056e-05	3.443e-08	3.568e-08
1.3652	4.611e-01	4.821e-08	4.992e-08	8.314e-11	8.609e-11
TOTALS:	2.209e+03	1.214e-04	1.279e-04	2.301e-07	2.432e-07

Calc 016-01 (MY)
Rev 2, Att 2
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MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC1000.MS5
Run Date: June 12, 2002
Run Time: 10:25:27 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 1,000 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	3.2e+3 cm	103 ft 8.9 in
Height	3.2e+3 cm	103 ft 8.9 in

Dose Points

	X	Y	Z
# 1	100 cm	1581 cm	1581 cm
	3 ft 3.4 in	51 ft 10.4 in	51 ft 10.4 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	2.4796e-008	9.1744e+002	2.4800e-009	9.1760e-005
Co-57	1.3798e-011	5.1051e-001	1.3800e-012	5.1060e-008
Co-60	2.6295e-009	9.7293e+001	2.6300e-010	9.7310e-006
Cs-134	2.0496e-010	7.5837e+000	2.0500e-011	7.5850e-007
Cs-137	2.4796e-008	9.1744e+002	2.4800e-009	9.1760e-005
Fe-55	2.1696e-010	8.0276e+000	2.1700e-011	8.0290e-007
H-3	1.0598e-009	3.9213e+001	1.0600e-010	3.9220e-006
Ni-63	1.5997e-008	5.9190e+002	1.6000e-009	5.9200e-005
Sr-90	1.2598e-010	4.6612e+000	1.2600e-011	4.6620e-007
Y-90	1.2598e-010	4.6612e+000	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.899e+01	7.105e-08	9.075e-08	5.918e-10	7.559e-10
0.0318	1.626e-02	6.082e-11	7.768e-11	5.066e-13	6.470e-13
0.0322	3.504e+01	1.330e-07	1.701e-07	1.071e-09	1.369e-09
0.0322	3.000e-02	1.139e-10	1.456e-10	9.164e-13	1.172e-12
0.0364	1.275e+01	5.615e-08	7.224e-08	3.190e-10	4.104e-10
0.0364	1.092e-02	4.806e-11	6.183e-11	2.731e-13	3.513e-13
0.1221	4.365e-01	6.931e-09	8.168e-09	1.087e-11	1.281e-11

Calc 016-97 (MT)
Rev 2, Att 2
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Page : 2
DOS File: CC1000.MS5
Run Date: June 12, 2002
Run Time: 10:25:27 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	5.413e-02	9.639e-10	1.125e-09	1.550e-12	1.809e-12
0.2769	2.685e-03	9.887e-11	1.079e-10	1.855e-13	2.024e-13
0.4753	1.107e-01	7.100e-09	7.533e-09	1.393e-11	1.478e-11
0.536	1.463e-04	1.061e-11	1.120e-11	2.081e-14	2.197e-14
0.5632	6.355e-01	4.849e-08	5.109e-08	9.494e-11	1.000e-10
0.5693	1.170e+00	9.027e-08	9.508e-08	1.767e-10	1.861e-10
0.6047	7.402e+00	6.074e-07	6.383e-07	1.185e-09	1.245e-09
0.6616	8.255e+02	7.428e-05	7.780e-05	1.440e-07	1.508e-07
0.692	8.163e-04	7.690e-11	8.042e-11	1.485e-13	1.553e-13
0.6938	1.587e-02	1.499e-09	1.568e-09	2.894e-12	3.027e-12
0.7958	6.476e+00	7.039e-07	7.328e-07	1.340e-09	1.395e-09
0.8019	6.621e-01	7.252e-08	7.548e-08	1.379e-10	1.435e-10
1.0386	7.584e-02	1.082e-08	1.119e-08	1.981e-11	2.049e-11
1.1679	1.365e-01	2.196e-08	2.264e-08	3.927e-11	4.050e-11
1.1732	9.729e+01	1.572e-05	1.621e-05	2.809e-08	2.897e-08
1.3325	9.729e+01	1.790e-05	1.841e-05	3.106e-08	3.194e-08
1.3652	2.305e-01	4.348e-08	4.469e-08	7.498e-11	7.706e-11
TOTALS:	1.104e+03	1.098e-04	1.144e-04	2.082e-07	2.175e-07

CNC 016-02 (MY)
Rev 2, Att 2
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MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC500.MS5
Run Date: June 12, 2002
Run Time: 10:26:42 AM
Duration: 00:00:05

File Ref:
Date:
By: DR
Checked:

Case Title: AF 500 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	2.2e+3 cm	73 ft 4.3 in
Height	2.2e+3 cm	73 ft 4.3 in

Dose Points

	X	Y	Z
# 1	100 cm	1118 cm	1118 cm
	3 ft 3.4 in	36 ft 8.2 in	36 ft 8.2 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	1.2399e-008	4.5877e+002	2.4800e-009	9.1760e-005
Co-57	6.8996e-012	2.5528e-001	1.3800e-012	5.1060e-008
Co-60	1.3149e-009	4.8652e+001	2.6300e-010	9.7310e-006
Cs-134	1.0249e-010	3.7923e+000	2.0500e-011	7.5850e-007
Cs-137	1.2399e-008	4.5877e+002	2.4800e-009	9.1760e-005
Fe-55	1.0849e-010	4.0143e+000	2.1700e-011	8.0290e-007
H-3	5.2997e-010	1.9609e+001	1.0600e-010	3.9220e-006
Ni-63	7.9995e-009	2.9598e+002	1.6000e-009	5.9200e-005
Sr-90	6.2996e-011	2.3309e+000	1.2600e-011	4.6620e-007
Y-90	6.2996e-011	2.3309e+000	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.498e+00	6.506e-08	7.939e-08	5.420e-10	6.613e-10
0.0318	8.130e-03	5.569e-11	6.796e-11	4.639e-13	5.661e-13
0.0322	1.752e+01	1.218e-07	1.486e-07	9.799e-10	1.196e-09
0.0322	1.500e-02	1.042e-10	1.272e-10	8.388e-13	1.024e-12
0.0364	6.377e+00	5.116e-08	6.253e-08	2.907e-10	3.553e-10
0.0364	5.459e-03	4.379e-11	5.353e-11	2.488e-13	3.041e-13
0.1221	2.183e-01	6.228e-09	7.088e-09	9.765e-12	1.111e-11

CAlC 016-?? ????
Rev 2, Att 2
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Page : 2
DOS File: CC500.MS5
Run Date: June 12, 2002
Run Time: 10:26:42 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	2.707e-02	8.656e-10	9.780e-10	1.392e-12	1.573e-12
0.2769	1.342e-03	8.845e-11	9.477e-11	1.659e-13	1.778e-13
0.4753	5.537e-02	6.333e-09	6.637e-09	1.243e-11	1.302e-11
0.536	7.317e-05	9.461e-12	9.876e-12	1.855e-14	1.937e-14
0.5632	3.178e-01	4.322e-08	4.504e-08	8.462e-11	8.819e-11
0.5693	5.851e-01	8.045e-08	8.383e-08	1.575e-10	1.641e-10
0.6047	3.701e+00	5.411e-07	5.628e-07	1.056e-09	1.098e-09
0.6616	4.128e+02	6.615e-05	6.862e-05	1.282e-07	1.330e-07
0.692	4.082e-04	6.847e-11	7.094e-11	1.322e-13	1.370e-13
0.6938	7.936e-03	1.335e-09	1.383e-09	2.577e-12	2.670e-12
0.7958	3.239e+00	6.264e-07	6.466e-07	1.192e-09	1.231e-09
0.8019	3.311e-01	6.453e-08	6.660e-08	1.227e-10	1.266e-10
1.0386	3.792e-02	9.617e-09	9.873e-09	1.761e-11	1.808e-11
1.1679	6.826e-02	1.950e-08	1.998e-08	3.489e-11	3.574e-11
1.1732	4.865e+01	1.396e-05	1.431e-05	2.495e-08	2.557e-08
1.3325	4.865e+01	1.589e-05	1.625e-05	2.757e-08	2.819e-08
1.3652	1.153e-01	3.860e-08	3.944e-08	6.656e-11	6.801e-11
TOTALS:	5.522e+02	9.769e-05	1.010e-04	1.853e-07	1.919e-07

Calc [illegible] (MY)
Rev 2, Att 2
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MicroShield v5.05 (5.05-00105)
Stone & Webster

File Ref:
Date:
By: DR
Checked:

Page : 1
DOS File: CC300.MS5
Run Date: June 12, 2002
Run Time: 10:27:56 AM
Duration: 00:00:05

Case Title: AF 300 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	1.7e+3 cm	56 ft 9.9 in
Height	1.7e+3 cm	56 ft 9.9 in

Dose Points

	X	Y	Z
# 1	100 cm	866 cm	866 cm
	3 ft 3.4 in	28 ft 4.9 in	28 ft 4.9 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	7.4396e-009	2.7526e+002	2.4800e-009	9.1760e-005
Co-57	4.1398e-012	1.5317e-001	1.3800e-012	5.1060e-008
Co-60	7.8895e-010	2.9191e+001	2.6300e-010	9.7310e-006
Cs-134	6.1496e-011	2.2754e+000	2.0500e-011	7.5850e-007
Cs-137	7.4396e-009	2.7526e+002	2.4800e-009	9.1760e-005
Fe-55	6.5096e-011	2.4086e+000	2.1700e-011	8.0290e-007
H-3	3.1798e-010	1.1765e+001	1.0600e-010	3.9220e-006
Ni-63	4.7997e-009	1.7759e+002	1.6000e-009	5.9200e-005
Sr-90	3.7798e-011	1.3985e+000	1.2600e-011	4.6620e-007
Y-90	3.7798e-011	1.3985e+000	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	5.699e+00	5.996e-08	7.101e-08	4.995e-10	5.915e-10
0.0318	4.878e-03	5.132e-11	6.078e-11	4.275e-13	5.063e-13
0.0322	1.051e+01	1.122e-07	1.329e-07	9.027e-10	1.069e-09
0.0322	9.000e-03	9.602e-11	1.137e-10	7.727e-13	9.152e-13
0.0364	3.826e+00	4.699e-08	5.569e-08	2.670e-10	3.164e-10
0.0364	3.275e-03	4.022e-11	4.767e-11	2.285e-13	2.708e-13
0.1221	1.310e-01	5.672e-09	6.326e-09	8.894e-12	9.918e-12

Calc 076-01 (MK)
Rev 2, Att 2
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Page : 2
DOS File: CC300.MS5
Run Date: June 12, 2002
Run Time: 10:27:56 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.624e-02	7.881e-10	8.735e-10	1.267e-12	1.405e-12
0.2769	8.055e-04	8.034e-11	8.516e-11	1.507e-13	1.597e-13
0.4753	3.322e-02	5.743e-09	5.975e-09	1.127e-11	1.172e-11
0.536	4.390e-05	8.576e-12	8.892e-12	1.682e-14	1.744e-14
0.5632	1.907e-01	3.917e-08	4.056e-08	7.669e-11	7.942e-11
0.5693	3.511e-01	7.291e-08	7.549e-08	1.427e-10	1.477e-10
0.6047	2.221e+00	4.904e-07	5.069e-07	9.567e-10	9.889e-10
0.6616	2.477e+02	5.993e-05	6.181e-05	1.162e-07	1.198e-07
0.692	2.449e-04	6.202e-11	6.390e-11	1.198e-13	1.234e-13
0.6938	4.762e-03	1.209e-09	1.246e-09	2.334e-12	2.405e-12
0.7958	1.943e+00	5.671e-07	5.826e-07	1.079e-09	1.109e-09
0.8019	1.986e-01	5.842e-08	6.000e-08	1.111e-10	1.141e-10
1.0386	2.275e-02	8.701e-09	8.897e-09	1.593e-11	1.629e-11
1.1679	4.096e-02	1.764e-08	1.801e-08	3.155e-11	3.221e-11
1.1732	2.919e+01	1.263e-05	1.289e-05	2.257e-08	2.304e-08
1.3325	2.919e+01	1.437e-05	1.464e-05	2.493e-08	2.540e-08
1.3652	6.917e-02	3.490e-08	3.554e-08	6.018e-11	6.129e-11
TOTALS:	3.313e+02	8.845e-05	9.094e-05	1.678e-07	1.728e-07

Calc 016-02 (...)
Rev 2, Att 2
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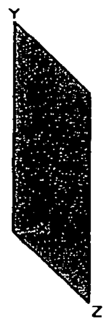
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC100.MS5
Run Date: June 12, 2002
Run Time: 10:29:07 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 100 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	1.0e+3 cm	32 ft 9.7 in
Height	1.0e+3 cm	32 ft 9.7 in

Dose Points

	X	Y	Z
# 1	100 cm	500 cm	500 cm
	3 ft 3.4 in	16 ft 4.9 in	16 ft 4.9 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	2.4800e-009	9.1760e+001	2.4800e-009	9.1760e-005
Co-57	1.3800e-012	5.1060e-002	1.3800e-012	5.1060e-008
Co-60	2.6300e-010	9.7310e+000	2.6300e-010	9.7310e-006
Cs-134	2.0500e-011	7.5850e-001	2.0500e-011	7.5850e-007
Cs-137	2.4800e-009	9.1760e+001	2.4800e-009	9.1760e-005
Fe-55	2.1700e-011	8.0290e-001	2.1700e-011	8.0290e-007
H-3	1.0600e-010	3.9220e+000	1.0600e-010	3.9220e-006
Ni-63	1.6000e-009	5.9200e+001	1.6000e-009	5.9200e-005
Sr-90	1.2600e-011	4.6620e-001	1.2600e-011	4.6620e-007
Y-90	1.2600e-011	4.6620e-001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.900e+00	4.758e-08	5.366e-08	3.963e-10	4.470e-10
0.0318	1.626e-03	4.073e-11	4.593e-11	3.392e-13	3.826e-13
0.0322	3.505e+00	8.896e-08	1.003e-07	7.160e-10	8.074e-10
0.0322	3.000e-03	7.615e-11	8.588e-11	6.128e-13	6.912e-13
0.0364	1.275e+00	3.708e-08	4.182e-08	2.107e-10	2.376e-10
0.0364	1.092e-03	3.174e-11	3.580e-11	1.803e-13	2.034e-13
0.1221	4.366e-02	4.413e-09	4.765e-09	6.919e-12	7.472e-12

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Page : 2
DOS File: CC100.MS5
Run Date: June 12, 2002
Run Time: 10:29:07 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	5.414e-03	6.128e-10	6.588e-10	9.853e-13	1.059e-12
0.2769	2.685e-04	6.223e-11	6.485e-11	1.167e-13	1.216e-13
0.4753	1.107e-02	4.436e-09	4.562e-09	8.705e-12	8.951e-12
0.536	1.464e-05	6.621e-12	6.792e-12	1.298e-14	1.332e-14
0.5632	6.356e-02	3.023e-08	3.099e-08	5.919e-11	6.067e-11
0.5693	1.170e-01	5.628e-08	5.767e-08	1.101e-10	1.129e-10
0.6047	7.403e-01	3.783e-07	3.873e-07	7.382e-10	7.556e-10
0.6616	8.257e+01	4.622e-05	4.724e-05	8.960e-08	9.158e-08
0.692	8.164e-05	4.782e-11	4.884e-11	9.236e-14	9.432e-14
0.6938	1.587e-03	9.323e-10	9.521e-10	1.800e-12	1.838e-12
0.7958	6.478e-01	4.370e-07	4.454e-07	8.318e-10	8.477e-10
0.8019	6.622e-02	4.502e-08	4.588e-08	8.561e-11	8.724e-11
1.0386	7.585e-03	6.697e-09	6.803e-09	1.226e-11	1.246e-11
1.1679	1.365e-02	1.357e-08	1.377e-08	2.428e-11	2.463e-11
1.1732	9.731e+00	9.717e-06	9.858e-06	1.737e-08	1.762e-08
1.3325	9.731e+00	1.105e-05	1.120e-05	1.917e-08	1.942e-08
1.3652	2.306e-02	2.683e-08	2.718e-08	4.627e-11	4.687e-11
TOTALS:	1.105e+02	6.816e-05	6.952e-05	1.294e-07	1.321e-07

Calc 016-01 (MY)
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MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC50.MS5
Run Date: June 12, 2002
Run Time: 10:31:55 AM
Duration: 00:00:05

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: AF 50 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	707.1 cm	23 ft 2.4 in
Height	707.1 cm	23 ft 2.4 in

Dose Points

	X	Y	Z
# 1	100 cm	353.6 cm	353.6 cm
	3 ft 3.4 in	11 ft 7.2 in	11 ft 7.2 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	1.2400e-009	4.5879e+001	2.4800e-009	9.1760e-005
Co-57	6.8999e-013	2.5530e-002	1.3800e-012	5.1060e-008
Co-60	1.3150e-010	4.8654e+000	2.6300e-010	9.7310e-006
Cs-134	1.0250e-011	3.7924e-001	2.0500e-011	7.5850e-007
Cs-137	1.2400e-009	4.5879e+001	2.4800e-009	9.1760e-005
Fe-55	1.0850e-011	4.0144e-001	2.1700e-011	8.0290e-007
H-3	5.2999e-011	1.9610e+000	1.0600e-010	3.9220e-006
Ni-63	7.9998e-010	2.9599e+001	1.6000e-009	5.9200e-005
Sr-90	6.2999e-012	2.3310e-001	1.2600e-011	4.6620e-007
Y-90	6.2999e-012	2.3310e-001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.498e-01	3.920e-08	4.324e-08	3.265e-10	3.602e-10
0.0318	8.130e-04	3.355e-11	3.701e-11	2.795e-13	3.083e-13
0.0322	1.752e+00	7.327e-08	8.082e-08	5.896e-10	6.505e-10
0.0322	1.500e-03	6.271e-11	6.918e-11	5.047e-13	5.568e-13
0.0364	6.377e-01	3.047e-08	3.361e-08	1.731e-10	1.909e-10
0.0364	5.459e-04	2.608e-11	2.877e-11	1.482e-13	1.634e-13
0.1221	2.183e-02	3.603e-09	3.835e-09	5.648e-12	6.013e-12

CRC 016-02 (MY)
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Page : 2
DOS File: CC50.MS5
Run Date: June 12, 2002
Run Time: 10:31:55 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	2.707e-03	5.001e-10	5.305e-10	8.041e-13	8.530e-13
0.2769	1.343e-04	5.070e-11	5.243e-11	9.511e-14	9.835e-14
0.4753	5.537e-03	3.610e-09	3.693e-09	7.083e-12	7.246e-12
0.536	7.318e-06	5.386e-12	5.499e-12	1.056e-14	1.078e-14
0.5632	3.178e-02	2.459e-08	2.509e-08	4.815e-11	4.912e-11
0.5693	5.852e-02	4.577e-08	4.669e-08	8.958e-11	9.138e-11
0.6047	3.701e-01	3.077e-07	3.136e-07	6.003e-10	6.119e-10
0.6616	4.128e+01	3.758e-05	3.826e-05	7.286e-08	7.416e-08
0.692	4.082e-05	3.888e-11	3.956e-11	7.509e-14	7.639e-14
0.6938	7.936e-04	7.579e-10	7.710e-10	1.463e-12	1.489e-12
0.7958	3.239e-01	3.552e-07	3.607e-07	6.761e-10	6.866e-10
0.8019	3.311e-02	3.659e-08	3.716e-08	6.958e-11	7.066e-11
1.0386	3.792e-03	5.441e-09	5.511e-09	9.962e-12	1.009e-11
1.1679	6.826e-03	1.102e-08	1.115e-08	1.972e-11	1.995e-11
1.1732	4.865e+00	7.892e-06	7.985e-06	1.410e-08	1.427e-08
1.3325	4.865e+00	8.973e-06	9.069e-06	1.557e-08	1.573e-08
1.3652	1.153e-02	2.179e-08	2.202e-08	3.757e-11	3.797e-11
TOTALS:	5.523e+01	5.541e-05	5.630e-05	1.052e-07	1.070e-07

[illegible]
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MicroShield v5.05 (5.05-00105)
Stone & Webster

File Ref:
Date:
By: DR
Checked:

Page : 1
DOS File: CC36.MS5
Run Date: June 12, 2002
Run Time: 10:33:04 AM
Duration: 00:00:05

Case Title: AF 36 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	600.0 cm	19 ft 8.2 in
Height	600.0 cm	19 ft 8.2 in

Dose Points

	X	Y	Z
# 1	100 cm	300 cm	300 cm
	3 ft 3.4 in	9 ft 10.1 in	9 ft 10.1 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	8.9280e-010	3.3034e+001	2.4800e-009	9.1760e-005
Co-57	4.9680e-013	1.8382e-002	1.3800e-012	5.1060e-008
Co-60	9.4680e-011	3.5032e+000	2.6300e-010	9.7310e-006
Cs-134	7.3800e-012	2.7306e-001	2.0500e-011	7.5850e-007
Cs-137	8.9280e-010	3.3034e+001	2.4800e-009	9.1760e-005
Fe-55	7.8120e-012	2.8904e-001	2.1700e-011	8.0290e-007
H-3	3.8160e-011	1.4119e+000	1.0600e-010	3.9220e-006
Ni-63	5.7600e-010	2.1312e+001	1.6000e-009	5.9200e-005
Sr-90	4.5360e-012	1.6783e-001	1.2600e-011	4.6620e-007
Y-90	4.5360e-012	1.6783e-001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	6.839e-01	3.518e-08	3.848e-08	2.931e-10	3.206e-10
0.0318	5.854e-04	3.012e-11	3.294e-11	2.509e-13	2.744e-13
0.0322	1.262e+00	6.576e-08	7.192e-08	5.292e-10	5.788e-10
0.0322	1.080e-03	5.629e-11	6.157e-11	4.530e-13	4.955e-13
0.0364	4.592e-01	2.732e-08	2.988e-08	1.552e-10	1.698e-10
0.0364	3.930e-04	2.339e-11	2.557e-11	1.329e-13	1.453e-13
0.1221	1.572e-02	3.222e-09	3.411e-09	5.052e-12	5.348e-12

CALC [illegible] (MI)
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Page : 2
DOS File: CC36.MS5
Run Date: June 12, 2002
Run Time: 10:33:04 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.949e-03	4.472e-10	4.720e-10	7.192e-13	7.589e-13
0.2769	9.666e-05	4.531e-11	4.672e-11	8.500e-14	8.764e-14
0.4753	3.987e-03	3.225e-09	3.292e-09	6.327e-12	6.460e-12
0.536	5.269e-06	4.811e-12	4.903e-12	9.435e-15	9.616e-15
0.5632	2.288e-02	2.196e-08	2.237e-08	4.300e-11	4.380e-11
0.5693	4.213e-02	4.088e-08	4.163e-08	8.001e-11	8.148e-11
0.6047	2.665e-01	2.748e-07	2.796e-07	5.362e-10	5.456e-10
0.6616	2.972e+01	3.356e-05	3.411e-05	6.507e-08	6.613e-08
0.692	2.939e-05	3.472e-11	3.527e-11	6.706e-14	6.812e-14
0.6938	5.714e-04	6.769e-10	6.875e-10	1.307e-12	1.327e-12
0.7958	2.332e-01	3.172e-07	3.217e-07	6.037e-10	6.122e-10
0.8019	2.384e-02	3.267e-08	3.314e-08	6.213e-11	6.301e-11
1.0386	2.731e-03	4.857e-09	4.914e-09	8.894e-12	8.998e-12
1.1679	4.915e-03	9.840e-09	9.946e-09	1.760e-11	1.779e-11
1.1732	3.503e+00	7.045e-06	7.121e-06	1.259e-08	1.273e-08
1.3325	3.503e+00	8.009e-06	8.087e-06	1.390e-08	1.403e-08
1.3652	8.301e-03	1.945e-08	1.963e-08	3.354e-11	3.386e-11
TOTALS:	3.976e+01	4.947e-05	5.020e-05	9.393e-08	9.538e-08

Calc 916-02 (MY)
Rev 2, Att 2
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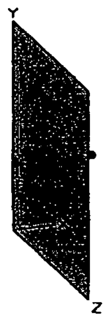
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC25.MS5
Run Date: June 12, 2002
Run Time: 10:34:39 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 25 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	500.0 cm	16 ft 4.9 in
Height	500.0 cm	16 ft 4.9 in

Dose Points

	X	Y	Z
# 1	100 cm	250 cm	250 cm
	3 ft 3.4 in	8 ft 2.4 in	8 ft 2.4 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm²	Bq/cm²
Ba-137m	6.2000e-010	2.2940e+001	2.4800e-009	9.1760e-005
Co-57	3.4500e-013	1.2765e-002	1.3800e-012	5.1060e-008
Co-60	6.5750e-011	2.4328e+000	2.6300e-010	9.7310e-006
Cs-134	5.1250e-012	1.8963e-001	2.0500e-011	7.5850e-007
Cs-137	6.2000e-010	2.2940e+001	2.4800e-009	9.1760e-005
Fe-55	5.4250e-012	2.0073e-001	2.1700e-011	8.0290e-007
H-3	2.6500e-011	9.8050e-001	1.0600e-010	3.9220e-006
Ni-63	4.0000e-010	1.4800e+001	1.6000e-009	5.9200e-005
Sr-90	3.1500e-012	1.1655e-001	1.2600e-011	4.6620e-007
Y-90	3.1500e-012	1.1655e-001	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	4.749e-01	3.077e-08	3.338e-08	2.563e-10	2.780e-10
0.0318	4.065e-04	2.634e-11	2.857e-11	2.194e-13	2.380e-13
0.0322	8.762e-01	5.750e-08	6.238e-08	4.628e-10	5.020e-10
0.0322	7.500e-04	4.922e-11	5.339e-11	3.961e-13	4.297e-13
0.0364	3.189e-01	2.387e-08	2.589e-08	1.356e-10	1.471e-10
0.0364	2.729e-04	2.043e-11	2.216e-11	1.161e-13	1.259e-13
0.1221	1.092e-02	2.808e-09	2.957e-09	4.403e-12	4.637e-12

Page : 2
DOS File: CC25.MS5
Run Date: June 12, 2002
Run Time: 10:34:39 AM
Duration: 00:00:05

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Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.353e-03	3.897e-10	4.092e-10	6.267e-13	6.580e-13
0.2769	6.713e-05	3.946e-11	4.057e-11	7.403e-14	7.611e-14
0.4753	2.769e-03	2.807e-09	2.860e-09	5.508e-12	5.612e-12
0.536	3.659e-06	4.187e-12	4.260e-12	8.212e-15	8.354e-15
0.5632	1.589e-02	1.912e-08	1.944e-08	3.743e-11	3.806e-11
0.5693	2.926e-02	3.558e-08	3.617e-08	6.964e-11	7.080e-11
0.6047	1.851e-01	2.392e-07	2.430e-07	4.666e-10	4.740e-10
0.6616	2.064e+01	2.921e-05	2.964e-05	5.662e-08	5.746e-08
0.692	2.041e-05	3.022e-11	3.065e-11	5.835e-14	5.919e-14
0.6938	3.968e-04	5.890e-10	5.974e-10	1.137e-12	1.153e-12
0.7958	1.619e-01	2.760e-07	2.795e-07	5.253e-10	5.320e-10
0.8019	1.655e-02	2.843e-08	2.879e-08	5.406e-11	5.475e-11
1.0386	1.896e-03	4.226e-09	4.270e-09	7.737e-12	7.819e-12
1.1679	3.413e-03	8.560e-09	8.643e-09	1.531e-11	1.546e-11
1.1732	2.433e+00	6.129e-06	6.188e-06	1.095e-08	1.106e-08
1.3325	2.433e+00	6.966e-06	7.028e-06	1.209e-08	1.219e-08
1.3652	5.765e-03	1.692e-08	1.706e-08	2.917e-11	2.942e-11
TOTALS:	2.761e+01	4.305e-05	4.362e-05	8.173e-08	8.288e-08

CWIC 016-02 (MY)
Rev 2, Att. 2
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MicroShield v5.05 (5.05-00105)
Stone & Webster

File Ref: ____
Date: ____
By: DR
Checked: ____

Page : 1
DOS File: CC16.MS5
Run Date: June 12, 2002
Run Time: 10:35:58 AM
Duration: 00:00:05

Case Title: AF 16 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	400.0 cm	13 ft 1.5 in
Height	400.0 cm	13 ft 1.5 in

Dose Points

	X	Y	Z
# 1	100 cm	200 cm	200 cm
	3 ft 3.4 in	6 ft 6.7 in	6 ft 6.7 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	3.9680e-010	1.4682e+001	2.4800e-009	9.1760e-005
Co-57	2.2080e-013	8.1696e-003	1.3800e-012	5.1060e-008
Co-60	4.2080e-011	1.5570e+000	2.6300e-010	9.7310e-006
Cs-134	3.2800e-012	1.2136e-001	2.0500e-011	7.5850e-007
Cs-137	3.9680e-010	1.4682e+001	2.4800e-009	9.1760e-005
Fe-55	3.4720e-012	1.2846e-001	2.1700e-011	8.0290e-007
H-3	1.6960e-011	6.2752e-001	1.0600e-010	3.9220e-006
Ni-63	2.5600e-010	9.4720e+000	1.6000e-009	5.9200e-005
Sr-90	2.0160e-012	7.4592e-002	1.2600e-011	4.6620e-007
Y-90	2.0160e-012	7.4592e-002	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	3.040e-01	2.552e-08	2.745e-08	2.126e-10	2.287e-10
0.0318	2.602e-04	2.185e-11	2.350e-11	1.820e-13	1.957e-13
0.0322	5.608e-01	4.769e-08	5.129e-08	3.838e-10	4.128e-10
0.0322	4.800e-04	4.082e-11	4.391e-11	3.285e-13	3.533e-13
0.0364	2.041e-01	1.978e-08	2.127e-08	1.124e-10	1.208e-10
0.0364	1.747e-04	1.693e-11	1.821e-11	9.619e-14	1.034e-13
0.1221	6.986e-03	2.321e-09	2.431e-09	3.639e-12	3.811e-12

Calc 016-92 (MT)
Rev 2, Att 2
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Page : 2
DOS File: CC16.MS5
Run Date: June 12, 2002
Run Time: 10:35:58 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	8.662e-04	3.221e-10	3.364e-10	5.179e-13	5.410e-13
0.2769	4.296e-05	3.259e-11	3.341e-11	6.113e-14	6.267e-14
0.4753	1.772e-03	2.317e-09	2.356e-09	4.546e-12	4.624e-12
0.536	2.342e-06	3.456e-12	3.510e-12	6.778e-15	6.883e-15
0.5632	1.017e-02	1.578e-08	1.601e-08	3.089e-11	3.135e-11
0.5693	1.873e-02	2.937e-08	2.980e-08	5.747e-11	5.833e-11
0.6047	1.184e-01	1.974e-07	2.002e-07	3.851e-10	3.906e-10
0.6616	1.321e+01	2.410e-05	2.442e-05	4.673e-08	4.734e-08
0.692	1.306e-05	2.493e-11	2.525e-11	4.815e-14	4.877e-14
0.6938	2.540e-04	4.860e-10	4.923e-10	9.384e-13	9.504e-13
0.7958	1.036e-01	2.277e-07	2.303e-07	4.334e-10	4.384e-10
0.8019	1.059e-02	2.346e-08	2.373e-08	4.461e-11	4.512e-11
1.0386	1.214e-03	3.486e-09	3.519e-09	6.383e-12	6.443e-12
1.1679	2.184e-03	7.060e-09	7.122e-09	1.263e-11	1.274e-11
1.1732	1.557e+00	5.055e-06	5.099e-06	9.034e-09	9.113e-09
1.3325	1.557e+00	5.746e-06	5.791e-06	9.969e-09	1.005e-08
1.3652	3.689e-03	1.395e-08	1.406e-08	2.406e-11	2.425e-11
TOTALS:	1.767e+01	3.552e-05	3.594e-05	6.744e-08	6.828e-08

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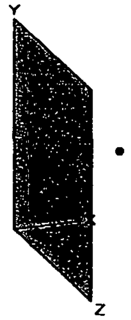
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC10.MS5
Run Date: June 12, 2002
Run Time: 10:36:56 AM
Duration: 00:00:05

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: AF 10 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	316.2 cm	10 ft 4.5 in
Height	316.2 cm	10 ft 4.5 in

Dose Points

	X	Y	Z
# 1	100 cm	158.1 cm	158.1 cm
	3 ft 3.4 in	5 ft 2.2 in	5 ft 2.2 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	2.4796e-010	9.1744e+000	2.4800e-009	9.1760e-005
Co-57	1.3798e-013	5.1051e-003	1.3800e-012	5.1060e-008
Co-60	2.6295e-011	9.7293e-001	2.6300e-010	9.7310e-006
Cs-134	2.0496e-012	7.5837e-002	2.0500e-011	7.5850e-007
Cs-137	2.4796e-010	9.1744e+000	2.4800e-009	9.1760e-005
Fe-55	2.1696e-012	8.0276e-002	2.1700e-011	8.0290e-007
H-3	1.0598e-011	3.9213e-001	1.0600e-010	3.9220e-006
Ni-63	1.5997e-010	5.9190e+000	1.6000e-009	5.9200e-005
Sr-90	1.2598e-012	4.6612e-002	1.2600e-011	4.6620e-007
Y-90	1.2598e-012	4.6612e-002	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.899e-01	2.032e-08	2.170e-08	1.693e-10	1.807e-10
0.0318	1.626e-04	1.739e-11	1.857e-11	1.449e-13	1.547e-13
0.0322	3.504e-01	3.796e-08	4.054e-08	3.055e-10	3.262e-10
0.0322	3.000e-04	3.250e-11	3.470e-11	2.615e-13	2.793e-13
0.0364	1.275e-01	1.573e-08	1.680e-08	8.939e-11	9.543e-11
0.0364	1.092e-04	1.347e-11	1.438e-11	7.652e-14	8.168e-14
0.1221	4.365e-03	1.842e-09	1.920e-09	2.888e-12	3.011e-12

Page : 2
DOS File: CC10.MS5
Run Date: June 12, 2002
Run Time: 10:36:56 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	5.413e-04	2.556e-10	2.658e-10	4.110e-13	4.275e-13
0.2769	2.685e-05	2.585e-11	2.643e-11	4.849e-14	4.959e-14
0.4753	1.107e-03	1.837e-09	1.865e-09	3.605e-12	3.660e-12
0.536	1.463e-06	2.740e-12	2.778e-12	5.373e-15	5.448e-15
0.5632	6.355e-03	1.251e-08	1.268e-08	2.449e-11	2.482e-11
0.5693	1.170e-02	2.328e-08	2.359e-08	4.556e-11	4.617e-11
0.6047	7.402e-02	1.565e-07	1.585e-07	3.053e-10	3.092e-10
0.6616	8.255e+00	1.911e-05	1.933e-05	3.704e-08	3.748e-08
0.692	8.163e-06	1.976e-11	1.999e-11	3.817e-14	3.861e-14
0.6938	1.587e-04	3.853e-10	3.897e-10	7.438e-13	7.524e-13
0.7958	6.476e-02	1.805e-07	1.824e-07	3.435e-10	3.471e-10
0.8019	6.621e-03	1.859e-08	1.878e-08	3.535e-11	3.572e-11
1.0386	7.584e-04	2.762e-09	2.786e-09	5.058e-12	5.101e-12
1.1679	1.365e-03	5.595e-09	5.639e-09	1.001e-11	1.009e-11
1.1732	9.729e-01	4.006e-06	4.037e-06	7.158e-09	7.214e-09
1.3325	9.729e-01	4.553e-06	4.585e-06	7.899e-09	7.955e-09
1.3652	2.305e-03	1.105e-08	1.113e-08	1.906e-11	1.920e-11
TOTALS:	1.104e+01	2.815e-05	2.845e-05	5.346e-08	5.406e-08

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MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC8.MS5
Run Date: June 12, 2002
Run Time: 10:38:20 AM
Duration: 00:00:05

File Ref:
Date:
By: DR
Checked:

Case Title: AF 8 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical

Source Dimensions

Width	282.8 cm	9 ft 3.3 in
Height	282.8 cm	9 ft 3.3 in

Dose Points

	X	Y	Z
# 1	100 cm	141.4 cm	141.4 cm
	3 ft 3.4 in	4 ft 7.7 in	4 ft 7.7 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	$\mu Ci/cm^2$	Bq/cm^2
Ba-137m	1.9834e-010	7.3386e+000	2.4800e-009	9.1760e-005
Co-57	1.1037e-013	4.0836e-003	1.3800e-012	5.1060e-008
Co-60	2.1034e-011	7.7824e-001	2.6300e-010	9.7310e-006
Cs-134	1.6395e-012	6.0662e-002	2.0500e-011	7.5850e-007
Cs-137	1.9834e-010	7.3386e+000	2.4800e-009	9.1760e-005
Fe-55	1.7355e-012	6.4213e-002	2.1700e-011	8.0290e-007
H-3	8.4774e-012	3.1367e-001	1.0600e-010	3.9220e-006
Ni-63	1.2796e-010	4.7346e+000	1.6000e-009	5.9200e-005
Sr-90	1.0077e-012	3.7285e-002	1.2600e-011	4.6620e-007
Y-90	1.0077e-012	3.7285e-002	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate $MeV/cm^2/sec$ No Buildup	Fluence Rate $MeV/cm^2/sec$ With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.519e-01	1.801e-08	1.918e-08	1.501e-10	1.598e-10
0.0318	1.300e-04	1.542e-11	1.642e-11	1.284e-13	1.367e-13
0.0322	2.803e-01	3.366e-08	3.583e-08	2.709e-10	2.884e-10
0.0322	2.399e-04	2.881e-11	3.067e-11	2.319e-13	2.469e-13
0.0364	1.020e-01	1.394e-08	1.484e-08	7.922e-11	8.433e-11
0.0364	8.732e-05	1.194e-11	1.270e-11	6.781e-14	7.218e-14
0.1221	3.492e-03	1.631e-09	1.697e-09	2.558e-12	2.661e-12

Calc 016-02 (MR)
Rev 2, Att. 2
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Page : 2
DOS File: CC8.MS5
Run Date: June 12, 2002
Run Time: 10:38:20 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	4.330e-04	2.263e-10	2.350e-10	3.639e-13	3.779e-13
0.2769	2.147e-05	2.288e-11	2.338e-11	4.293e-14	4.385e-14
0.4753	8.857e-04	1.626e-09	1.650e-09	3.191e-12	3.237e-12
0.536	1.170e-06	2.425e-12	2.457e-12	4.756e-15	4.819e-15
0.5632	5.083e-03	1.107e-08	1.121e-08	2.167e-11	2.195e-11
0.5693	9.360e-03	2.061e-08	2.087e-08	4.033e-11	4.084e-11
0.6047	5.921e-02	1.385e-07	1.402e-07	2.702e-10	2.735e-10
0.6616	6.603e+00	1.691e-05	1.710e-05	3.278e-08	3.315e-08
0.692	6.530e-06	1.749e-11	1.768e-11	3.378e-14	3.415e-14
0.6938	1.269e-04	3.410e-10	3.447e-10	6.583e-13	6.655e-13
0.7958	5.181e-02	1.597e-07	1.613e-07	3.040e-10	3.070e-10
0.8019	5.296e-03	1.645e-08	1.662e-08	3.129e-11	3.159e-11
1.0386	6.066e-04	2.444e-09	2.464e-09	4.476e-12	4.512e-12
1.1679	1.092e-03	4.951e-09	4.988e-09	8.856e-12	8.922e-12
1.1732	7.782e-01	3.545e-06	3.571e-06	6.334e-09	6.382e-09
1.3325	7.782e-01	4.029e-06	4.056e-06	6.989e-09	7.037e-09
1.3652	1.844e-03	9.781e-09	9.847e-09	1.687e-11	1.698e-11
TOTALS:	8.834e+00	2.492e-05	2.517e-05	4.731e-08	4.782e-08

Calc 916-01 (MY)
Rev 2, Att 2
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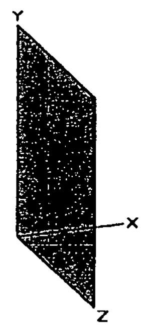
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC4.MS5
Run Date: June 12, 2002
Run Time: 10:39:24 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 4 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	200.0 cm	6 ft 6.7 in
Height	200.0 cm	6 ft 6.7 in

Dose Points

	X	Y	Z
# 1	100 cm	100 cm	100 cm
	3 ft 3.4 in	3 ft 3.4 in	3 ft 3.4 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	9.9200e-011	3.6704e+000	2.4800e-009	9.1760e-005
Co-57	5.5200e-014	2.0424e-003	1.3800e-012	5.1060e-008
Co-60	1.0520e-011	3.8924e-001	2.6300e-010	9.7310e-006
Cs-134	8.2000e-013	3.0340e-002	2.0500e-011	7.5850e-007
Cs-137	9.9200e-011	3.6704e+000	2.4800e-009	9.1760e-005
Fe-55	8.6800e-013	3.2116e-002	2.1700e-011	8.0290e-007
H-3	4.2400e-012	1.5688e-001	1.0600e-010	3.9220e-006
Ni-63	6.4000e-011	2.3680e+000	1.6000e-009	5.9200e-005
Sr-90	5.0400e-013	1.8648e-002	1.2600e-011	4.6620e-007
Y-90	5.0400e-013	1.8648e-002	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	7.599e-02	1.176e-08	1.243e-08	9.796e-11	1.036e-10
0.0318	6.504e-05	1.007e-11	1.064e-11	8.385e-14	8.864e-14
0.0322	1.402e-01	2.197e-08	2.323e-08	1.768e-10	1.869e-10
0.0322	1.200e-04	1.881e-11	1.988e-11	1.514e-13	1.600e-13
0.0364	5.102e-02	9.095e-09	9.613e-09	5.167e-11	5.462e-11
0.0364	4.367e-05	7.785e-12	8.229e-12	4.423e-14	4.675e-14
0.1221	1.746e-03	1.062e-09	1.100e-09	1.665e-12	1.724e-12

Calc 016-01 (MY)
Rev 2, Att 2
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Page : 2
DOS File: CC4.MS5
Run Date: June 12, 2002
Run Time: 10:39:24 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	2.166e-04	1.473e-10	1.523e-10	2.369e-13	2.449e-13
0.2769	1.074e-05	1.489e-11	1.517e-11	2.792e-14	2.846e-14
0.4753	4.430e-04	1.057e-09	1.071e-09	2.074e-12	2.101e-12
0.536	5.854e-07	1.577e-12	1.595e-12	3.092e-15	3.129e-15
0.5632	2.542e-03	7.197e-09	7.279e-09	1.409e-11	1.425e-11
0.5693	4.681e-03	1.340e-08	1.355e-08	2.622e-11	2.651e-11
0.6047	2.961e-02	9.003e-08	9.100e-08	1.756e-10	1.775e-10
0.6616	3.303e+00	1.099e-05	1.110e-05	2.131e-08	2.152e-08
0.692	3.266e-06	1.137e-11	1.148e-11	2.196e-14	2.217e-14
0.6938	6.349e-05	2.216e-10	2.238e-10	4.279e-13	4.321e-13
0.7958	2.591e-02	1.038e-07	1.047e-07	1.976e-10	1.993e-10
0.8019	2.649e-03	1.069e-08	1.079e-08	2.034e-11	2.051e-11
1.0386	3.034e-04	1.589e-09	1.600e-09	2.909e-12	2.930e-12
1.1679	5.461e-04	3.217e-09	3.239e-09	5.755e-12	5.793e-12
1.1732	3.892e-01	2.303e-06	2.319e-06	4.116e-09	4.144e-09
1.3325	3.892e-01	2.618e-06	2.633e-06	4.541e-09	4.569e-09
1.3652	9.223e-04	6.356e-09	6.393e-09	1.096e-11	1.102e-11
TOTALS:	4.418e+00	1.619e-05	1.634e-05	3.075e-08	3.104e-08

MicroShield v5.05 (5.05-00105)
Stone & Webster

Rev 2, Att 2
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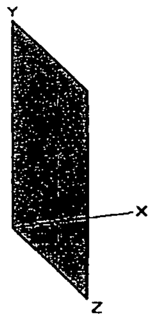
Page : 1
DOS File: CC3.MS5
Run Date: June 12, 2002
Run Time: 10:51:35 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 3 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	173.2 cm	5 ft 8.2 in
Height	173.2 cm	5 ft 8.2 in

Dose Points

	X	Y	Z
# 1	100 cm	86.6 cm	86.6 cm
	3 ft 3.4 in	2 ft 10.1 in	2 ft 10.1 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	7.4396e-011	2.7526e+000	2.4800e-009	9.1760e-005
Co-57	4.1398e-014	1.5317e-003	1.3800e-012	5.1060e-008
Co-60	7.8895e-012	2.9191e-001	2.6300e-010	9.7310e-006
Cs-134	6.1496e-013	2.2754e-002	2.0500e-011	7.5850e-007
Cs-137	7.4396e-011	2.7526e+000	2.4800e-009	9.1760e-005
Fe-55	6.5096e-013	2.4086e-002	2.1700e-011	8.0290e-007
H-3	3.1798e-012	1.1765e-001	1.0600e-010	3.9220e-006
Ni-63	4.7997e-011	1.7759e+000	1.6000e-009	5.9200e-005
Sr-90	3.7798e-013	1.3985e-002	1.2600e-011	4.6620e-007
Y-90	3.7798e-013	1.3985e-002	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	5.699e-02	9.622e-09	1.015e-08	8.015e-11	8.454e-11
0.0318	4.878e-05	8.236e-12	8.688e-12	6.860e-14	7.237e-14
0.0322	1.051e-01	1.797e-08	1.896e-08	1.447e-10	1.526e-10
0.0322	9.000e-05	1.539e-11	1.623e-11	1.238e-13	1.306e-13
0.0364	3.826e-02	7.439e-09	7.846e-09	4.227e-11	4.458e-11
0.0364	3.275e-05	6.368e-12	6.716e-12	3.618e-14	3.816e-14
0.1221	1.310e-03	8.678e-10	8.977e-10	1.361e-12	1.408e-12

CAlC 016-02 (MY)
Rev 2, Att 2
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Page : 2
DOS File: CC3.MS5
Run Date: June 12, 2002
Run Time: 10:51:35 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.624e-04	1.204e-10	1.243e-10	1.936e-13	1.999e-13
0.2769	8.055e-06	1.216e-11	1.239e-11	2.282e-14	2.324e-14
0.4753	3.322e-04	8.639e-10	8.746e-10	1.695e-12	1.716e-12
0.536	4.390e-07	1.288e-12	1.303e-12	2.526e-15	2.555e-15
0.5632	1.907e-03	5.881e-09	5.945e-09	1.151e-11	1.164e-11
0.5693	3.511e-03	1.095e-08	1.106e-08	2.142e-11	2.165e-11
0.6047	2.221e-02	7.356e-08	7.433e-08	1.435e-10	1.450e-10
0.6616	2.477e+00	8.981e-06	9.068e-06	1.741e-08	1.758e-08
0.692	2.449e-06	9.290e-12	9.377e-12	1.794e-14	1.811e-14
0.6938	4.762e-05	1.811e-10	1.828e-10	3.496e-13	3.529e-13
0.7958	1.943e-02	8.483e-08	8.554e-08	1.614e-10	1.628e-10
0.8019	1.986e-03	8.738e-09	8.811e-09	1.662e-11	1.675e-11
1.0386	2.275e-04	1.298e-09	1.307e-09	2.376e-12	2.393e-12
1.1679	4.096e-04	2.628e-09	2.645e-09	4.702e-12	4.732e-12
1.1732	2.919e-01	1.882e-06	1.894e-06	3.363e-09	3.384e-09
1.3325	2.919e-01	2.139e-06	2.151e-06	3.710e-09	3.732e-09
1.3652	6.917e-04	5.192e-09	5.222e-09	8.954e-12	9.005e-12
TOTALS:	3.313e+00	1.323e-05	1.335e-05	2.513e-08	2.536e-08

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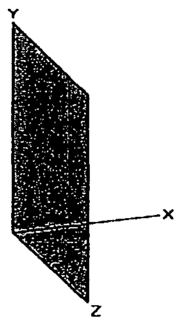
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC2.MS5
Run Date: June 12, 2002
Run Time: 10:59:23 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 2 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	141.4 cm	4 ft 7.7 in
Height	141.4 cm	4 ft 7.7 in

Dose Points

	X	Y	Z
# 1	100 cm	70.71 cm	70.71 cm
	3 ft 3.4 in	2 ft 3.8 in	2 ft 3.8 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	4.9585e-011	1.8346e+000	2.4800e-009	9.1760e-005
Co-57	2.7592e-014	1.0209e-003	1.3800e-012	5.1060e-008
Co-60	5.2584e-012	1.9456e-001	2.6300e-010	9.7310e-006
Cs-134	4.0988e-013	1.5165e-002	2.0500e-011	7.5850e-007
Cs-137	4.9585e-011	1.8346e+000	2.4800e-009	9.1760e-005
Fe-55	4.3387e-013	1.6053e-002	2.1700e-011	8.0290e-007
H-3	2.1194e-012	7.8416e-002	1.0600e-010	3.9220e-006
Ni-63	3.1990e-011	1.1836e+000	1.6000e-009	5.9200e-005
Sr-90	2.5192e-013	9.3212e-003	1.2600e-011	4.6620e-007
Y-90	2.5192e-013	9.3212e-003	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	3.798e-02	7.091e-09	7.462e-09	5.906e-11	6.215e-11
0.0318	3.251e-05	6.069e-12	6.387e-12	5.056e-14	5.320e-14
0.0322	7.008e-02	1.325e-08	1.394e-08	1.066e-10	1.122e-10
0.0322	5.999e-05	1.134e-11	1.193e-11	9.125e-14	9.602e-14
0.0364	2.550e-02	5.481e-09	5.766e-09	3.114e-11	3.276e-11
0.0364	2.183e-05	4.691e-12	4.936e-12	2.665e-14	2.804e-14
0.1221	8.730e-04	6.389e-10	6.599e-10	1.002e-12	1.035e-12

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Page : 2
DOS File: CC2.MS5
Run Date: June 12, 2002
Run Time: 10:59:23 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.082e-04	8.864e-11	9.138e-11	1.425e-13	1.469e-13
0.2769	5.369e-06	8.954e-12	9.110e-12	1.680e-14	1.709e-14
0.4753	2.214e-04	6.358e-10	6.433e-10	1.248e-12	1.262e-12
0.536	2.926e-07	9.481e-13	9.583e-13	1.859e-15	1.879e-15
0.5632	1.271e-03	4.328e-09	4.373e-09	8.473e-12	8.562e-12
0.5693	2.340e-03	8.055e-09	8.139e-09	1.576e-11	1.593e-11
0.6047	1.480e-02	5.413e-08	5.467e-08	1.056e-10	1.067e-10
0.6616	1.651e+00	6.609e-06	6.670e-06	1.281e-08	1.293e-08
0.692	1.632e-06	6.836e-12	6.897e-12	1.320e-14	1.332e-14
0.6938	3.174e-05	1.333e-10	1.345e-10	2.573e-13	2.596e-13
0.7958	1.295e-02	6.242e-08	6.292e-08	1.188e-10	1.197e-10
0.8019	1.324e-03	6.430e-09	6.481e-09	1.223e-11	1.232e-11
1.0386	1.517e-04	9.550e-10	9.613e-10	1.749e-12	1.760e-12
1.1679	2.730e-04	1.934e-09	1.946e-09	3.459e-12	3.480e-12
1.1732	1.946e-01	1.385e-06	1.393e-06	2.474e-09	2.489e-09
1.3325	1.946e-01	1.573e-06	1.582e-06	2.730e-09	2.745e-09
1.3652	4.610e-04	3.820e-09	3.841e-09	6.588e-12	6.624e-12
TOTALS:	2.208e+00	9.736e-06	9.817e-06	1.849e-08	1.865e-08

Calc 016-01 (MY)
Rev 2, Att 2
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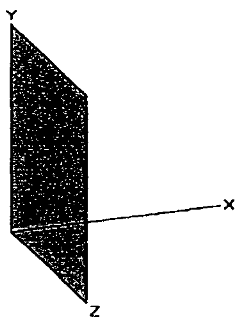
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC1.MS5
Run Date: June 12, 2002
Run Time: 11:00:09 AM
Duration: 00:00:05

File Ref: ________
Date: ________
By: DR
Checked: ________

Case Title: AF 1 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	100.0 cm	3 ft 3.4 in
Height	100.0 cm	3 ft 3.4 in

Dose Points

	X	Y	Z
# 1	100 cm	50 cm	50 cm
	3 ft 3.4 in	1 ft 7.7 in	1 ft 7.7 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm²	Bq/cm²
Ba-137m	2.4800e-011	9.1760e-001	2.4800e-009	9.1760e-005
Co-57	1.3800e-014	5.1060e-004	1.3800e-012	5.1060e-008
Co-60	2.6300e-012	9.7310e-002	2.6300e-010	9.7310e-006
Cs-134	2.0500e-013	7.5850e-003	2.0500e-011	7.5850e-007
Cs-137	2.4800e-011	9.1760e-001	2.4800e-009	9.1760e-005
Fe-55	2.1700e-013	8.0290e-003	2.1700e-011	8.0290e-007
H-3	1.0600e-012	3.9220e-002	1.0600e-010	3.9220e-006
Ni-63	1.6000e-011	5.9200e-001	1.6000e-009	5.9200e-005
Sr-90	1.2600e-013	4.6620e-003	1.2600e-011	4.6620e-007
Y-90	1.2600e-013	4.6620e-003	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	1.900e-02	3.996e-09	4.193e-09	3.329e-11	3.493e-11
0.0318	1.626e-05	3.421e-12	3.589e-12	2.849e-14	2.990e-14
0.0322	3.505e-02	7.465e-09	7.833e-09	6.008e-11	6.304e-11
0.0322	3.000e-05	6.390e-12	6.705e-12	5.142e-14	5.396e-14
0.0364	1.275e-02	3.088e-09	3.239e-09	1.754e-11	1.840e-11
0.0364	1.092e-05	2.643e-12	2.773e-12	1.502e-14	1.575e-14
0.1221	4.366e-04	3.596e-10	3.708e-10	5.639e-13	5.813e-13

C91C 016-01 (MY)
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Page : 2
DOS File: CC1.MS5
Run Date: June 12, 2002
Run Time: 11:00:09 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	5.414e-05	4.989e-11	5.135e-11	8.022e-14	8.257e-14
0.2769	2.685e-06	5.039e-12	5.122e-12	9.452e-15	9.608e-15
0.4753	1.107e-04	3.577e-10	3.617e-10	7.019e-13	7.098e-13
0.536	1.464e-07	5.334e-13	5.389e-13	1.046e-15	1.057e-15
0.5632	6.356e-04	2.435e-09	2.459e-09	4.767e-12	4.814e-12
0.5693	1.170e-03	4.532e-09	4.576e-09	8.869e-12	8.956e-12
0.6047	7.403e-03	3.046e-08	3.074e-08	5.942e-11	5.998e-11
0.6616	8.257e-01	3.718e-06	3.751e-06	7.208e-09	7.271e-09
0.692	8.164e-07	3.846e-12	3.879e-12	7.428e-15	7.490e-15
0.6938	1.587e-05	7.497e-11	7.561e-11	1.448e-13	1.460e-13
0.7958	6.478e-03	3.512e-08	3.538e-08	6.683e-11	6.734e-11
0.8019	6.622e-04	3.617e-09	3.645e-09	6.878e-12	6.930e-12
1.0386	7.585e-05	5.372e-10	5.406e-10	9.837e-13	9.898e-13
1.1679	1.365e-04	1.088e-09	1.094e-09	1.946e-12	1.957e-12
1.1732	9.731e-02	7.789e-07	7.834e-07	1.392e-09	1.400e-09
1.3325	9.731e-02	8.851e-07	8.897e-07	1.536e-09	1.544e-09
1.3652	2.306e-04	2.149e-09	2.160e-09	3.706e-12	3.725e-12
TOTALS:	1.105e+00	5.478e-06	5.521e-06	1.040e-08	1.049e-08

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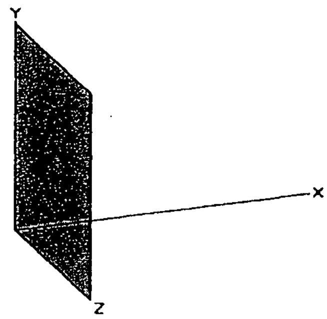
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC_5.MS5
Run Date: June 12, 2002
Run Time: 11:01:09 AM
Duration: 00:00:05

File Ref:
Date:
By: DR
Checked:

Case Title: AF 0.5 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	70.71 cm	2 ft 3.8 in
Height	70.71 cm	2 ft 3.8 in

Dose Points

	X	Y	Z
# 1	100 cm	35.36 cm	35.36 cm
	3 ft 3.4 in	1 ft 1.9 in	1 ft 1.9 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm²	Bq/cm²
Ba-137m	1.2400e-011	4.5879e-001	2.4800e-009	9.1760e-005
Co-57	6.8999e-015	2.5530e-004	1.3800e-012	5.1060e-008
Co-60	1.3150e-012	4.8654e-002	2.6300e-010	9.7310e-006
Cs-134	1.0250e-013	3.7924e-003	2.0500e-011	7.5850e-007
Cs-137	1.2400e-011	4.5879e-001	2.4800e-009	9.1760e-005
Fe-55	1.0850e-013	4.0144e-003	2.1700e-011	8.0290e-007
H-3	5.2999e-013	1.9610e-002	1.0600e-010	3.9220e-006
Ni-63	7.9998e-012	2.9599e-001	1.6000e-009	5.9200e-005
Sr-90	6.2999e-014	2.3310e-003	1.2600e-011	4.6620e-007
Y-90	6.2999e-014	2.3310e-003	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	9.498e-03	2.142e-09	2.245e-09	1.785e-11	1.870e-11
0.0318	8.130e-06	1.834e-12	1.921e-12	1.528e-14	1.600e-14
0.0322	1.752e-02	4.002e-09	4.193e-09	3.221e-11	3.374e-11
0.0322	1.500e-05	3.426e-12	3.589e-12	2.757e-14	2.888e-14
0.0364	6.377e-03	1.655e-09	1.734e-09	9.403e-12	9.850e-12
0.0364	5.459e-06	1.417e-12	1.484e-12	8.049e-15	8.431e-15
0.1221	2.183e-04	1.927e-10	1.984e-10	3.021e-13	3.111e-13

Calc 016-01 (MT)
Rev 2, Att 2
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Page : 2
DOS File: CC_5.MS5
Run Date: June 12, 2002
Run Time: 11:01:09 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	2.707e-05	2.673e-11	2.748e-11	4.298e-14	4.419e-14
0.2769	1.343e-06	2.699e-12	2.742e-12	5.063e-15	5.144e-15
0.4753	5.537e-05	1.916e-10	1.937e-10	3.760e-13	3.800e-13
0.536	7.318e-08	2.857e-13	2.885e-13	5.603e-16	5.659e-16
0.5632	3.178e-04	1.304e-09	1.317e-09	2.553e-12	2.578e-12
0.5693	5.852e-04	2.427e-09	2.450e-09	4.751e-12	4.796e-12
0.6047	3.701e-03	1.631e-08	1.646e-08	3.183e-11	3.211e-11
0.6616	4.128e-01	1.992e-06	2.008e-06	3.861e-09	3.893e-09
0.692	4.082e-07	2.060e-12	2.077e-12	3.978e-15	4.011e-15
0.6938	7.936e-06	4.016e-11	4.048e-11	7.753e-14	7.816e-14
0.7958	3.239e-03	1.881e-08	1.895e-08	3.580e-11	3.606e-11
0.8019	3.311e-04	1.937e-09	1.952e-09	3.684e-12	3.711e-12
1.0386	3.792e-05	2.877e-10	2.895e-10	5.268e-13	5.300e-13
1.1679	6.826e-05	5.826e-10	5.859e-10	1.042e-12	1.048e-12
1.1732	4.865e-02	4.172e-07	4.195e-07	7.455e-10	7.496e-10
1.3325	4.865e-02	4.740e-07	4.764e-07	8.224e-10	8.266e-10
1.3652	1.153e-04	1.151e-09	1.157e-09	1.985e-12	1.995e-12
TOTALS:	5.523e-01	2.934e-06	2.956e-06	5.571e-09	5.616e-09

[illegible]
Rev 2, Att 2
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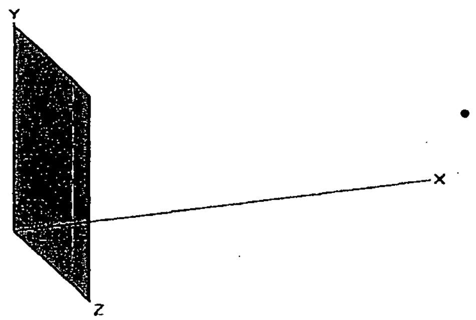
MicroShield v5.05 (5.05-00105)
Stone & Webster

Page : 1
DOS File: CC_25.MS5
Run Date: June 12, 2002
Run Time: 11:01:46 AM
Duration: 00:00:05

File Ref: ______
Date: ______
By: DR
Checked: ______

Case Title: AF 0.25 m^2
Description: Contaminated Concrete - MY nuclide mix
Geometry: 4 - Rectangular Area - Vertical



Source Dimensions

Width	50.0 cm	1 ft 7.7 in
Height	50.0 cm	1 ft 7.7 in

Dose Points

	X	Y	Z
# 1	100 cm	25 cm	25 cm
	3 ft 3.4 in	9.8 in	9.8 in

Shields

Shield Name	Material	Density
Air Gap	Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Ba-137m	6.2000e-012	2.2940e-001	2.4800e-009	9.1760e-005
Co-57	3.4500e-015	1.2765e-004	1.3800e-012	5.1060e-008
Co-60	6.5750e-013	2.4328e-002	2.6300e-010	9.7310e-006
Cs-134	5.1250e-014	1.8963e-003	2.0500e-011	7.5850e-007
Cs-137	6.2000e-012	2.2940e-001	2.4800e-009	9.1760e-005
Fe-55	5.4250e-014	2.0073e-003	2.1700e-011	8.0290e-007
H-3	2.6500e-013	9.8050e-003	1.0600e-010	3.9220e-006
Ni-63	4.0000e-012	1.4800e-001	1.6000e-009	5.9200e-005
Sr-90	3.1500e-014	1.1655e-003	1.2600e-011	4.6620e-007
Y-90	3.1500e-014	1.1655e-003	1.2600e-011	4.6620e-007

Buildup
The material reference is : Air Gap

Integration Parameters

Z Direction	20
Y Direction	20

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0318	4.749e-03	1.113e-09	1.165e-09	9.271e-12	9.704e-12
0.0318	4.065e-06	9.527e-13	9.972e-13	7.935e-15	8.307e-15
0.0322	8.762e-03	2.079e-09	2.176e-09	1.673e-11	1.751e-11
0.0322	7.500e-06	1.780e-12	1.863e-12	1.432e-14	1.499e-14
0.0364	3.189e-03	8.597e-10	8.997e-10	4.884e-12	5.112e-12
0.0364	2.729e-06	7.359e-13	7.702e-13	4.181e-15	4.376e-15
0.1221	1.092e-04	1.001e-10	1.030e-10	1.569e-13	1.615e-13

C4IC 016-01 (?)
Rev 2, Att 2
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Page : 2
DOS File: CC_25.MS5
Run Date: June 12, 2002
Run Time: 11:01:46 AM
Duration: 00:00:05

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.1365	1.353e-05	1.388e-11	1.427e-11	2.232e-14	2.294e-14
0.2769	6.713e-07	1.402e-12	1.424e-12	2.629e-15	2.670e-15
0.4753	2.769e-05	9.949e-11	1.006e-10	1.952e-13	1.973e-13
0.536	3.659e-08	1.484e-13	1.498e-13	2.909e-16	2.938e-16
0.5632	1.589e-04	6.772e-10	6.835e-10	1.326e-12	1.338e-12
0.5693	2.926e-04	1.260e-09	1.272e-09	2.467e-12	2.490e-12
0.6047	1.851e-03	8.470e-09	8.545e-09	1.653e-11	1.667e-11
0.6616	2.064e-01	1.034e-06	1.043e-06	2.005e-09	2.021e-09
0.692	2.041e-07	1.070e-12	1.078e-12	2.066e-15	2.082e-15
0.6938	3.968e-06	2.085e-11	2.102e-11	4.026e-14	4.058e-14
0.7958	1.619e-03	9.765e-09	9.836e-09	1.859e-11	1.872e-11
0.8019	1.655e-04	1.006e-09	1.013e-09	1.913e-12	1.926e-12
1.0386	1.896e-05	1.494e-10	1.503e-10	2.735e-13	2.752e-13
1.1679	3.413e-05	3.025e-10	3.042e-10	5.411e-13	5.441e-13
1.1732	2.433e-02	2.166e-07	2.178e-07	3.871e-10	3.892e-10
1.3325	2.433e-02	2.461e-07	2.473e-07	4.270e-10	4.291e-10
1.3652	5.765e-05	5.975e-10	6.005e-10	1.030e-12	1.035e-12
TOTALS:	2.761e-01	1.523e-06	1.535e-06	2.893e-09	2.915e-09

Attachment 6-16
Forebay Sediment Dose Assessment

(Has been replaced by Attachment 2H)

Attachment 6-17
Unitized Dose Factors for Activated Rebar

- Deleted -

Attachment 6-18
NRC Screening Levels for Contaminated Basement and Special Areas

Contaminated Basement Surfaces DCGL
For Building Occupancy NRC Screening Levels

Nuclide	Nuclide Fractions *nf*	Screening Level dpm/100 cm^2	Beta Fraction	*nf*/Screening Level
Sr-90	2.80E-03	8.71E+03	2.80E-03	3.22E-07
Cs-134	4.55E-03	1.27E+04	4.55E-03	3.58E-07
Cs-137	5.50E-01	2.80E+04	5.50E-01	1.97E-05
Co-60	5.84E-02	7.05E+03	5.84E-02	8.29E-06
Co-57	3.06E-04	2.11E+05		1.45E-09
Fe-55	4.81E-03	4.50E+06		1.07E-09
H-3	2.36E-02	1.24E+08		1.90E-10
Ni-63	3.55E-01	1.82E+06		1.95E-07
		sum	**6.16E-01**	**2.88E-05**

2.14E+04
dpm/100 cm^2
gross beta

NRC Screening Level Special Area Surfaces*

TRU values use RF_0 9.6E-07 m^{-1}

EC-011-01 2004 Mean nf	Nuclide	Pipe Tunnel & O/A Trench 2004 Mean nf	Screening Level dpm/100 cm^2	Beta Fraction	nf/Screening Level
4.028E-04	Mn-54	4.028E-04	3.15E+04		1.279E-08
2.235E-02	Fe-55	2.235E-02	4.50E+06		4.967E-09
3.639E-01	Co-60	3.639E-01	7.05E+03	3.639E-01	5.162E-05
3.024E-01	Ni-63	3.024E-01	1.82E+06		1.661E-07
6.874E-03	Sr-90	6.874E-03	8.71E+03	6.874E-03	7.892E-07
4.523E-03	Sb-125	4.523E-03	4.43E+04	4.523E-03	1.021E-07
2.815E-03	Cs-134	2.815E-03	1.27E+04	2.815E-03	2.216E-07
2.890E-01	Cs-137	2.890E-01	2.80E+04	2.890E-01	1.032E-05
1.165E-04	Pu-238	1.165E-04	4.24E+02		2.750E-07
8.752E-05	Pu-239	8.752E-05	3.85E+02		2.272E-07
8.750E-05	Pu-240	8.750E-05	3.85E+02		2.272E-07
6.705E-03	Pu-241	6.705E-03	1.97E+04		3.406E-07
5.929E-04	Am-241	5.929E-04	3.73E+02		1.591E-06
4.649E-05	Cm-243	4.649E-05	5.43E+02		8.555E-08
4.454E-05	Cm-244	4.454E-05	6.79E+02		6.556E-08
		detectable beta fraction ==>		6.672E-01	
			sum of nf/screening level ==>		6.605E-05

sum beta fraction divided by sum of nf divided by screening level ==> **1.010E+04**

detectable beta in dpm/100 cm^2

*Using 9.6E-07 resuspension factor for TRU's (from NUREG-1720)

Attachment 6-19
Special Areas Unitized Dose Factors

MAINE YANKEE

LTP SECTION 7

UPDATE OF SITE- SPECIFIC DECOMMISSIONING COSTS

TABLE OF CONTENTS

List of Tables

7.0 UPDATE OF SITE- SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

In accordance with 10 CFR 50.82(a)(9)(ii)(F) and the guidance of Regulatory Guide 1.179, the site-specific cost estimates and funding plans are provided. Regulatory Guide 1.179 discusses the details of the information to be presented.

The License Termination Plan (LTP) must:

Provide an estimate of the remaining decommissioning costs, and compare the estimated costs with the present funds set aside for decommissioning. The financial assurance instrument required by 10 CFR 50.75 must be funded to the amount of the cost estimate. If there is a deficit in the present funding, the LTP must indicate the means for ensuring adequate funds to complete the decommissioning.

Maine Yankee has previously submitted its Site-Specific Decommissioning Cost Estimate (Reference Letter: G. Zinke, Maine Yankee to USNRC; 10 CFR 50.82(a)(8)(iii) Site Specific Decommissioning Cost Estimate and PSDAR Update; MN-98-65, dated November 3, 1998). The report submitted with this letter, "Decommissioning Cost Analysis for the Maine Yankee Atomic Power Station" dated October 1997 provides a detailed analysis of the projected costs for decommissioning activities.

Regulatory Guide 1.179 requires that the decommissioning cost estimate in the LTP should include an evaluation of the following cost elements.

- Cost assumptions used, including a contingency factor
- Major decommissioning activities and tasks
- Unit cost factors
- Estimated costs of decontamination and removal of equipment and structures
- Estimated costs of waste disposal, including applicable disposal site surcharges
- Estimated final site survey costs
- Estimated total costs

The cost estimate should focus on the remaining work, detailed activity by activity including costs of labor, materials, equipment, energy, and services.

MYAPC has docketed a site-specific cost estimate prepared by TLG Services in accordance with 10 CFR 50.82(a)(8)(iii). (Reference Letter: George Zinke, MYAPC to USNRC; 10 CFR 50.82(a)(8)(iii) Site Specific Decommissioning Cost Estimate and PSDAR Update; MN-98-65, dated November 3, 1998). This TLG Cost Estimate focuses on all decommissioning costs from 1997 through 2023, with the assumed final removal of all fuel from the site.

Maine Yankee has received an order from the Federal Energy Regulatory Commission (FERC), dated June 1, 1999 and effective August 1, 1999, concerning the recovery of decommissioning costs.

The Nuclear Regulatory Commission (NRC) staff, in its initial acceptance review of the Maine Yankee License Termination Plan, requested that Maine Yankee provide an updated Site Specific Cost Estimate using the methodology of the 10 CFR 50.82(a)(8)(iii). Maine Yankee declines to expend the financial and schedule resources to provide an updated estimate formatted to the elements discussed above (e.g. unit cost factors, estimated costs of removal of equipment, estimated waste disposal costs for major commodities, etc.) for the following reasons.

The techniques used for projecting future costs of decommissioning in Maine Yankee's 1997 TLG Site Specific Cost Estimate submittal (e.g. unit cost factors) are no longer relevant or meaningful for a project over 50% complete.

Maine Yankee has completed the initial radiological site characterization and an extensive radiologically contaminated asbestos removal program. Funds for these activities have been expended and are included in the expenditures included in Table 7-2, column (2). Extensive radiologically contaminated commodity removal has been accomplished since the start of decommissioning activities, including the removal of the three steam generators, pressurizer and reactor coolant system piping. These and many other components have been shipped to the GTS Duratek facility in Memphis, TN for decontamination and disposal. As with the asbestos removal program, funds for these activities have also been included in the expenditures included in Table 7-2, column (2).

The segmentation of the reactor vessel internals was completed in 2001 and most of the associated costs were expended in 2001. The preparation for the shipment of the reactor vessel with some internal components to the Barnwell facility are currently underway.

The cost of removal, preparation, and disposal of the reactor vessel is fixed by existing contracts. Those internals components classified as greater-than-Class C waste have been packaged and moved to the on-site (ISFSI) storage facility. Shipments of Class A waste to the Envirocare facility in Utah have been ongoing over the last 36 months. Maine Yankee has commissioned an on-site rail loading facility to facilitate the rail shipment of bulk commodities to Utah and of non-radiological bulk waste (primarily concrete) to other disposal facilities.

Because of the progress of the Maine Yankee decommissioning effort to date, the current financial planning cost estimate (dated January 2002) includes actual expenditures to date, expected future expenditures associated with fixed price and other contracts, and estimates based on more detailed knowledge of cost than previously available.

Maine Yankee's current financial planning cost estimate of January 2002 is used to demonstrate financial assurance in Section 7.3. This cost estimate includes costs incurred since 1997 for the above activities plus the projected costs through 2023. The impact of this cost estimate on DTF balances is summarized in Table 7-2.

Finally, Maine Yankee has used its current financial planning cost estimate to test incurred or projected costs against the information presented in the TLG Site Specific Cost Estimate. A recent comparison is summarized in Table 7-3.

7.2 Decommissioning Cost Estimate

7.2.1 Cost Estimate Previously Docketed in Accordance with 10 CFR 50.82

As stated earlier, Maine Yankee has docketed a site-specific cost estimate prepared by TLG Services in accordance with 10 CFR 50.82(a)(8)(iii).

This section provides the result of and the basis for the 1997 TLG cost estimate. The TLG cost estimate was prepared using unit cost factors and site specific and schedule driven considerations in accordance with the methodology suggested in AIF/NESP-036, "Guidelines to Producing Decommissioning Cost Estimates." PSDAR Page 15, revision 1, summarizes decommissioning costs and is appended to this report as Table 7-1. This table presents costs derived from the TLG estimate but organized to reflect an estimated allocation of components in accordance with Reg. Guide 1.179 guidance.

As stated earlier, Maine Yankee's current financial planning decommissioning cost estimate dated January 2002 is used to demonstrate financial assurance and is the basis for Table 7-2 (col. 2 and col. 6). This current estimate includes costs to dismantle and decontaminate the plant, plus budgets for contingency, remediation and ISFSI engineering, licensing, construction and operation. These costs, totaling $705 M, are presented in nominal dollars. $311.9 M of expenditures have been accrued through 2001 for all decommissioning costs.

The projected costs presented in Table 7-2, and the balances in the Decommissioning trust funds, are updated periodically as actual expenditures are incurred.

Maine Yankee was recently awarded $44M for settlement of performance and payment bonds in connection with the decommissioning operations contract with Stone & Webster. Maine Yankee deposited the payment in its decommissioning trust fund and the payment is included in the revised Table 7-2 column 5. Maine Yankee is continuing to pursue its claim for damages that was originally filed against Stone & Webster and its parent corporations in August 2000 in the Bankruptcy Court in Delaware.

The current financial planning cost estimate of $635M in 2001 dollars, summarized in Table 7-2 ($705M nominal dollars (col. 2 and col. 6)), is consistent with the 1997 TLG report of $508M in mid-1997 dollars or $589M in 2001 dollars when escalated at 3.8%, as shown in Table 7-1.

Therefore, Maine Yankee continues to rely on the TLG cost estimate for this submittal since it meets the requirements for format and methodology discussed in AIF/NESP-036, and will continue to monitor future estimates to ensure that costs are within 20% of the TLG estimate.

Maine Yankee recognizes that certain assumptions of the TLG estimate are no longer applicable, but has continually affirmed that the overall costs of decommissioning remain within the margin of the estimate and within the contingency assumed in the estimate.

The balance of information for this section provides the result of and basis for the 1997 TLG Site Specific Cost Estimate.

7.2.2 Radiological Decontamination Costs

Based on the TLG estimate, the costs for radiological decontamination activities are estimated to be $343.3M in mid-1997 dollars or $398.5M in 2001 dollars when escalated at 3.8%, and are summarized in Table 7-1. Consistent with current NRC policy, Maine Yankee decontamination costs consider only those costs that are associated with normal decommissioning activities necessary for termination of the Part 50 license and release of the site for unrestricted use. This cost estimate remains valid for the enhanced state clean-up standards which are more restrictive than 10 CFR 20.1402 and the use of MARSSIM methodology for performing Final Status Surveys. It does not include costs associated with spent fuel management or the disposal of non-radioactive materials and structures beyond that necessary to terminate the Part 50 license.

Concrete demolition debris is classified as special waste in accordance with Maine's Hazardous Waste, Seepage, and Solid Waste Management Act (38 MRSA, section 1301. et. seq.). A percentage of the concrete to be removed may be slightly contaminated with radioactive nuclides. Radiologically contaminated concrete materials will be shipped off site for disposal at a LLRW disposal facility or other appropriate disposal facility. Consequently the waste volumes estimated in Table 5.1 of the TLG Report, "Decommissioning Cost Analysis for the Maine Yankee Atomic Power Station", would be increased. The incremental cost increase associated with additional burial volumes is tempered by the reduction in the Final Status Survey scope of work, the expanded use of an existing rail line servicing the site for bulk shipments, the use of the volume reduction technology by Duratek's facility in Memphis, and the use of the Envirocare disposal facility for the disposal of the bulk of Class A low level waste rather than the use of the Barnwell facility for all Class A waste.

Maine Yankee projects the resulting incremental cost of increased waste disposal which supports site decontamination to the enhanced state of Maine cleanup standards will be within the costs and contingency identified in Table 7-1.

For comparison purposes, Table 7-3 summarizes actual costs through 2001 and remaining projected Radiological Decommissioning costs. The format of Table 7-3 and 7-1 are reflective of the definition of decommissioning activities defined in 10 CFR 50.2 and is based on an estimated allocation of such components.

7.2.3 Spent Fuel Management Costs

Maine Yankee acknowledges that the costs to construct and operate an independent spent fuel storage installation (ISFSI) and other spent fuel related costs are not considered by the NRC staff as part of decommissioning costs. Nevertheless a presentation of those costs is required because other stakeholders, recognized by the NRC as legitimate participants in the decommissioning and license termination proceedings, do not subscribe to the definition of "decommissioning costs" delineated in 10 CFR 50.75(c) with footnote. Also the staff recognized, as discussed in 10 CFR 50.75(a), that funding for the decommissioning of power reactors may be subject to the jurisdiction of other Federal and State agencies.

In order to satisfy other stakeholders in the decommissioning process, spent fuel management costs, based on the TLG estimate, are separately summarized in Table 7-1 and estimated to be $128.7M in 1997 dollars or $149.4M in 2001 dollars when escalated at 3.8%. These costs include ISFSI engineering, licensing, construction, and operation until possession of the spent fuel is transferred to the Department of Energy (DOE) which is assumed in this estimate to begin in 2018. The cost of decommissioning the ISFSI facility is included in the fuel management costs.

7.2.4 Site Restoration (Remediation)

As discussed in Section 7.2.3, Maine Yankee recognizes that site restoration costs, including the treatment of non-radiological wastes (primarily concrete and structural materials) may be considered outside the scope of 10 CFR 50.75 and 10 CFR 50.82(a)(9)(ii)(F). The following information is provided in deference to other stakeholder requirements.

The cost of site restoration, based on the TLG estimate, is estimated to be $35.7M in mid-1997 dollars or $41.4M in 2001 dollars when escalated at 3.8% and as shown in Table 7-1. This cost includes demolition of non-radiological affected buildings and costs associated with non-radiological remediation required by Federal and State agencies, e.g., Resource Conservation and Recovery Act (RCRA) closure, asbestos disposal, etc..

Based on extensive input from State regulatory agencies, the Community Advisory Panel, and other key Stakeholders, the concrete waste from plant areas outside of the radiologically controlled area will be disposed of at commercial facilities licensed or permitted to handle special waste as defined by the State of Maine or to out of state facilities, provided out-of-State disposal is cost effective relative to other disposal options. This waste form also is to be shipped primarily by rail.

Table 7-1
Maine Yankee Summary of Decommissioning Costs (1)
TLG Site Specific Cost Estimate

Plant Radiological Decontamination	**1997 Dollars**	**2001 Dollars(4)**
Staffing	$91,128	$105,789
LLW Burial	$64,816	$75,244
Equipment Removal	$44,310	$51,439
LLW Packaging and Shipping	$16,663	$19,344
Decontamination Activities	$ 6,376	$ 7,402
Contingency	$60,265	$69,961
Other Costs(2)	$59,719	$69,327
Subtotal	**$343,279 (3) (5)**	**$398,505 (3)**
Spent Fuel Management		
Staffing and Security	$33,189	$38,529
Property Taxes	$25,445	$29,539
Construction Costs	$52,249	$60,655
NRC and State Fees	$10,093	$11,717
Insurance	$ 3,018	$3,504
Other Costs(2)	$ 4,683	$5,436
Subtotal	**$128,677 (3) (5)**	**$149,379(3)**
Site Restoration (Remediation)		
Licensing Termination Survey	$10,701	$12,423
Major Component Removal	$10,805	$12,543
Close-out activities	$ 3,222	$3,740
Demolition of site buildings	$10,973	$12,738
Subtotal	**$35,701 (3) (5)**	**$41,445 (3)**
Total Decommissioning Costs Estimate	**$508,000 (3) (5)**	**$589,329(3)**

Notes:
(1) Prompt Decommissioning Technique (DECON), costs in thousands of dollars. Components estimated to reflect allocation of cost categories to conform with NRC definition.
(2) Other costs include insurance, property taxes, energy, NRC and State fees, etc.
(3) Sums may not be exact due to rounding to nearest thousand
(4) Escalation rate of 3.8% used
(5) All values derived from the TLG Site Specific Cost Estimate as discussed in LTP Section 7.2

7.2.5 Summary of Maine Yankee Decommissioning Cost Estimate

The "Decommissioning Cost Analysis for the Maine Yankee Atomic Power Station"(Site Specific Decommissioning Cost Estimate prepared by TLG Services), is the basis for the company's decommissioning cost estimate and was provided in a format consistent with regulatory guidance with additional detail. Maine Yankee will continue to monitor actual costs to ensure future financial planning cost estimates, which include a significant portion of fixed costs and contain a greater scope than the TLG cost estimate, continue to be within 20% of the TLG estimate.

Finally, Maine Yankee has used its current January 2002 financial planning cost estimate of $635M in 2001 dollars (which exceeds the formal TLG cost estimate of $589 M in 2001$), which includes dismantlement and decommissioning, spent fuel construction and management costs, site restoration, and remaining Maine Yankee projected decommissioning costs through 2023, to project Decommissioning Trust Fund (DTF) balances and to demonstrate financial assurance. The projections are presented in Table 7-2.
The projections presented in Table 7-2 include the expenditures for waste disposal and decontamination in accordance with the enhanced state clean-up standards that are more restrictive than 10 CFR 20.1402.

7.3 Decommissioning Funding Plan

As stated above, Maine Yankee has used its January 2002 financial planning estimate of $635M (2001$) and projections of decommissioning collections which consider the 1999 FERC rate case settlement to project DTF balances and to evidence financial assurance along with other funding avenues available to the Company as described below. (See Table 7-2.).

Table 7-2 column 2 combines the annual projections for the costs specifically associated with plant (radiological) dismantlement, spent fuel management, and site-restoration as "Escalated Expenditures."

Maine Yankee is currently collecting decommissioning funds through its Power Contracts and Amendatory Agreements under FERC regulation. These contracts have been filed with the FERC. Table 7-2 column 1 identifies the decommissioning funds currently being collected and those projected to be collected under the contracts and includes the funding of radiological decommissioning, spent fuel management, and remediation.

Table 7-2

Maine Yankee Decommissioning Trust Summary for the Current Financial Planning Estimate

Decommissioning Cost $635M (2001$) @ 3.8%: Assumed New Rate Filing 1/1/04

(Dollars in Thousands)

	DECOMMISSIONING TRUST							
Year	(Col. 1) Annual Decommissioning Accrued Contributions	(Col. 2) Escalated Expenditures*	(Col. 3) After-Tax Trust Earnings and Adjustments	(Col. 4) Funding Per Section 7.4	(Col. 5) Accrued Decommissioning Trust Balance		(Col. 6) ISFSI Expenditures From SPENT FUEL DISPOSAL TRUST	
1997		$1,965			$199,457		$0	
1998	33,901	40,441	19,746		212,664		0	
1999	34,051	64,568	5,561		187,708		6,613	
2000	27,709	70,669	11,489		156,236		22,355	
2001	69,577	77,242	8,521		153,887	(1)	28,019	(2)
2002	25,577	93,875	3,737		89,326		15,973	
2003	25,577	61,954	2,163	0	55,112		4,407	
2004	21,627	36,149	1,455	0	42,045	(3)(4)	0	
2005	21,627	22,758	1,261	0	42,174		0	
2006	21,627	6,492	2,238	0	59,548		0	
2007	21,627	6,261	3,025		77,939		0	
2008	18,022	6,477	3,767		93,251		0	
2009	0	5,385	4,075		91,940			
2010	0	5,585	4,012		90,367			
2011	0	5,792	3.936		88,512			
2012	0	6,007	3,848		86,353			
2013	0	6,230	3,746		83,868			
2014	0	6,462	3,629		81,034			
2015	0	6,703	3,496		77,827			
2016	0	6,953	3,346		74,220			
2017	0	7,213	3,178		70,184			
2018	0	8,432	2,969		64,721			
2019	0	9,726	2,694		57,689			
2020	0	10,092	2,369		49,966			
2021	0	9,206	2,041		42,802			
2022	0	18,598	1,508		25,711			
2023	0	26,277	566		0			
Total	$320,923	$627,512	$108,373				$77,368	

* Excludes ISFSI-related expenditures

Notes:

Expenditures (columns 2 and 6) represent the current financial planning decommissioning cost estimate as of January 2002. Balances (columns 1, 2, and 5) include amounts for site restoration and long term spent fuel storage management.

(1) The Decommissioning Trust Fund Balance as of December 31, 2001 was $ 157.1M which included a $44 million accrual for settlement of performance and payment bonds and a $3.2 million accrual for unrealized gains. As of December 31, 2001, $311.9M had been expended for all decommissioning costs.

(2) The Spent Fuel Disposal Trust Fund Balance as of December 31, 2001 was $88.7M.

(3) Includes a reserve for SAFSTOR as discussed in Section 7.4

(4) Assumes annual decommissioning collections decrease from $25.6M to $21.6M with approval effective 1/1/04. Tax Billing Change (Column 3) where Trust pays for all decommissioning income taxes commencing 1/1/04.

As a result of the FERC order dated June 1, 1999 and effective August 1, 1999, Maine Yankee has agreed to file with the FERC no later than January 1, 2004 for the purpose of examining any further rate adjustments specifically, although not limited to the future cost of spent fuel storage management. Maine Yankee expects that case to determine any adjustments to decommissioning collections. Based on Maine Yankee's current financial planning estimate dated January 2002, the Company plans to fully fund all decommissioning costs and spent fuel storage costs by 2008 and would require a decrease in annual decommissioning collections from approximately $26M to approximately $22M given current assumptions. Maine Yankee will evaluate its revenue requirements for the appropriate rate adjustment prior to the 2004 filing.

As a result of State of Maine Legislative action effective September 18, 1999, Maine Yankee has access to its state-mandated Spent Fuel Disposal Trust (SFDT). As of December 31, 2001 the SFDT balance was $88,748,000. This Trust is separate and distinct from the DTF pursuant to 10 CFR 50.75 and 10 CFR 50.82. Effective October 1, 1999, Maine Yankee is permitted by State law to withdraw funds from the SFDT to meet expenditures for interim spent fuel storage costs and to offset those interim spent fuel storage costs already incurred by Maine Yankee. Expenditures from the SFDT are incorporated in the FERC Rate Settlement. Table 7-2 column 6 identifies the estimated costs associated with the construction and placing into service of an ISFSI which will be funded from the SFDT.

As of December 31, 2001, the accrued MY Decommissioning Trust Balance was $157.1 million. This balance includes amounts in the trust for all decommissioning costs including remediation and long term spent fuel management as well as decommissioning as defined in 10 CFR 50.75 and the PSDAR. The balance also includes an accrual for the performance and payment bonds settlement of $44 million.

Note that as of December 31, 2001, Maine Yankee had incurred $311.9 million of decommissioning expenditures, which includes $59M accrued for ISFSI construction.

Maine Yankee recognizes that the staff does not consider the cost to construct and operate the ISFSI and other spent fuel-related costs as part of the decommissioning cost, nor does it consider the cost to complete all environmental restoration activities at the site as part of the decommissioning cost estimate. However, Table 7-2 includes all such costs, including contingency.

As indicated in Table 7-2, column 5, the DTF pursuant to 10 CFR 50.75 is sufficient, together with current FERC-approved collections and an assumed rate decrease in 2004, to cover all of the expenditures related to decommissioning. Refer to Section 7.4 for a description of additional financing options, if necessary, available to Maine Yankee.

7.4 Reserve Requirements

10 CFR 50.82(8)(i)(B) and 10 CFR 50.82(8)(i)(C) require that a reserve be maintained in the DTF to accommodate a sudden unexpected delay in decommissioning activities.

All spent fuel is expected to have been transferred to the completed ISFSI and the existing Spent Fuel Building (SFB) is expected to be decommissioned by the end of 2004. Fuel management costs will consist of operational costs until 2023 (assuming DOE has completed all HLW removal from the site by that date) and decommissioning costs associated with the ISFSI.

All D&D activities are scheduled for completion by year end 2004. Assuming a SAFSTOR condition, DTF expenditures would be minimized to D&D activities only. Maine Yankee forecasts sufficient DTF balances should a SAFSTOR condition occur during the period between 2001 and 2004.

After 2004, the majority of expenditures from the DTF are related to the ISFSI. As shown in Table 7-2 column 5, sufficient funding will exist, based on Maine Yankee's assumption that the DOE will meet its responsibility for Spent Fuel disposal by 2023.

As demonstrated in Table 7-2, Maine Yankee will maintain adequate DTF balances for **ALL** $635M of decommissioning expenditures. To further strengthen this position, Maine Yankee has identified additional funding options, if necessary, as listed below:

- Maine Yankee is projecting decommissioning trust fund balances above projected minimum requirements and therefore has the capacity to incur costs greater than assumed in the current financial planning cost estimate dated January 2002.
- Any proceeds resulting from a favorable outcome in litigation against Stone and Webster and its parent corporations.
- Lower expenditures than estimated due to successfully managing the use of unallocated contingency.
- Maine Yankee has projected corporate cash which could be used to fund decommissioning.
- Agreements with FERC to file no later than January 1, 2004 for further rate adjustments.
- Maine Yankee's ability to defer decommissioning activities in order to reduce DTF expenditures.

Pursuant to 10 CFR 50.75 and 10 CFR 50.82 regulations, we believe we have demonstrated a financial plan, which includes adequate reserves for the entire decommissioning and ISFSI-related costs and therefore, meets the requirements for costs associated with decommissioning and dismantlement as defined by these regulations and, in fact, have demonstrated capability beyond the required NRC definition of decommissioning.

Table 7-3
Maine Yankee Comparison Summary of Decommissioning Costs

Plant Radiological Decontamination	TLG Site Specific Cost Estimate [1][2]	Maine Yankee Financial Planning Estimate [1] Expended [3][4]	Projected	Total
Staffing	$105,789	$72,433	$59,162	$131,595 [7]
LLW Burial	$ 75,244	$52,463	$29,136	$ 81,599
Equipment Removal	$ 51,439	$34,427	$ 4,354	$ 38,781
LLW Packaging and Shipping	$ 19,344	$20,959	$ 5,934	$ 26,893
Decontamination Activities	$ 7,402	$21,328	$10,606	$ 31,934
Contingency	$ 69,961	$ 135	$19,118	$ 19,253
Other Costs[5]	$ 69,327	$41,318	$23,566	$ 64,884
Subtotal		**$243,063**	**$151,876**	
Total[6]	**$398,505**			**$394,939**

Notes:
(1) Reported in thousands (000's) of 2001 dollars. Allocation of costs based on Maine Yankee's estimate of grouping the cost categories per Reg. Guide 1.179.
(2) Estimate as shown on Table 7-1
(3) Expended through December 2001
(4) Until May 2000, Maine Yankee maintained a fixed price contract for Decommissioning Services. Amounts reported as expended include identifiable contract costs, and in some cases, a prorated distribution based on the best information available to Maine Yankee.
(5) Other Costs include insurance, property taxes, energy, NRC and State fees, etc.
(6) Amounts are based on activities related to the definition of "Decommissioning" in 10 CFR 50.2 and do not include the cost of removal and disposal of spent fuel or of non-radioactive structures and materials beyond that necessary to terminate the license.
(7) This value includes License Termination Survey costs of approximately $7M previously included under Site Restoration in the TLG Site Specific Cost Estimate (Table 7-1). Termination survey (FSS) costs are dominated by staffing expenses. Because of the decision to demolish above grade structures in the industrial area, the final status survey scope has been reduced, and the associated costs are now lower than that determined by TLG in 1997. (See Table 7-1.)

7.5 References

7.5.1 NRC Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors

7.5.2 Maine Yankee letter to NRC MN-98-65, November 3, 1998, Site Specific Decommissioning Cost Estimate and PSDAR Update

7.5.3 Decommissioning Cost Analysis for the Maine Yankee Atomic Power Station, October 1997 TLG Services Inc.

7.5.4 Federal Energy Regulatory Commission Order, June 1, 1999

7.5.5 Guidelines to Producing Decommissioning Cost Estimates, AIF/NESP-036