YANKEE ATOMIC ELECTRIC COMPANY

Telephone (413) 424-5261

49 Yankee Road, Rowe, Massachusetts 01367

September 2, 2004
BYR 2004-092

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

References: (a) YNPS Possession Only License No. DPR-3 (Docket No. 50-29)

Subject: Submittal of Draft Revision 1 to the Yankee Nuclear Power Station's License Termination Plan (LTP)

This letter submits draft Revision 1 to the Yankee Atomic Electric Company (YAEC) License Termination Plan (LTP) for the Yankee Nuclear Power Station (YNPS)[1]. Only the revised pages are included. This revision incorporates our responses to the Request for Additional Information (RAIs)[2] as well as the modifications to our materials management program as presented at the June 17, 2004 meeting[3]. Some editorial and minor clarifications have also been incorporated. All changes have been clearly marked and changes associated with a response to an RAI have been indicated with the corresponding RAI number.

We trust this information is satisfactory; however, should you have any questions or require additional information, please contact me at (860) 267-3938.

Sincerely,

YANKEE ATOMIC ELECTRIC COMPANY

G. van Noordennen

Gerry P. van Noordennen
Regulatory Affairs Manager

Attachment

---

[1] YAEC Letter to USNRC, "Submittal of YNPS License Termination Plan and Proposed Revision to Possession Only License," dated November 24, 2003, BYR 2003-080.

[2] Letter, YAEC to USNRC, Responses to NRC Requests for Additional Information, dated August 2, 2004, BYR 2004-073.

[3] NRC Memorandum, J. Hickman to D. Gillen, Summary of the June 17, 2004 Meeting, dated August 5, 2004.

cc: J. Hickman, NRC, Senior Project Manager, NMSS
J. Wray, Inspector, NRC Region I (w/o enclosure)
R. Gallaghar, MA DPH
D. Howland, MA DEP
M. Rosenstein, EPA, Region 1
W. Perlman, Executive Committee Chair, FRCOG
T. W. Hutcheson, Chair, Franklin Regional Planning Board
L. Dunlavy, Executive Director, FRCOG
P. Sloan, Director of Planning & Development, FRCOG
D. Katz, CAN

Matrix of Changes to the License Termination Plan
(Draft Revision 1)

Area of Change(s)	Reason for Change
Page 1-2, first paragraph	Clarification and consistency with Section 2
Page 1-5, paragraph above Section 1.4.4	Added discussion of use of concrete debris as backfill.
Page 1-6, Section 1.4.6	Add volumetric concrete to the media types treated with RESRAD and for which DCGLs are calculated.
Section 1.6 , change criteria	Response to RAI #3
Section 1.6, paragraph under bullets	Response to RAI #6
Page 2-1, first paragraph	Clarifies entire history, rather than just "operating history"
Page 2-1, second paragraph	Simplified sentence.
Page 2-2, third paragraph	Change tense to present, as final status survey has not yet been designed.
Page 2-4, Section 2.1.2	Reworded for clarification
Page 2-5	Deleted extra hard returns
Page 2-7, second paragraph	Reworded for clarification.
Page 2-9	Changed punctuation
Page 2-10	Added hard return
Page 2-12	Added information on incinerator releases identified since original issue of the LTP.
Page 2-13	Added clarification regarding characterization for subsurface soils.
Page 2-15, third paragraph	Clarified "Class 2" was MARSSIM Class 2
Page 2-15, last paragraph	Capitalized "figure"
Page 2-19, second paragraph	Clarification and typographical error
Page 2-19, third paragraph	Clarified characterization for subsurface soil.
Section 2.6	Response to RAI #37
Section 2.7	Updated to include information in submittals provided since issuance of LTP.
A summary of the "Hydrological Report of 2003 Supplemental Investigation" will be added to LTP Section 2.	Response to RAI #51-56
Section 2.9	Added references.
Table 2-1	Added footnote for structures to remain intact

Area of Change(s)	Reason for Change
Table 2-5	Updated table for statistics using the current DCGLs (associated with HSA revision)
Table 2-6	Added half lives of radionuclides of concern.
Table 2-7	Updates sample results with 4th quarter data.
Figure 2-7 thru 2-16	Revised figure and adds figures to reflect information from the 2003 Hydrogeologic report
Page 3-2, last two bullets	Revised to reflect use of debris as backfill (versus removal as waste)
Page 3-3	Spelled out VC for first use
Page 3-4, second paragraph from bottom	Revised to reflect use of debris as backfill
Page 3-11, sentence above Section 3.2.2.2.3	Reflects changes in decommissioning activities for Vapor Container.
Page 3-12, sentence above Section 3.2.2.2.4	Reflects potential use of RSS debris as backfill
Page 3-14	Reflects potential use of IX Pit debris as backfill
Page 3-15	Reflects change in decommissioning strategy for SFP and new fuel vault and potential use of debris as backfill
Page 3-16	Reflects change in decommissioning strategy for PAB, Waste Disposal Building, and Safe Shutdown Building and potential use of debris as backfill
Page 3-17	Reflects change in decommissioning strategy for Warehouses and Compactor Building and potential use of debris as backfill
Page 3-18	Reflects change in decommissioning strategy for the Service Building and potential use of debris as backfill
Page 3-20, Section 3.4.2	Reflects change in waste management approach
Page 4-1	Changed punctuation
Page 4-2, first paragraph under Section 4.2.2	Reflects change in decommissioning strategy to use debris as backfill
Page 4-2, Section 4.2.1	Response to RAI #42
Page 4-4, Section 4.2.2	Revised tense

Area of Change(s)	Reason for Change
Page 4-4, Section 4.2.2	Revised to clarify options that can be taken
Page 4-4, Section 4.3.2, last paragraph	Response to RAI #44
Table 4A-1	Response to RAI #45
Section 4A.2	Response to RAI #46
Section 5.1	Response to RAI #1
Section 5.1, last paragraph	Response to RAI #2, 19, 27, 28, 29
Sections 1.6 and 5.4.1 under "Specify Tolerable Limits on Decision Errors"	Response to RAI #3
Section 5.5.3.5, second paragraph	Response to RAI #4, 17
Section 5.4.2, first paragraph	Response to RAI #5
Section 5.4.2, last paragraph	Response to RAI #6, 24
Section 5.4.3	Response to RAI #7, 20
Section 5.4.4	Response to RAI #18
Section 5.4.5.2, first paragraph under bullets	Response to RAI #9
Section 5.4.5.2, last paragraph	Response to RAI #10
Section 5.4.6.1, last paragraph	Response to RAI #11
Section 5.4.6.2	Response to RAI #12, 13
Section 5.5	Response to RAI #14
Section 5.5	Response to RAI #15
Section 5.5	Response to RAI #16
Section 5.5.1.2, sentence after two bullets and 5.4.4	Response to RAI #18
Page 5-21, equation mid-page	Renumbered equation
Page 5-22, equations and sentence in first paragraph of 5.5.1.5	Renumbered equations
Page 5-23, equations	Renumbered equations
Page 5-23, paragraph after Equation 5-11b	Clarifies the application of scanning MDC soil containing ETD only and HTD radionuclides
Page 5-24, equations	Renumbered equations
Page 5-25, equations	Renumbered equations
Page 5-26, first paragraph	Response to RAI #19
Page 5-27, equations and first paragraph	Renumbered equations
Section 5.5.2	Response to RAI #21
Section 5.5.3.3	Response to RAI #22
Section 5.5.3.3	Response to RAI #23
Section 5.5.3.5, second paragraph	Response to RAI #17, 25
Section 5.5.3.5, fourth paragraph	Response to RAI #26
New Section 5.6.1.4	Added section to describe bulk spectroscopy monitor to be used for volumetrically contaminated debris
Section 5.6.1.6, first paragraph	Response to RAI #30

Area of Change(s)	Reason for Change
Section 5.6.2.3	Response to RAI #31
Page 5-38, equation	Renumbered equation
Page 5-39, equation and middle paragraph	Renumbered equations
Section 5.6.2.4, last sentence	Response to RAI #32
Section 5.6.2.4.4	Response to RAI #33
Section 5.6.2.4.4, second paragraph, last sentence	Response to RAI #34
Page 5-41, equation	Renumbered equation
Page 5-43 Table 5-4	Added bulk spectroscopy monitor to list of available instruments
New Section 5.6.3.1.4	Added new section to describe final status survey activities for concrete debris
Section 5.6.3.2.2	Response to RAI #36
Section 5.7.2	Response to RAI #39
Page 5-53 equations and second paragraph above Section 5.7.4	Renumbered equation
Section 5.7.5	Response to RAI #40
Section 5.7.5, last three sentences of section	Response to RAI #41
Page 6-5	Clarification
Page 6-6	Clarification
Page 6-7	Clarification
Page 6-8	Clarified that DCGL in appendix was for 25 mrem/yr
Section 6.4.1.4	Clarified that DCGL in appendix was for 25 mrem/yr
Section 6.4.3	Rewritten for new waste management approach
Section 6.5	Added new section for dose due to groundwater at the MCLs
Section 6.6	Added new section to discuss how dose contributions from different media will be combined.
Section 6.7	Added decay discussion
Table 6-1	Response to RAI #57
Page 6-16	Renumbered appendices
Page 6-17, top of page	Renumbered appendix
Section 6.9	Added references for concrete discussions
Appendices 6K-6S	Adds appendices to support concrete discussions, renumbers appendices for areas factors

Area of Change(s)	Reason for Change
Page 8-4, top of page	Added information on use of concrete debris as backfill
Section 8.1.3.3.10	Added that concrete debris that has passed a final status survey may also be used as backfill
Section 8.2.3, last paragraph on page 5-11	Added information about use of concrete debris as backfill

YAEC, or USGen New England, Inc. (referred hereafter as "USGen"), owns all of the land located within the licensed site property boundary (see Figure 1-1), and all of the property within the exclusion area is under the control of YAEC. The USGen property is generally located along the Deerfield River and Sherman Reservoir. Portions of the USGen are considered impacted by licensed activities and are generally located at the northeastern end of the YAEC industrial area, the southern reaches of Sherman Reservoir, and the property outside of the industrial area fence located between Yankee Road and the Deerfield River. These impacted areas are included in license termination activities. Notable plant structures located on USGen property are the circulating water discharge seal pit, the Screenwell Pump House, and the meteorological tower located on a peninsula at the northeast corner of the site. The current nearest resident is located approximately 0.8 miles from the plant site (Reference 1-5).

The significant features of the site are shown in Figure 1-2.

1.2.2 Surrounding Areas

The following paragraphs describe the features and uses of land within 5 miles of the plant. Included is a summary of the population centers within 10 miles of the YNPS site.

Major Bodies of Water: In addition to Sherman Reservoir and the Deerfield River (including tributaries and brooks feeding it), other major bodies of water are located within 5 miles of the YNPS site. These include: Sadawga Pond (184 acres), Shippee Pond (25 acres), North Pond (17 acres), and Clara Lake (12 acres) in Whittingham, Vermont; Howe Pond (42 acres) in Readsboro, Vermont; and Bear Swamp Upper Reservoir (128 acres) and Pelham Lake (89 acres). in Rowe, Massachusetts.

Industry: There are no exclusively commercial areas within 5 miles of the plant. The only industry within the area is the YNPS and the USGen hydroelectric stations. USGen has five powerhouses within 5 miles of YNPS. There are three stations as a part of the Deerfield River Project. They are the Harriman, Sherman, and No. 5 Stations. In addition the Bear Swamp and Fife Brook stations are a part of the Bear Swamp Pumped storage facility.

Public Lands and Conservation Areas: There are several public lands/conservation areas within 5 miles of the YNPS site. These areas offer a variety of recreational opportunities including fishing, hunting, boating, swimming, picnicking, and hiking.

Schools: There are two schools within 5 miles of the plant: Rowe Elementary located about 2.5 miles southeast of the site on Pond Road in Rowe, Massachusetts and Readsboro Central School, located off of Route 100 near the center of Readsboro, Vermont.

Farms: Information was collected by YAEC to document the current nearest garden and milk animal locations. These locations may include farms or simply private gardens or dairying locations. Table 1-1 identifies these locations by sector.

Water Supplies: Water supplies within the Deerfield River Drainage Basin, including the entire area within 5 miles of the plant, generally consist of private wells. The only communal source of

data collected during this process provides a basis for developing plans for remediation and Final Status Surveys.

Operational radiation surveys and additional measurements and samples obtained during decommissioning activities will be used to confirm the area classification and effectiveness of the cleanup activities before completing the Final Status Survey.

As a result of the HSA, and site classification, approximately 2170 acres of the 2200-acre plant site have been identified as "non impacted" as defined in MARSSIM. Tables 2-1 and 2-2 provide the area classifications for the various survey areas of the YNPS site.

1.4.3 Identification of Remaining Site Dismantlement Activities

In previous phases of decommissioning, major plant systems and components were removed from site buildings. These included the steam generators, reactor vessel, and reactor coolant piping, as well as the turbines, generator and other plant systems not serving spent fuel pit support functions. After component removal, some buildings and land areas were remediated in preparation for the Final Status Survey and some underground and embedded piping were removed. As previously discussed, LTP-related and Final Status Survey activities were halted in September of 1999, based upon the availability of new survey guidance in MARSSIM. The focus then shifted from decommissioning activities to spent fuel storage activities, and all fuel and greater-than-class-C (GTCC) waste was removed from the spent fuel pit and placed in storage casks on the pad at the onsite independent spent fuel storage installation (ISFSI). Removal of spent fuel and GTCC waste from the pool and placement on the ISFSI pad was completed in June of 2003.

In the current phase of decommissioning, YAEC, with the assistance of a demolition contractor, is demolishing most site structures to grade. Structural demolition debris may be surveyed using site procedures that invoke the "no detectable radioactivity" criterion (consistent with the guidance in NRC Circular IEC 81-07, "Control of Radioactively Contaminated Material") or may be subjected to a final status survey using the DCGLs, discussed in Section 6 of this LTP. Materials meeting this criterion may remain onsite and may be used as backfill, subject to regulations on the use of such materials by the Commonwealth of Massachusetts, or removed offsite for disposal. The Vapor Container is being dismantled, decontaminated, and removed from the plant site. The Reactor Support Structure will be subjected to a survey and the associated debris may be used as backfill.

1.4.4 Site Remediation Plans

Section 4 of the LTP describes various methods that can be used during YNPS decommissioning to reduce the levels of radioactivity to those which meet the NRC radiological release criteria, that is, do not exceed 25 mrem/yr total effective dose equivalent (TEDE) and are as low as reasonably achievable (ALARA). This section describes the methodology that will be used to demonstrate that the residual radioactivity has been reduced to a level that is in compliance with the NRC requirements.

1.4.5 Final Status Survey Plan

The primary objectives of the Final Status Survey are to:

- verify proper survey unit classification (or reclassify survey unit),
- demonstrate that the level of residual radioactivity for each survey unit is below the release criterion, and
- demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit.

The purpose of the Final Status Survey Plan is to describe the methods to be used in planning, designing, conducting, and evaluating Final Status Surveys at the YNPS site to demonstrate that the site meets the NRC's radiological criteria for unrestricted use. Section 5 of the LTP describes the Final Status Survey Plan, which is consistent with the guidelines of MARSSIM. The plan also describes methods and techniques used to implement isolation controls to prevent re-contaminating previously remediated areas.

1.4.6 Compliance with the Radiological Criteria for License Termination

Section 6 together with Section 5, Final Status Survey Plan, describes the process to demonstrate compliance with the radiological criteria of 10CFR20.1402 for unrestricted use for the YNPS site. YAEC has selected the RESRAD computer code (Version 6.21) to model the dose from soils and volumetric concrete and its counterpart, RESRAD-BUILD (Version 3.21), to model the dose from structural surfaces.

Two scenarios have been selected for use with the RESRAD family of codes for calculating the radionuclide-specific derived concentration guideline levels (DCGLs). These scenarios are the resident farmer scenario for site soils and volumetric concrete. The building occupancy scenario is being used for surficial contamination in structures. DCGLs are the concentration and surface radioactivity limits that will be the basis for performing the Final Status Survey.

1.4.7 Update of the Site-Specific Decommissioning Costs

In accordance with 10CFR50.82 (a)(9)(ii)(F), Section 7 provides an updated, site-specific estimate of the remaining decommissioning costs. It also includes a comparison of these estimated costs with the present funds set aside for decommissioning and a description of the means to ensure that there will be sufficient funds for completing decommissioning.

1.4.8 Supplement to the Environmental Report

In accordance with 10CFR50.82 (a)(9)(ii)(G), Section 8 demonstrates that decommissioning activities will be accomplished with no significant adverse environmental impacts. Supplement 1 to NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (FGEIS)" (Reference 1-11) provides an assessment of the aspects of decommissioning with the potential to impact the environment. This assessment includes an evaluation of the significance of the impact of the activity (SMALL, MODERATE, or LARGE), as well as its applicability (generic to all or to a group of plants or site-specific).

3. A letter of intent to remove a portion of the property from the Part 50 license will be sent to the NRC, no later than sixty (60) days before the anticipated date for release of the subject survey area(s). This letter will contain a summary of the assessments performed, as described above, and, for areas designated as "impacted" will include the FSS report for the subject survey units(s) or area(s).

4. Once the land area(s), and any associated building(s), have been verified ready for release, no additional surveys or decontamination of the subject building or area will be required (beyond those outlined in Section 5.4.5 intended for isolation and controls) unless administrative controls to prevent recontamination are known or suspected to have been compromised. Following completion of the Final Status Survey and submittal of the associated report, the NRC will review the report and conduct, as appropriate, the applicable NRC confirmatory inspections.

5. Upon completion of the YNPS Decommissioning Project, a final report will be prepared, summarizing the release of areas of the YNPS site from the 10CFR50 license.

1.6 Change Criteria for the License Termination Plan

YAEC is submitting this License Termination Plan as a supplement to the FSAR. Accordingly, the License Termination Plan will be updated in accordance with 10CFR50.71(e). Once the LTP has been approved, the following change criteria will be used, in addition to those criteria specified in 10CFR50.59 and 10CFR50.82(a)(6). A change to the LTP requires NRC approval prior to being implemented, if the change:

(a) Increases the probability of making a Type I decision error above the level stated in the LTP;

(b) Increases the radionuclide-specific derived concentration guideline levels (DCGLs) and related minimum detectable concentrations;

(c) Increases the radioactivity level, relative to the applicable DCGL, at which investigation occurs;

(d) Changes the statistical test applied to one other than the Sign Test or Wilcoxon Rank Sum Test.

(e) Results in use of a null hypothesis other than that stated in Section 5.4.1; that is, "The survey unit exceeds the release criteria." RAI#3

Re-classification of survey areas from a less to a more restrictive classification (e.g., from a Class 3 to a Class 2 area) may be assigned without prior NRC notification; however, re-classification to a less restrictive classification (e.g., Class 1 to a Class 2 area) and/or subdivision of a survey area will require NRC notification at least 14 days prior to implementation. RAI #6

2 SITE CLASSIFICATION

2.1 Historical Site Assessment and Survey Area Delineation

2.1.1 Approach and Rationale

The Historical Site Assessment (HSA) (Reference 2-1) for the Yankee Nuclear Power Station (YNPS) documents those events and circumstances occurring during the history of the facility that contributed to the contamination of the site environs above background levels. Information relevant to changes in the radiological status of the site following publication of the HSA will be considered a part of the continuing characterization evaluations (see Section 2.6). The continuing evaluations include ongoing decommissioning activities, the expansion of the site groundwater investigation and evaluations of subsurface contamination. The results of the ongoing investigations into the extent of subsurface contamination will drive continuing remediation and/or mitigation efforts as appropriate.

The HSA approach collected, organized and evaluated information that described the YNPS site in terms of physical configuration and the extent to which the site was radioactively contaminated as a result of plant operations and decommissioning activities. The HSA information was used to bound and classify survey areas. The boundaries of the identified survey areas as depicted in Figures 2-1a, 2-1b and 2-2 were selected based on operational history including recorded significant events, common radiological profiles and where appropriate, parcel ownership boundaries. The preliminary survey area classifications and sizes are shown in Table 2-1 for structures and Table 2-2 for open land areas. Survey areas for structures will be broken into multiple survey units where appropriate in order to meet the survey unit size limitations recommended by NUREG-1575 (Reference 2-2). All open land survey area boundaries have been sized to meet the NUREG-1575 size limitation constraints.

The general criteria used to classify the identified survey areas was drawn from the regulatory guidance of NUREG-1575 (MARSSIM) as follows:

Non-impacted Area: Areas where there is no reasonable possibility (extremely low probability) of residual contamination. Non-impacted areas are typically off-site and may be used as background reference areas.

Impacted Area: Any area that is not classified as non-impacted. Areas with a possibility of containing residual radioactivity in excess of natural background or fallout levels. All impacted areas must be classified as Class 1, 2 or 3 as described in NUREG-1575.

Class 1 Area: An area that is projected to require a Class 1 final status survey. Impacted areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) above the DCGL. Size limitations are ≤100 sq. m. for structures and ≤2000 sq. m. open land areas.

Class 2 Area: Impacted areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGL. Size limitations are >100 sq. m. and ≤1000 sq. m. for structures and > 2000 sq. m. and ≤ 10,000 sq. m. for open land areas.

Class 3 Area: Impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGL, based on site operating history and previous radiological surveys. There are no size limitations for Class 3 areas.

The collection and evaluation of site radiological information is conducted under approved site procedures. The output of this process is in the form of information generated for each survey area that will be used in the preparation of survey plans. Information generated for each survey area contains a detailed operational history, the current radiological status, an evaluation of radionuclide past and current translocation pathways that have been or continue to be operable and a description and status of decommissioning work performed. The decommissioning work description includes the results or status of any subsurface characterization or remediation efforts.

The general process for integration of the HSA with continuing characterization and Final Status Survey is shown in the following flowchart.

Over the operational history of the YNPS site, the term "remediation" was often used to refer to any process involving the removal of radioactive media. For the purpose of license termination activities, "remediation" is narrowly defined as efforts specifically conducted to reduce the quantity or concentration of radioactivity to a level below the appropriate Derived Concentration Guideline Level (DCGL). Other processes may be referred to as "mitigation" or routine decommissioning activities.

2.1.2 Boundaries of the Site

The YNPS site consists of about 2,200 acres on both sides of the Deerfield River in the towns of Rowe and Monroe, in Franklin County, Commonwealth of Massachusetts. Figure 1-1 shows the boundary of the site and plant exclusion area.

The "YAEC Deed Study Project Rowe and Monroe, Massachusetts," dated December 18,1998, (Reference 2-3) provides information concerning properties that make up the YAEC site and current abutments.

YAEC or USGen New England, Inc. (USGen) own all of the land located within the licensed site property boundary. All of the property within the exclusion boundary is under the control of YAEC. The USGen property is generally located along the Deerfield River and Sherman Reservoir. Portions of the USGen property are considered impacted by licensed activities and are generally located at the northeastern end of the YAEC industrial area, the southern reaches of Sherman Reservoir and the property outside of the industrial area fence located between Yankee Road and the Deerfield River. These impacted areas are included in license termination activities. Notable impacted plant structures on the USGen property within the site industrial area include the circulating water discharge seal pit, the Screenwell Pump House, and the meteorological tower located on peninsula at the northeast corner of the site.

Two public secondary roads traverse the exclusion area. The first, Tunnel Road, is across the river from the plant, approximately 1,500 feet away at its closest point, and runs north and south along the river connecting the towns of Monroe, Massachusetts and Readsboro, Vermont. The second, Monroe Hill Road, is approximately 2500 feet away from the plant at its nearest point and is located southwest of the plant and runs between the towns of Rowe and Monroe, Massachusetts. During the early site history, a public rail line ran through the industrial area. This rail line and the associated spur facilitated early construction and spent fuel shipments. Currently, there are no rail lines that traverse or are adjacent to the YNPS site.

Most of the site area is wooded with very steep grades on both sides of the Deerfield River. Features of the site include the Yankee Nuclear Power Station, the YNPS Independent Spent Fuel Storage Installation (ISFSI), the USGen Sherman Station hydroelectric plant, Sherman Reservoir and Dam, the transmission lines running through the site, the Yankee Administration Building and the Yankee Visitor Center (Furlon House).

2.1.3 Documents Reviewed

In performing the YNPS Historical Site Assessment (HSA) the following documents were reviewed:

- License and Technical Specifications
  - Technical Specification Changes
  - License amendments
- Original Plant Design
  - Function and purpose of systems and structures
  - Plant operating parameters
  - Plant operating procedures
- Original Plant Construction Drawings and Photographs
  - Specifications for systems and structures
  - Field Changes/as built drawings
  - Site Conditions
- Plant Operating History
  - Abnormal Operating Reports (AOR)
  - Licensee Event Reports (LER)
  - Plant Information Reports (PIR)
  - Radiological Occurrence Reports (ROR)
  - Radiological Incident Reports (RIR)
  - Condition Reports (CR)
  - Plant Operating Procedures Regarding Spills and Unplanned Releases
  - Plant Operations Logbooks
  - Radiological Environmental Monitoring Program and Radiological Environmental Technical Specification Reports (REMP & RETS)
  - Monthly Plant Operations Reports
  - Semi-Annual Plant Operations Reports
- Work Control Documents and Site Modifications
  - Job Orders
  - Plant Alterations
  - Engineering Design Change Requests (EDCR)
  - Plant Modifications
  - Maintenance Requests
- Radiological Surveys and Assessments
  - Radiological surveys performed in support of normal plant operations and maintenance
  - Radiological surveys performed in support of special plant operations and maintenance
  - Radiological assessments performed in response to radioactive spills or events

Decommissioning activities have resulted in the disturbance and/or excavation of soils in certain survey areas. Extensive soil evaluations have been performed in support of soil excavation. The soil excavations were associated with removal of sub-grade components/systems and site modifications necessary for the construction of the ISFSI and the upgrade of security measures around the spent fuel pool. Piles of excavated soil are located in several areas of the site.

Controls were in place to track the location of these soils from the point of origin (excavation) through temporary onsite storage to final disposition. Disturbed/excavated soils, evaluated and verified by sampling and analysis protocols to be non-detectable for radiological constituents (below environmental Lower Limit of Detection [LLD] level for soils) were used as backfill in some excavated areas. Excavated soils contaminated above a Guide Line Value (GLV) protocol were packaged and disposed of as radioactive waste. This protocol allowed some soils contaminated above background to be used as backfill in some locations. Retrospectively, the criteria is lower than the proposed DCGL. As these areas are evaluated for survey planning, the backfilled soil results will be evaluated against the soil DCGL for mitigation action.

During the evaluation of survey areas, walk-downs of each area were performed to document the types of survey media remaining or expected to remain at end-state. The walk-downs also documented the current decommissioning status of the area and identify any potential radionuclide translocation pathways that impacted the area or any contiguous survey areas. Such pathways include ongoing decommissioning activities or environmental transport pathways, such as sub-surface migration of radioactivity by surface water infiltration, wind, surface water run-off or wildlife.

2.1.5 Personnel Interviews

At the time of plant shutdown in 1992, personnel interviews were conducted as a part of an exit interview process. Since that time personnel have provided additional information on plant operations and practices when additional data was needed or desired relative to condition of the plant or activities performed.

2.2 History and Current Status

2.2.1 Licensing History

Yankee Atomic Electric Company is the holder of Yankee Nuclear Power Station Facility Operating License DPR-3 issued under the authority of the Atomic Energy Commission (AEC). Yankee Nuclear Power Station achieved initial criticality in 1960 and began commercial operations in 1961. The original thermal power design limit of 485Mwt was upgraded to 600Mwt in 1963.

On February 26, 1992, the YAEC Board of Directors decided to cease power operations permanently at YNPS. On August 5, 1992 the NRC amended the YNPS Facility Operating License to a possession only status.

The YNPS Decommissioning Plan (Reference 2-6) was submitted March 29, 1994 and received final approval in October 28, 1996. In May 1997, Yankee submitted to the NRC for approval a

specific decommissioning activities, and others are for existing YNPS site facilities and ongoing activities that are necessary to support decommissioning. The following is a partial listing of permits and approvals for decommissioning activities.

- Air emissions from the burning of diesel fuel are regulated by the Commonwealth of Massachusetts Department of Environmental Protection, Air Quality Control Division.
- Non-radioactive liquid effluents are administered by the Commonwealth of Massachusetts Department of Environmental Protection, Division of Water Pollution Control.
- Liquid effluents are controlled under the National Pollutant Discharge Elimination System (NPDES permit) under the EPA and State (Commonwealth) approvals.
- Building permits may be required by the Town of Rowe, Massachusetts, for temporary field office facilities constructed on the plant site to support decommissioning activities. The Town of Rowe uses the Uniform Building Code for evaluating building permit applications.
- The site make-up water wells are operated under permits from the Commonwealth of Massachusetts Department of Environmental Protection, Division of Water Supply.
- Hazardous waste generation is regulated by the Commonwealth of Massachusetts Department of Environmental Protection, Division of Hazardous Waste. Notification of the generator status and annual reporting are conducted in accordance with Massachusetts regulations.
- The Commonwealth of Massachusetts, Department of Labor and Industries, Division of Industrial Safety, regulates the installation, removal and encapsulation of friable asbestos-containing materials and lead-based paint. All non-radiological solid waste will be handled and disposed of in accordance with State and local rules and regulations.
- The Commonwealth of Massachusetts, Department of Public Health, Radiological Control Program, and the Vermont State Health Department, Division of Occupational and Radiological Health, are notified in advance of all placarded shipments of radioactive waste. In addition, the Governors of all affected States receive advance notifications in accordance with 10 CFR 71.97, "Advance notification of shipment of nuclear waste."
- Licenses are required for radio communications by the Federal Communications Commission.
- PCB paints will be removed from all exposed concrete surfaces as required by the Alternate Method of Disposal Authorization (AMDA) requirements prior to demolition of the structures as authorized by the EPA on October 8, 2002 and subsequent changes thereto.

2.2.3 Description of Operations Impacting Site Radiological Status

Normal plant operations were expected to result in contamination of certain areas of the site and these areas were designed to contain such material; however, early in the plant life, certain events and conditions resulted in radioactive material being deposited in other locations. As a result, the plant design and operational procedures evolved to accommodate or eliminate these

circumstances. Review of the early operational history of the site drew heavily on the Plant Superintendent's "Monthly Operating Reports".

The following principal events and circumstances listed in chronological order generally contributed to the various aspects of residual contamination found on the site to be dispositioned at decommissioning.

- Release of elemental silver and nickel into the reactor coolant due to mechanical wear and corrosion from the initial set of control rods resulted in distribution of radioactive silver in plant systems and on equipment used during the first refueling. [circa 1960's]

- Storage of the refueling equipment and prepared radioactive waste outdoors resulted in distribution of contamination, including radioactive silver, within the RCA yard area.

- Snow removal activities performed in the RCA caused a redistribution of accumulated surface contamination to the areas outside the RCA where snow was relocated.

- Rain falling on the surface of yard areas in the RCA caused redistribution of the contamination into low areas in the RCA and into the storm drain system.

- Leaks in the radioactive systems in the Ion Exchange (IX) Pit resulted in contamination of the water in the IX Pit. A defect in the construction of the IX Pit concrete allowed the contaminated water to leak, resulting in contamination of the subsurface soils, asphalt and concrete around the IX Pit and adjoining structures.

- Wear on internal valve components made of stellite resulted in the introduction of wear particles into the reactor primary system. These particles were activated to gamma emitting Co-60 during plant power operations. Some particles associated with fuel fragments were also generated during plant operations. Maintenance on primary system components resulted in the distribution of these activated particles onto tools and equipment. Although not a frequent occurrence, Co-60 particles have been identified and removed during surveys of the yard area. The particles associated with fuel fragments have not been identified in open yard areas but were mostly confined to controlled contamination areas.

- A failure of a check valve allowed a backflow of shutdown cooling water to enter the seal water system resulting in contamination of the normally clean seal water system up to and including the vent port on the PAB roof.

- Out of doors decontamination facilities (North and South decontamination pads) resulted in contamination of the soils around the pads.

- The repair of a damaged reactor cooling pump motor on the normally clean turbine deck resulted in contamination of the turbine building generally and on the turbine deck and control room specifically.

2.2.4.1 Unplanned Gaseous Releases

Over the lifetime of the plant, a number of unplanned gaseous release events occurred. Short descriptions of these gaseous events as described in AOR/PIR/LER's are documented in the HSA. A careful review of these unplanned discharges did not reveal any unmonitored particulate component that could have significantly contributed to the long-term contamination of the site or its environs.

A detailed study of planned particulate releases during the operating history of YNPS is presented in Section 2.5 as partial justification for the non-impacted status of a majority of the YAEC owned property. This study considered the impact of the particulate emissions from the primary vent stack. In this study (Ref. 2-13) it was presumed that the radioactive waste incinerator operated until 1964. The four years of batch incinerator emissions were considered to be of negligible impact when compared to the particulate releases from the primary vent stack over the life of the plant. Follow-up investigation of the history of the radioactive waste incinerator revealed that the incinerator actually operated until 1975. The particulate emissions from the radioactive waste incinerator were re-evaluated, and this re-evaluation also concluded that operation of the incinerator has had an insignificant impact on site environs (Ref. 2-18).

2.2.4.2 Unplanned Liquid Releases

Several AOR's and PIR's reviewed documented unplanned liquid releases that resulted in contamination of the site grounds, buildings and subsurface locations. When subsurface contamination investigations were not performed due to inaccessibility or were not completed to the level suitable for license termination, these locations are targeted for continuing characterization investigation. Table 2-3 provides a listing of the events identified by the HSA that have resulted in contamination of the site. Appendix 2A provides a brief summary of each event based on documentation prepared at the time of the incidents and an assessment of which survey areas were impacted by the events.

2.3 Findings

2.3.1 Overview

As described in Section 2.1.1 above, the preliminary boundaries of the survey areas as depicted on Figures 2-1a, 2-1b and 2-2 were selected based upon operational radiological history. An in-depth assessment of the operational history performed during compilation of the HSA was used to bound and classify the survey areas in accordance with the guidance of NUREG-1575. Survey area classifications are shown in Figures 2-3 and 2-4 in a color-coded site map format. Table 2-1 and Table 2-2 list the survey area dimensions and their classifications in a tabular format.

Generally, of the approximately 2200 acres of land that comprise the YNPS site, less than 30 acres was impacted by plant operations. The majority of these 30 acres is minimally impacted and, as such, is classified as a group of Class 3 open land survey areas. The Class 3 open land survey areas identified at a distance from the site industrial area are areas that received material, primarily soil, from locations within the plant that are impacted areas. The survey areas that form

the perimeter of the impacted areas of the site proper were classified as Class 3 open land survey areas and account for the potential translocation pathways of site-related radioactivity into the surrounding environment by winds, surface water, groundwater, and wildlife intrusion.

The Class 2 open land survey areas that abut the Class 1 open land survey areas are potentially contaminated or known to be contaminated, but are not expected to exceed the DCGL. This creates a buffer zone that will receive a higher level of assessment based upon its likelihood to contain radioactivity at some fraction of DCGL.

Class 1 open land survey areas are identified based upon historical information indicating the potential presence of radioactivity at levels greater than DCGL. Table 2-5 summarizes the radiological conditions of open land areas and the associated MARSSIM classifications as well as the total land area by survey area. The radiological condition of each area is expressed as the minimum, maximum and mean of the sum of fractions of a DCGL for soils.

Subsurface soils and subsurface structures/systems located within or that traverse an open land survey area will be evaluated separately as part of the continuing characterization process described in Section 2.6 of this document.

All YNPS structures associated with the site are considered impacted to some extent by plant operations and are located within an impacted land survey area. Few of the structures on site will remain in use after the current phase of decommissioning is complete. The majority of the structures will be demolished to grade with the debris being used as back fill. The remaining portions of the structures will consist of reinforced concrete floor slabs, foundations and sub-grade structures. The floor slabs, adjoining interior walls and above grade exterior walls may all be included within a given survey unit dependent on surface area size limitations. The sub-grade reinforced concrete walls and undersides of floor slabs will be investigated separately. Table 2-1 summarizes the structure survey area classifications and the total interior area to be surveyed. A summary of the current radiological conditions of structures and buildings tabulated by survey area is presented in Table 2-4. This information was further evaluated in consideration of the decommissioning activities previously performed, the potential impact of future decommissioning activities, and the projected end-state of the site at conclusion of all decommissioning activities in order to select the preliminary classification status.

2.3.2 Radionuclides of Concern at YNPS

An analysis has been preformed to determine the radionuclides that have potential dose significance at License Termination (Reference 2-9). This analysis has used three sources of radionuclide data to assure that all significant nuclides associated with plant operations are identified. The sources are selected Part 61 analyses representing several media types spanning a time period from pre-shutdown to the present, radionuclide distributions identified in the YNPS Decommissioning Plan (Reference 2-6) and source term information from NRC published reports. The significant radionuclides identified from the Part 61 analyses encompassed those identified from the latter two sources. The final listing of potentially significant radionuclides is shown on Table 2-6.

2.4.2 Buildings, Structures and Open Land Areas Outside of the RCA

The following designations are used in identifying survey areas outside of the RCA (Figures 2-3 and 2-4):

OMB	Support Buildings Outside the RCA
OOL	Open Land Areas Outside the RCA
SVC	Service Building
TBN	Turbine Building

Summary individual Survey Area assessments are described in Appendix 2C. In general, the impacted areas immediately outside the confines of the historical RCA have been assigned a NUREG-1575 Class 2 status. These buffer zones are areas where radionuclides may have migrated beyond the RCA boundary due to environmental or other translocation vectors.

The exceptions are Survey Areas OOL-12 and OOL-13 where radionuclides are known to have migrated beyond the RCA boundary due to the combination of a recorded contaminating event (PIR 81-09) and a significant rain event. Surface run-off from the RCA yard not channeled into the storm drain system migrated down grade along the rail spur in these areas toward Sherman Reservoir. Although the surfaces of these areas were quickly decontaminated and cleared for general access, some of the contamination carried by the run-off filtered into the crevices of the rails and rail bed remain embedded. These areas have been assigned a Class 1 status.

Survey Area OOL-07 has been assigned a Class 2 status as it contains soils removed from other class 2 areas and soils that have only been evaluated by composite sampling techniques.

The remaining impacted areas are assigned a Class 3 status. These areas were designated as impacted areas for a wide variety of reasons. None of these areas are expected to contain radioactivity in excess of a small fraction of the appropriate DCGL.

2.5 Non-Impacted Area Justification

2.5.1 Non-Impacted Area Description

The majority of the land surrounding the industrial area of the site is classified as non-impacted according to MARSSIM criteria. This portion of the site is open land consisting of approximately 2170 acres. The non-impacted land surrounds the industrial area and all other routinely utilized areas. The non-impacted area is bounded on the east and south by Monroe State Forest, on the southeast by USGen property, on the west by Readsboro Road (with the exception of an 89 acre plot on Kingsley Hill Road), and on the north by the Massachusetts/Vermont state line. The non-impacted area was not involved in plant operations and consists mostly of rugged terrain which is forested and undisturbed. Power lines traverse the area in a northeast by east direction (see Figure 2-5). The general site is shown on USGS map Rowe, Massachusetts-Vermont (Reference 2-10).

Appropriate samples will be obtained to identify the depth at which contamination, if any, above DCGL limits occurs. The evaluation of soil under concrete and asphalt will also be addressed. Survey plans will be developed for sampling of soil under contaminated slabs, especially at the location of expansion joints, cracks, and other potential contamination pathways from the concrete surface to the sub-slab soil.

Subsurface investigations will include collection of soil cores. Evaluation of these cores may include segregating them into smaller increments, based upon measurements from field screening techniques. Figure 2-6 illustrates the locations where targeted subsurface investigations will be performed. A finding of subsurface soil above the DCGL will prompt further investigation in order to determine the horizontal and vertical extent of the contamination. The investigation will continue until the area of contamination is well defined. This is generally accomplished when soil from peripheral cores are less than the DCGL. The conclusion in that case is that the investigation has bounded the extent of contamination. All subsurface areas known to be impacted will be investigated and soil radioactivity levels will be reduced to less than the soil DCGL.

Following the remediation/mitigation of all targeted subsurface locations and as part of the final status survey program, a series of systematic subsurface borings will be conducted in the area delineated in Figure 2-6. Radiological evaluations of volumetric material in the vertical column at each subsurface survey location will be performed to substantiate the evaluation that all subsurface locations have been identified and are below the clean-up criteria.

2.7 Continuing Investigation of Groundwater Contamination

2.7.1 History

The basic site geology has been well documented in licensing studies and documents. Figure 2-7 illustrates the locations of existing and proposed groundwater monitoring wells. The first site monitoring wells, B-1 and B-3, were installed within the Radiologically Controlled Area (RCA) in December 1977 and October 1979, respectively. Well B-3 was used to monitor groundwater level only; and no samples were analyzed for radionuclides. Well B-3 was closed in January 1997.

Following the decision to terminate plant operation, monitoring wells CB-1, -2, -3, and -4, and CW-1, -2, -3, -4, -5, and -6 were installed just down gradient of locations where spills or leaks are known to have occurred. The location, extent and impact of leaks resulting in the contamination of the site are discussed in the Historical Site Assessment and have been summarized in previous subsections of this LTP.

The YNPS Radiological Environmental Monitoring Program (REMP) has identified tritium in Sherman Spring. Tritium was also identified in samples routinely drawn for REMP from monitoring well B-1. The identification of H-3 in the groundwater as a substance of concern was documented in the YNPS Decommissioning Plan; however, recent samples have not detected tritium in Sherman Spring.

The additional wells installed after 1993 further defined the extent of H-3 migration beneath the plant industrial area and toward the Deerfield River and Sherman Dam. Analyses for H-3 from wells, along with REMP results for Sherman Spring, provided a working model for groundwater flow in the shallow outwash aquifer beneath the site. They also served as a basis to help locate additional monitoring wells (CB-6, -8, -9, CW-7, and -8) installed in 1994 to further define general groundwater flow and the H-3 plume at the site. The shape of the H-3 plume, based on analyses from the above wells, can be seen in Figure 2-8.

Additional core borings that serve as draw points for groundwater samples (CB-5, -7, -8, -10, and -12, and CW-10) were installed up gradient or cross-gradient of the PAB/SFP/IX Pit complex, in impacted locations beneath building slabs. While these are not actual monitoring wells with installed screens, they do provide scoping type groundwater data when water is present within the bore holes.

A series of deep-bedrock wells were installed during the summer of 2003 in order to investigate the possible existence of a deep plume of contamination. The wells currently in existence, that were installed prior to 2003, are at the level of the glacial outwash or in unfractured till. These wells monitor the concentration of the radionuclides in the groundwater to depths of about 30-70 feet. The new wells investigated depths to bedrock which ranged from 43 to 280 feet.

Figure 2-7 shows the location of these new bedrock monitoring wells (MW100-107). The designation 'A', 'B', or 'C' for these wells signifies outwash, bedrock, or intermediate depth wells, respectively. Intermediate wells were installed at depths where aquifers were encountered that yielded positive tritium results.

2.7.2 Evaluation of Historical Data

Figure 2-8 shows the current data for H-3 in samples taken from wells near the plant structures.

CB-11A was installed in the PAB following detection of H-3 in samples from standing water exposed during concrete floor removal in that building in 1997. Subsequent samples from that well revealed elevated H-3 concentrations in a highly localized zone. Several new monitoring wells were placed in the vicinity of that well to assure that any significant related information was investigated.

A document had been prepared to address the set of groundwater data existing as of 2001 (Reference 2-19). This document was reviewed, and the review, and resulting recommendations, were documented in Reference 2-20. These recommendations led to revisions to the current groundwater monitoring program.

2.7.3 Groundwater Monitoring Program

During the second quarter of 2003, the recommendations provided in Reference 2-20 were used to update YNPS procedures in order to continue and expand the groundwater investigation effort. These updated procedures address:

- Ground and Well Water Monitoring
- Radiochemical Data Quality Assessment
- Site Characterization and Site Release Quality Assurance Program Plan for Sample Data Quality and
- Groundwater Level Measurements and Sample Collection in Observation Wells.

The revised program includes analyses of a standard suite of radionuclides based upon known contaminants from plant spills and leaks, and historical evidence from other facilities undergoing decommissioning (see Section 2.3.2). This program also implements a standard "low-flow" method for sample collection. Preconditions for well purging and limits on sample turbidity and changes in pH prior to sampling were implemented for the round of sampling performed during the summer of 2003. These controls minimize the entrainment of particulate matter in the well water samples and avoid bias due to inclusion of particulate matter.

The groundwater monitoring program is an iterative process. Accordingly, data obtained from the groundwater monitoring program are reviewed and analyzed and results are discussed internally and with various regulators and stakeholders. These discussions may result in planning of additional investigative activities (e.g., to include or remove radionuclides for which sampling is performed or addition of monitoring wells). Any program changes are formally approved and documented.

Reports were developed to discuss the findings from the third and fourth quarter 2003 well drilling and sampling campaigns (References 2-21 and 2-22). As documented in this report, tritium is the only plant-related radionuclide positively detected in groundwater at the Yankee Rowe site. The data indicate that tritium levels have declined substantially in the shallow aquifer over the period of record. Tritium concentrations exceed the MCL in a relatively small area in the glaciolacustrine sediments that lie beneath the shallow aquifer. The data indicate that this area is localized and within about 100 feet (laterally) of the SFP/IX Pit complex. Figures 2-9(a) and 2-9(b) map the tritium plume for the shallow aquifer. The dose associated with the tritium in the groundwater is low. On this basis, the corresponding risk to human health and the environment also appears to be low. RAIs #51 to 56

It appears likely that leaks from the SFP/IX Pit complex were a source of tritium in the groundwater at Rowe. The Primary Auxiliary Building was another potential source of tritium contamination. The Spent Fuel Pit and IX Pit are adjacent and share a common wall.

Historical monitoring data for Sherman Spring suggest that groundwater in the shallow stratified drift aquifer was impacted in the early 1960s, before the leak in the IX Pit was repaired. Water quality in the shallow aquifer has improved dramatically since the repair. The relatively large

hydraulic conductivity of the stratified drift allows groundwater to flow through the shallow aquifer at the comparatively fast rate of about one foot per day. That flow has allowed natural attenuation of the tritium in the shallow system to proceed relatively quickly.

The underlying glaciolacustrine sediments also have been impacted by tritium. The aquitard separating the stratified drift aquifer from deeper sand aquifers within the glaciolacustrine sequence may have been breached by original plant construction activities, allowing downward migration of tritium from the contaminated surficial aquifer. Alternatively, a naturally occurring window in the stratigraphy, possibly in the form of a lens of sand within the upper part of the glaciolacustrine sequence (or till), may have allowed communication between the shallow sediments in the vicinity of the SFP/IX Pit complex and deeper impacted sand aquifers. The sand aquifers interlayered within the glaciolacustrine sequence have much higher hydraulic conductivities than the surrounding sediments and provide a pathway through which the dominant flow occurs within this sequence. Figures 2-9(c) and 2-9(d) map the tritium plume for the intermediate-depth aquifer. RAIs #51 to 56

Because the sand aquifers within the glaciolacustrine sediments may be discontinuous and the silty matrix of the glaciolacustrine unit has a relatively low hydraulic conductivity, circulation of groundwater flow within this unit is relatively restricted and net groundwater flow through the intermediate depth system is comparatively slow. Therefore, tritium has not been flushed from these deeper sand aquifers as quickly as it has from the shallow system.

The ultimate fate of the tritium impacted groundwater is to flow down the natural hydraulic gradient and discharge to the Deerfield River. The rate of that flow is greatest in the stratified drift aquifer, which has resulted in more flushing of the shallow aquifer by groundwater recharge infiltrating from the surface and mixing with non-impacted groundwater flowing from areas upgradient of the tritium source. The plume of tritium within the glaciolacustrine sequence is also moving toward the Deerfield River, but at a slower rate than the plume in the shallow aquifer. Figures 2-10(a) through 2-10(e) map cross sections showing the extent and concentration of the tritium plume vertically, in both the shallow and intermediate-depth aquifers.

Groundwater potentiometric maps for the shallow (stratified drift), intermediate depth (glaciolacustrine) and bedrock aquifers in July and November 2003 are provided in Figures 2-11 through 2-16. Groundwater flow directions are shown on each map. The hydraulic gradient can be determined between any two points on each map by noting the groundwater elevations at the points of interest and dividing the difference between these elevations (in feet) by the horizontal distance between the points (in feet).

Since these potentiometric maps were produced, the ongoing groundwater monitoring investigation has revealed that groundwater flow within the intermediate depth aquifer may be more complex than depicted. YAEC believes that discrete aquifers comprised of relatively thin sand layers within the glaciolacustrine sediments each have unique potentiometric surfaces. Preliminary evaluation of more recent water level data indicates that the groundwater flow

direction in a sand aquifer at about the 30-foot depth is to the north, while flow in deeper sand at about 100 feet below grade is to the northwest.

Groundwater levels continued to be monitored in all available monitoring wells at the site on a quarterly basis since November 2003. Potentiometric maps for the shallow, intermediate depth and bedrock aquifers will be produced from these more recent quarterly data sets and will be provided in YAEC's next summary report of ongoing hydrogeologic investigations. Comparison of a chronological set of maps for each aquifer will provide an indication of seasonal fluctuations in groundwater levels. Additional wells are being installed that will provide further data for future refinements to the groundwater characterization. RAIs #51 to 56

2.7.4 Ongoing Groundwater Investigations

The preliminary assessment of the groundwater and soil data indicate that the only radionuclide identified in migration towards the Sherman Dam area is tritium. Some of the new wells had tritium concentrations that were in excess of what had been measured for existing wells and in one case greater than the EPA standard for tritium in drinking water. This indicates that the plume may have a more complicated flow path than previously considered. To support futher investigation, the YNPS QA program has been adjusted to account for this new information, and the following activities have commenced to provide further data to assist in the refinement of site characterization:

- Additional wells are being installed onsite and on USGen property.
- Transducers have been added to selected wells to facilitate synoptic measurements.
- A rain gage is being added to the site to monitor rainfall levels.

Although this new information shows concentrations in excess of the EPA drinking water standard, the dose consequence is insignificant and does not change the strategy for going forward towards FSS. Groundwater investigations will continue to be performed. As these investigations progress, actions will be taken, including further analyses or possibly remediation, to ensure that the site release criteria are met.

2.8 Continuing Characterization Activities

2.8.1 Introduction

Surveys of impacted site structures and open land areas will be performed to support final status surveys for surfaces, materials, and soils that will remain at the time of license termination. This includes concrete building floors at ground level, concrete building foundation walls and footings below ground level, asphalt covering the soil in open areas, and soil. Some of the soils to be characterized are located beneath the concrete floors and asphalt. Materials from structures will be dispositioned either under the free release criteria (consistent with the guidance of NRC Circular IEC-81-07, "Control of Radioactively Contaminated Material") or FSS and may be used as backfill. Sub-grade structures that are not part of a designated structural survey area (e.g., concrete support structures) will be evaluated within the overlying open land survey area or subsurface survey area when they are potentially impacted by the migration of sub-surface contamination. Confirmatory spot checks on other such sub-surface structures or objects will validate a non-impacted status where appropriate.

The remaining investigation activities are of two general types:

- Survey used to determine the presence of radioactivity (impacted or non-impacted), or
- Survey performed with final status survey quality requirements that may be used as a final status survey if the release criteria are met.

In the case of the first type of survey, the quality requirements invoked will be specific to the purpose of the investigation. If the survey will be used in support of FSS design elements, then the data quality objective (DQO) process applied to the FSS plan design will be applied to the data quality to ensure it is adequate for the intended purpose.

2.8.2 Characterization Survey Plans Prepared Under a Quality Assurance Project Plan (QAPP)

Characterization Survey planning includes review of the Historical Site Assessment (HSA), scoping survey data, DCGLs, and other relevant information supporting the initial classification of the survey area or unit.

The DQO process described in MARSSIM is implemented by generation of a survey plan. The DQO process is a series of planning steps for establishing criteria for data quality and developing survey designs. The goals of this process are to provide a more effective survey design and a basis for judging the usability of the data prior to collection. DQOs are statements intended to clarify the survey objectives, define the types of data to be collected, and specify the limits on the decision errors used as a basis for establishing data requirements. The impetus of this DQO planning process is a Quality Assurance Project Plan (QAPP). This QAPP integrates all technical and quality aspects of the project and details how these elements will be implemented.

2-9 Technical Basis Document YA-REPT-00-001-03, Radionuclide Selection for DCGL Determination, dated November 5, 2003.

2-10 USGS topographic quadrangle Rowe, Massachusetts – Vermont, 42072-F7-TM-025, dated 1990.

2-11 Technical Basis Document YA-REPT-00-006-03, "Statistical Evaluation of Non-Impacted Area, Evaluation of 137Cs Concentration in Soils of Non-impacted and Reference Areas in the Vicinity of YNPS."

2-12 EG&G 10617-1233, UC-702, "An Aerial Radiological Survey of the Yankee Rowe Nuclear Power Station and Surrounding Area," EG&G Energy Measurements, dated September 1993.

2-13 YRC-1178, Radionuclide Soil Concentrations Surrounding YNPS Resulting from Gaseous Release During Plant Operation, dated March,1998.

2-14 NCRP Report 47 "Tritium Measurement Techniques," dated May 28, 1976.

2-15 NCRP Report 50 "Environmental Radiation Measurements," dated December 27, 1976.

2-16 NCRP Report 81 "Carbon-14 in the Environment," dated May 15, 1985.

2-17 RP 98-20, "Technical Basis Document for Background Concentrations of Cesium-137 in Soil and Sediment," RP 98-20, dated March 3, 1998.

2-18 YA-REPT-00-002-04, "Evaluation of Effluent Releases from Onsite Incineration of Waste," dated May 24, 2004.

2-19 DESD-TD-YR-02-001, "Site Ground Water Data Collection for YNPS Decommissioning," dated February 2002.

2-20 Letter L02-91, from Eric L. Darois (RSCS) to Greg Babineau (YAEC), dated December 12, 2002.

2-21 YA-REPT-01-005-03, "Yankee Nuclear Power Station Report of Radionuclides in Groundwater, Rev. 1 (Third Quarter 2003, Interim)," dated January 2004.

2-22 YA-REPT-00-004-04, "Hydrogeological Report of 2003 Supplemental Investigation," dated March 15, 2004.

Table 2-1

Floor and Total Area of Buildings* and Features

SURVEY AREA	DESCRIPTION	MARSSIM CLASS	FLOOR AREA (m^2)	TOTAL AREA (m^2)	RATIO (total : floor)
SVC-01	NORTH PART OF SERVICE BLDG (CLEAN SIDE)	3	921	921	1
SVC-02	RAD PORTIONS OFSERVICE BLDG AND ANNEX	1	444	444	1
SVC-03	CLEAN SIDE OF SERVICE BLDG ANNEX	3	366	366	1
TBN-01	TURBINE BLDG AND OFFICE PADS	3	1517	1517	1
SPF-01	SPENT FUEL POOL AND TRANSFER CHUTE	1	60	302	5.03
SPF-02	NEW FUEL VAULT	1	95	141	1.48
BRT-01	CONCRETE PEDESTALS, PAD AND ANNULUS	1	2095	2095	1
NSY-01	NORTH AND SOUTH DECON PADS AND FTE	1	224	224	1
NSY-02	IX-PIT, VALVE GALLERY/ PAB STAIRWAY	1	95	390	4.1
NSY-03	SI DIESEL/ACCUMULATOR TANK/BATTERY ROOM	1	380	482	1.12
NSY-04	SAFE SHUTDOWN	1	103	120	1.16
NSY-05	FIRE WATER TANK AND PUMP HOUSE	1	184	184	1
NSY-06	PCA#2 (NEW)	1	219	219	1
NSY-07	WHT / ADT / WASTE GAS PADS	1	390	390	1
NSY-08	NEW SI TANK	1	80	80	1
NSY-09	ELEVATOR SHAFT	1	6	21	4.5
NSY-10	ISFSI	3	985	1078	1.09
NSY-11	CHEM WASTE PIT	1	17	78	4.5
NSY-12	TANK #1 BASE	1	31	31	1
NSY-13	TANK #39 BASE	1	70	70	1
WST-01	PCA #1 (OLD)	1	109	109	1
WST-02	PCA WAREHOUSE	1	604	604	1
WST-03	WASTE DISPOSAL BLDG	1	230	437	1.9
WST-04	COMPCTOR BLDG	1	165	165	1
AUX-01	PAB/ EAST END	1	289	772	2.6
AUX-02	PAB / WEST END	1	130	189	1.45
OMB-01	PUMPHOUSE AND SCREENWELL	3	230	541	2.35
OMB-02	SECURITY GATEHOUSE AND DIESEL GENERATOR	3	270	868	3.2
OMB-03	ADMINISTRATION BUILDING	3	297	798	2.6
OMB-04	WAREHOUSE AND LOADING DOCK PAD	3	625	625	1
OMB-05	FURLON HOUSE	3	432	1076	2.5
OMB-06	SEAL PIT	3	120	329	2.74

* Survey area designations apply to structures that will remain intact.

Table 2-5
Summary of Radiological Conditions of Open Land Areas
(SOF = Sum of Fractions of Proposed Soil DCGLs as submitted) *

SURVEY AREA	DESCRIPTION	MARSSIM CLASS	MEDIUM	SOF (min)	SOF (max)	SOF (mean)
OOL-01	Sherman Pond Sediments	3	Sediment	0.006	0.376	0.140
OOL-02	Yankee Non-Rad Yard Areas	3	Soil	0.005	0.064	0.027
OOL-03	Sherman Reservoir Dam and South Shoreline	3	Sediment Soil	0.208 0.006	0.208 0.411	0.208 0.049
OOL-04	USGen/Sherman Station Overlying Groundwater Plume	3	Sediment Soil	0.012 0.009	0.012 0.049	0.012 0.028
OOL-05	USGen/ Deerfield River Frontage	3	Sediment Soil	0.011 0.048	0.138 0.048	0.041 0.048
OOL-06	Yankee Western Access	3	Sediment Soil	0.009 0.005	0.060 0.114	0.028 0.040
OOL-07	Soils Deposit Area	2		no data		
OOL-08	Yankee Site Exclusion Zone	3	Sediment Soil	0.006 0.005	0.027 0.491	0.014 0.071
OOL-09	Southeast Construction Fill Area	3	Soil Asphalt	0.006 0.020	0.147 0.214	0.030 0.105
OOL-10	ISFSI/Access, Exclusion Zone, Buffer Zone	2	Soil	0.004	0.481	0.034
OOL-11	East RCA Buffer Zone	2		no data		
OOL-12	Warehouse Rail Spur	1	Soil	0.018	0.018	0.018
OOL-13	USGen/Rail Spur Terminus	1	Soil	0.006	0.041	0.019
OOL-14	USGen/Wheeler Brook Frontage	3	Soil	0.006	0.041	0.019
OOL-15	USGen/Sherman Reservoir East Shoreline	3	Soil	0.007	0.017	0.017
OOL-16	Furlon House Parking Lot	3		no data		
OOL-17	Asphalt, Brick and Concrete Storage yard	3		no data		
NOL-01	East Lower RCA Yard	1	Soil	0.006	0.651	0.207
NOL-02	Northeastern Upper RCA Yard	1	Soil	0.005	0.523	0.103
NOL-03	Southeastern Upper RCA Yard	1	Soil	0.005	272.0	5.232
NOL-04	Southwestern Upper RCA Yard	1	Soil	0.007	0.838	0.125
NOL-05	Northwestern Upper RCA Yard	1	Soil	0.005	0.171	0.028
NOL-06	West Lower RCA Yard	1	Soil	0.004	0.491	0.092
NOL-07	ISFSI RCA Yard	3	Soil	0.005	0.021	0.009

* Statistics (min, max and mean) are biased high since sample results are not decay corrected and only samples with results greater than 2 sigma are included in the evaluated population"

Table 2-6

Radionuclides of Concern At YNPS

Radionuclide	Half-Life (in years)
H-3	1.228E01
C-14	5.730E03
Fe-55	2.700E00
Co-60	5.271E00
Ni-63	1.001E02
Sr-90	2.860E01
Nb-94	2.030E04
Tc-99	2.130E05
Ag-108m	1.270E02
Sb-125	2.770E00
Cs-134	2.062E00
Cs-137	3.017E01
Eu-152	1.360E01
Eu-154	8.800E00
Eu-155	4.960E00
Pu-238	8.775E01
Pu-239,240	2.413E04
Pu-241	1.440E01
Am-241	4.322E02
Cm-243,244	2.850E01

Table 2-7

Well Depths and Sampling Results

Well No.	Well Type*	Depth of Well (feet)	3rd Quarter 2003 Results (pCi/l)			4th Quarter 2003 Results (pCi/l)		
			H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta
B-1	Intermediate Bedrock	79	1.36E03	2.80E00	9.16E00	9.00E02	-	6.53E00
CB-1	Shallow Intermediate	25	1.76E03	-	1.35E01	2.14E03	-	1.26E01
CB-2	Shallow Intermediate	24.5	4.11E02	-	1.62E01	1.16E03	-	1.18E01
CB-3	Shallow	13	-	4.50E00	2.48E01	-	-	-
CB-4	Shallow	19	-	-	1.41E01.	-	-	8.20E00
CB-5	Intermediate	59	-	1.54E00	2.44E00	-	-	-
CB-6	Shallow	25	-	-	1.90E01	4.30E02	-	1.14E01
CB-7	Shallow	17	-	-	2.60E01	-	-	-
CB-8	Shallow Intermediate	19	-	3.90E00	1.32E01	-	-	-
CB-9	Shallow Intermediate	24	2.33E03	-	6.70E00	2.62E03	-	7.60E00
CB-10	Shallow	11	9.00E02	-	1.91E01	1.21E03	-	1.25E01
CB-11A	Shallow	20	-	-	1.31E01	2.12E03	8.70E00	3.30E01
CB-12	Shallow	7	-	6.80E00	2.81E01	5.40E02	-	1.05E01
CW-1	Shallow Intermediate	21	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
CW-2	Shallow	20	-	9.20E00	4.25E01	-	-	-
CW-3	Intermediate and Bedrock	23	-	-	1.83E01	1.62E02	-	5.91E01
CW-4	Shallow Intermediate	17	-	-	1.77E01	-	-	-
CW-5	Shallow and Bedrock	16.5	-	-	1.28E01	-	3.50E00	6.60E00
CW-6	Shallow	22	-	-	1.10E01	1.58E02	-	4.01E00

* "Shallow" = outwash; "Shallow Intermediate" and "Intermediate" = till or lacustrine; "Bedrock" = bedrock
† Well has been closed and grouted over, and thus are no longer available for sampling.

Table 2-7

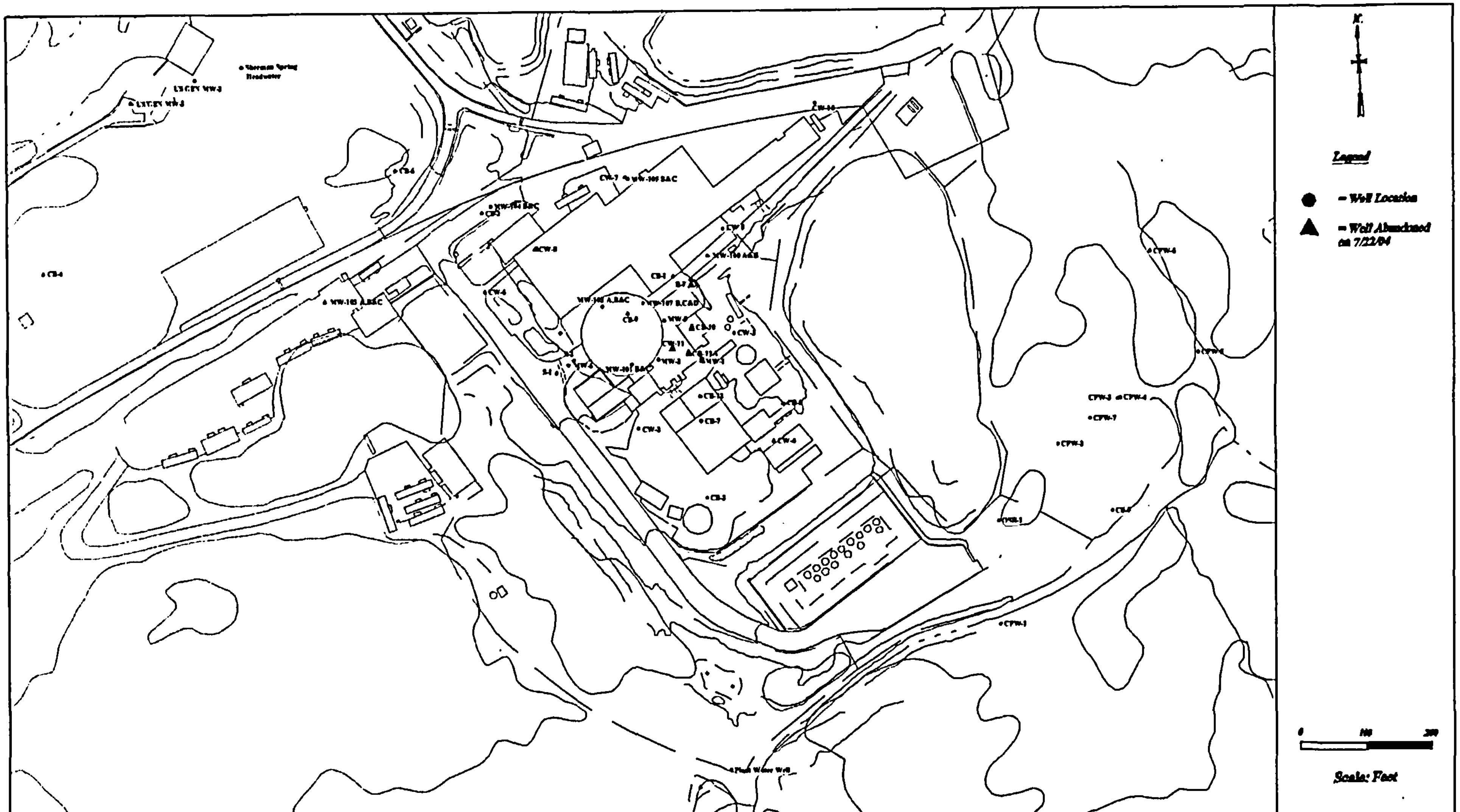
Well Depths and Sampling Results

Well No.	Well Type*	Depth of Well (feet)	3rd Quarter 2003 Results (pCi/l)			4th Quarter 2003 Results (pCi/l)		
			H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta
CW-7	Shallow Intermediate	31	-	2.50E00	1.13E01	-	-	-
CW-8	Shallow Intermediate	26	-	-	1.11E01	-	-	-
CW-9	Shallow Intermediate	17	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
CW-10	Bedrock	30	-	4.20E00	1.16E01	-	-	-
CW-11	Shallow	9	3.67E03	-	8.60E00	1.85E03	-	1.02E01
DW-1	Bedrock	280	-	-	3.89E00	-	-	-
MW-1	Shallow Intermediate	21	-	3.30E00	3.39E01	5.80E02	-	2.21E01
MW-2	Shallow	17	1.25E03	-	8.30E00	1.78E03	-	1.11E01
MW-3	Shallow	20	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
MW-5	No log available	20	3.81E03	-	9.00E00	2.99E03	-	7.50E00
MW-6	No log available	17	-	5.64E00	1.05E01	2.14E02	3.42E00	8.90E00
NSR-1	Shallow and Bedrock	23	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
OSR-1	Shallow	13	-	-	7.50E00	-	-	-
CFW-1	No log available	8	-	-	2.97E00	2.66E02	1.97E00	-
CFW-2	No log available	20	-	-	7.37E00	-	-	3.10E00
CFW-3	No log available	34	-	-	6.44E00	-	1.93E00	9.68E00
CFW-4	No log available	53	-	2.70E00	6.70E00	-	2.50E00	8.80E00
CFW-5	No log available	5	-	-	4.80E00	-	2.20E00	5.20E00
CFW-6	No log available	6	-	-	4.70E00	-	-	2.30E00
CFW-7	No log available	Not known	-	-	7.60E00	-	1.70E00	2.60E00
MW-100A	Shallow	20	-	3.70E00	1.02E01	-	-	-
MW-100B	Bedrock	43	2.50E02	3.30E00	1.31E01	-	-	-
MW-101B	Bedrock	152	-	-	3.90E00	2.52E2	3.15E00	1.27E01

Table 2-7

Well Depths and Sampling Results

Well No.	Well Type*	Depth of Well (feet)	3rd Quarter 2003 Results (pCi/l)			4th Quarter 2003 Results (pCi/l)		
			H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta
MW-101D	Intermediate	99	-	9.20E00	2.58E01	-	-	9.50E00
MW-102A	Shallow	38	4.58E03	-	4.80E00	4.91E03	-	2.71E00
MW-102B	Bedrock	130	3.90E02	-	5.20E00	-	1.60E00	5.15E00
MW-102C	Intermediate	99	5.75E03	-	5.20E00	6.59E03	2.13E00	3.42E00
MW-103A	Shallow	25	3.50E02	4.20E00	1.28E01	-	-	9.35E00
MW-103B	Bedrock	295	-	4.10E00	8.90E00	-	1.79E00	1.10E01
MW-103C	Intermediate	125	2.70E02	2.07E00	1.07E01	-	5.10E00	9.30E00
MW-104B	Bedrock	194	-	-	-	-	-	1.13E01
MW-104C	Intermediate	97	-	-	-	-	-	7.20E00
MW-105B	Bedrock	74	4.85E03	-	1.13E01	5.22E03	5.50E00	1.28E01
MW-105C	Intermediate	37	1.86E03	-	9.32E00	3.72E03	2.50E00	8.20E00
MW-107B	Bedrock	110	<2.00E03‡	-	-	-	2.70E00	1.05E01
MW-107C	Intermediate	32	4.8E04	-	-	4.58E04	-	5.00E00
MW-107D	Intermediate	80	9.15E03	-	-	9.71E03	-	1.12E01



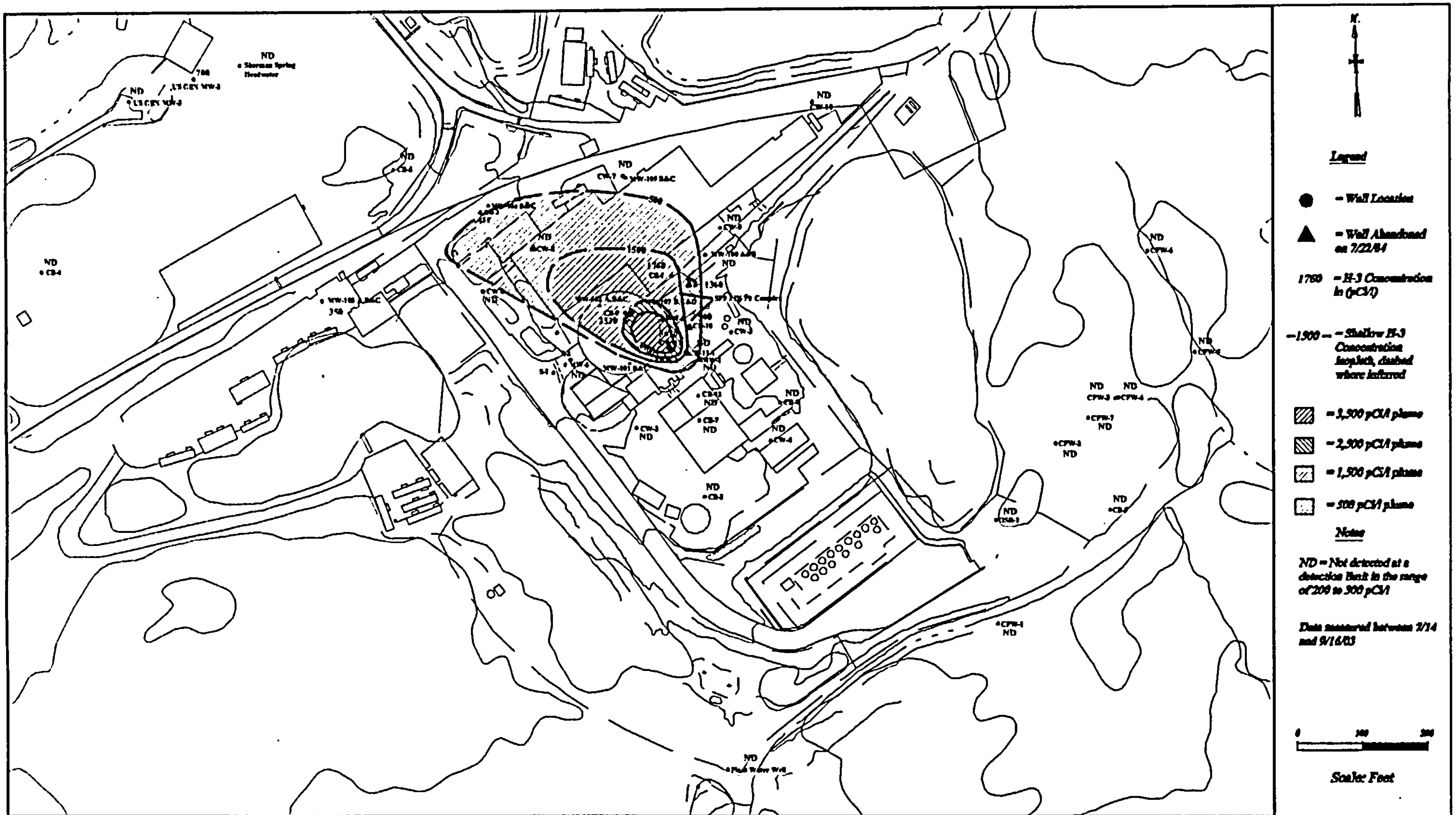
Yankee Nuclear Power Station

Monitoring Well Location Map

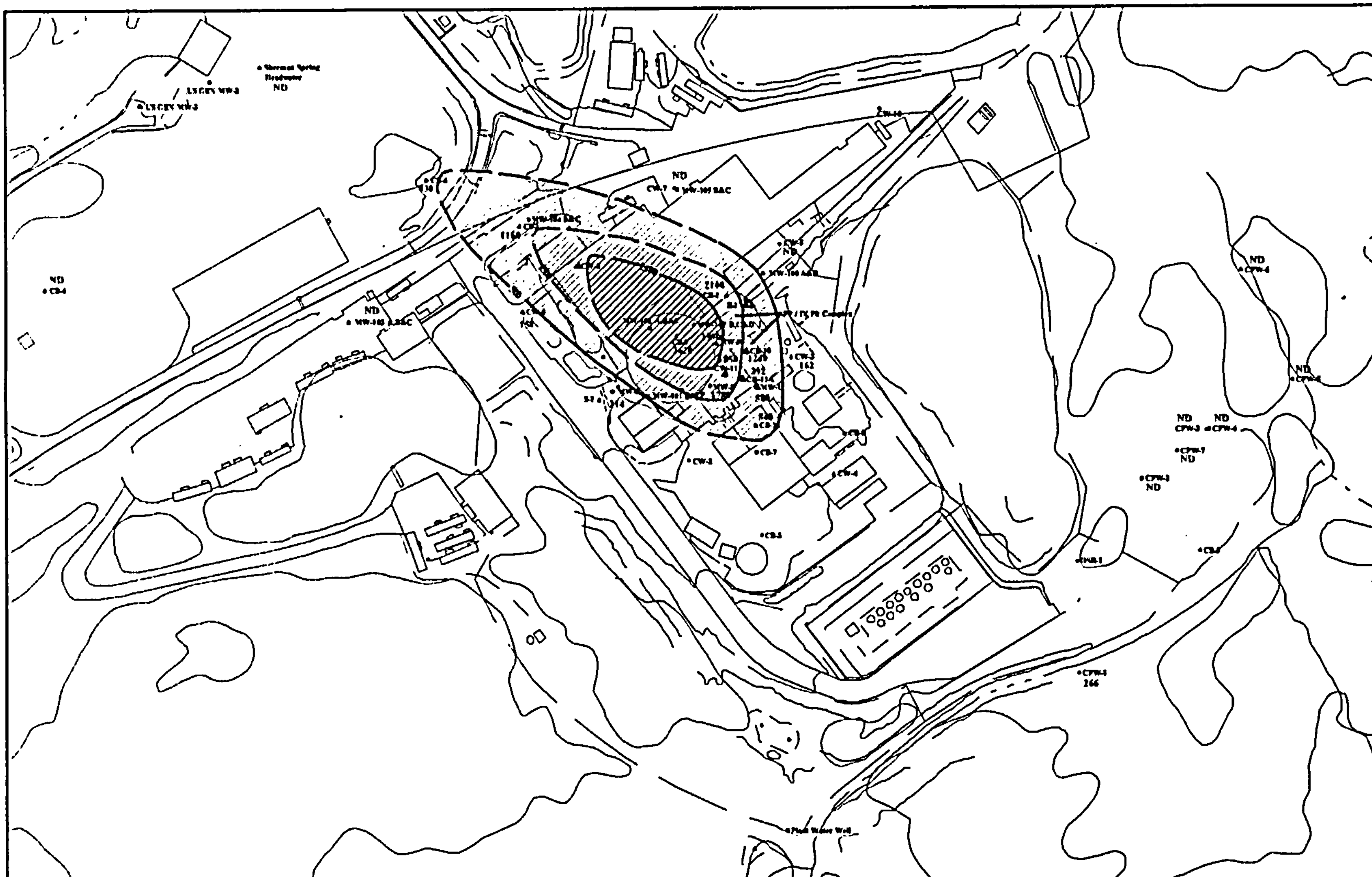
Date: August 2004

Revision: 1

Figure: 2-7



	Yankee Nuclear Power Station Shallow Tritium Plume Map for July 2003	Date: August 2004 Revision: 1	Figure: 2-9a



N

Legend

● = Well Location

▲ = Well Abandoned on 7/22/04

1160 = H-3 Concentration in (pCi/l)

-2500- = Shallow H-3 Concentration Isopleth, dashed where inferred

= 2,500 pCi/l plume

= 1,500 pCi/l plume

= 500 pCi/l plume

Notes

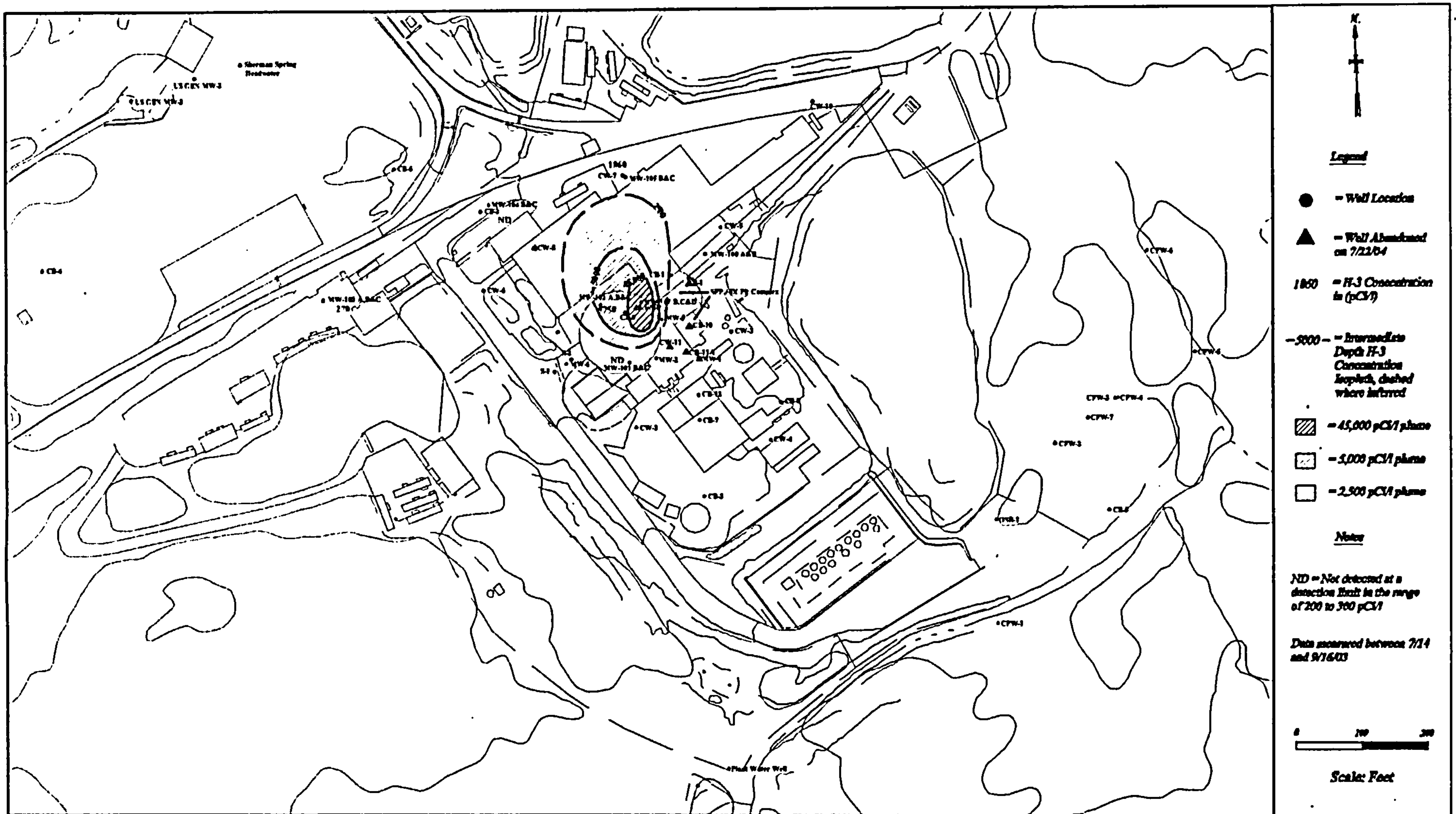
ND = Not detected at a detection limit in the range of 200 to 300 pCi/l

Data measured between 11/5 and 12/1/03

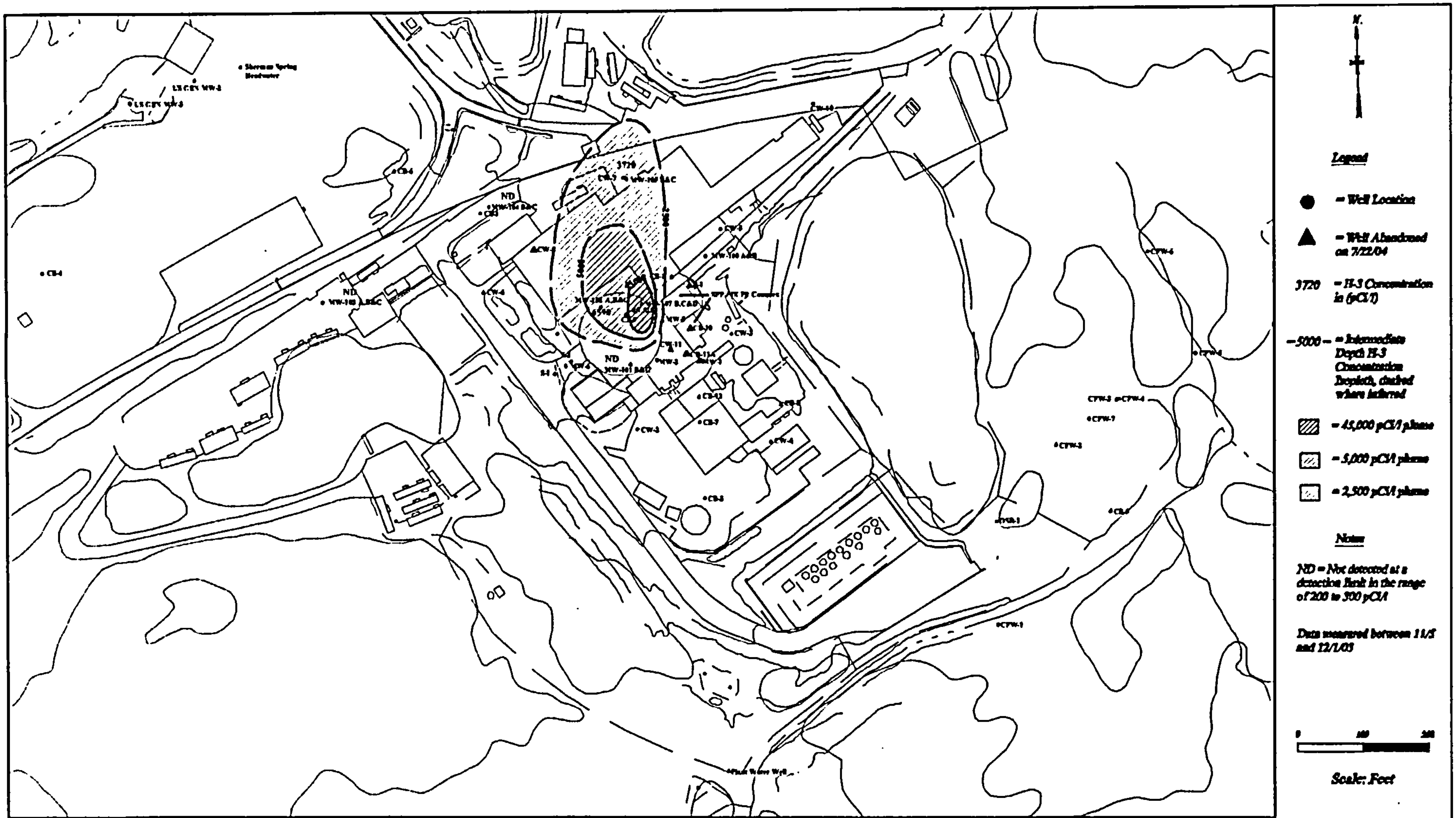
0 100 200

Scale: Feet

Yankee Nuclear Power Station *Shallow Tritium Plume Map* *for November 2003*	YANKEE	*Date: August 2004* *Revision: 1*	*Figure: 2-9b*



Yankee Nuclear Power Station *Intermediate Depth Tritium Plume Map* *for July 2003*	YANKEE	*Date: August 2004* *Revision: 1*	*Figure: 2-9c*



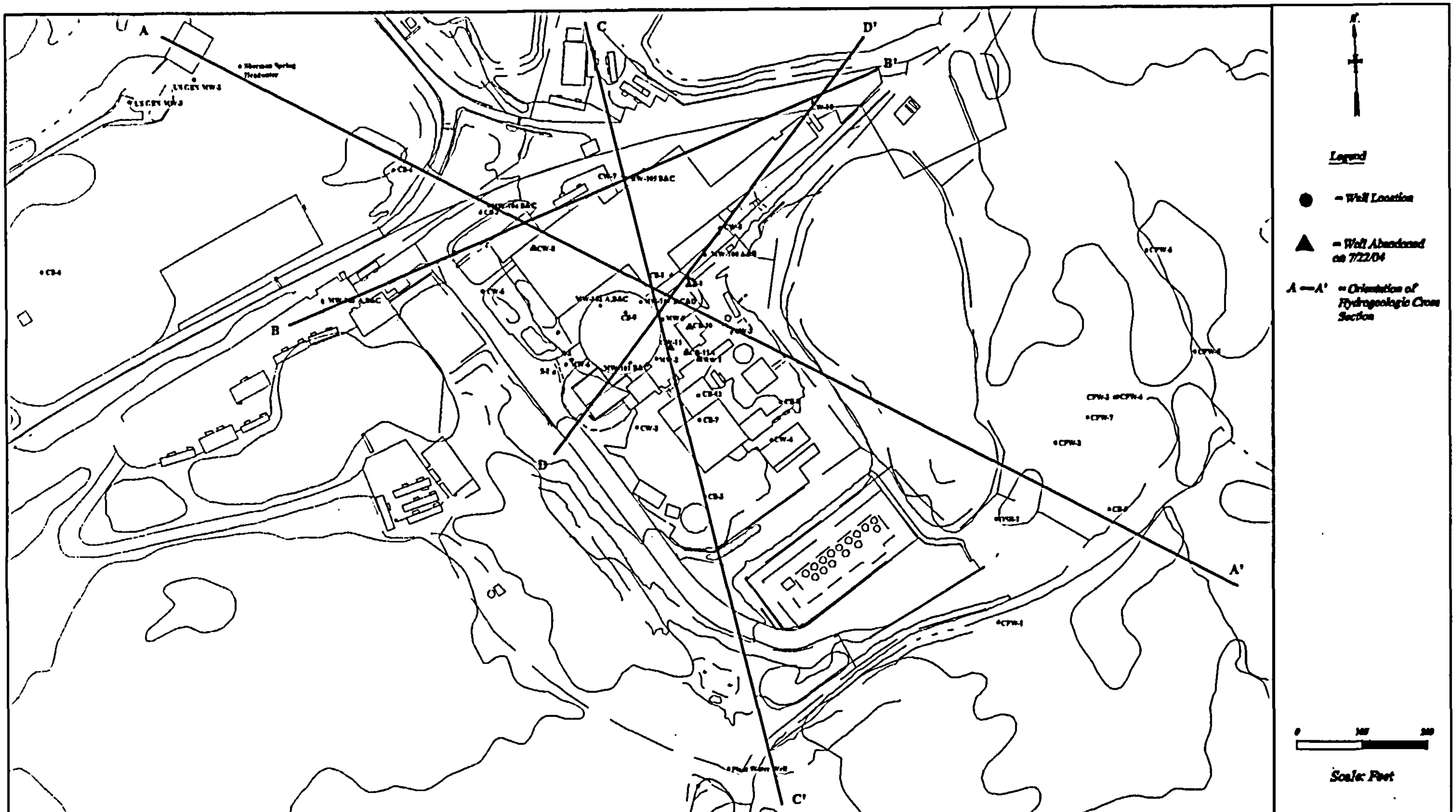
Yankee Nuclear Power Station
Intermediate Depth Tritium Plume Map
for November 2003

YANKEE

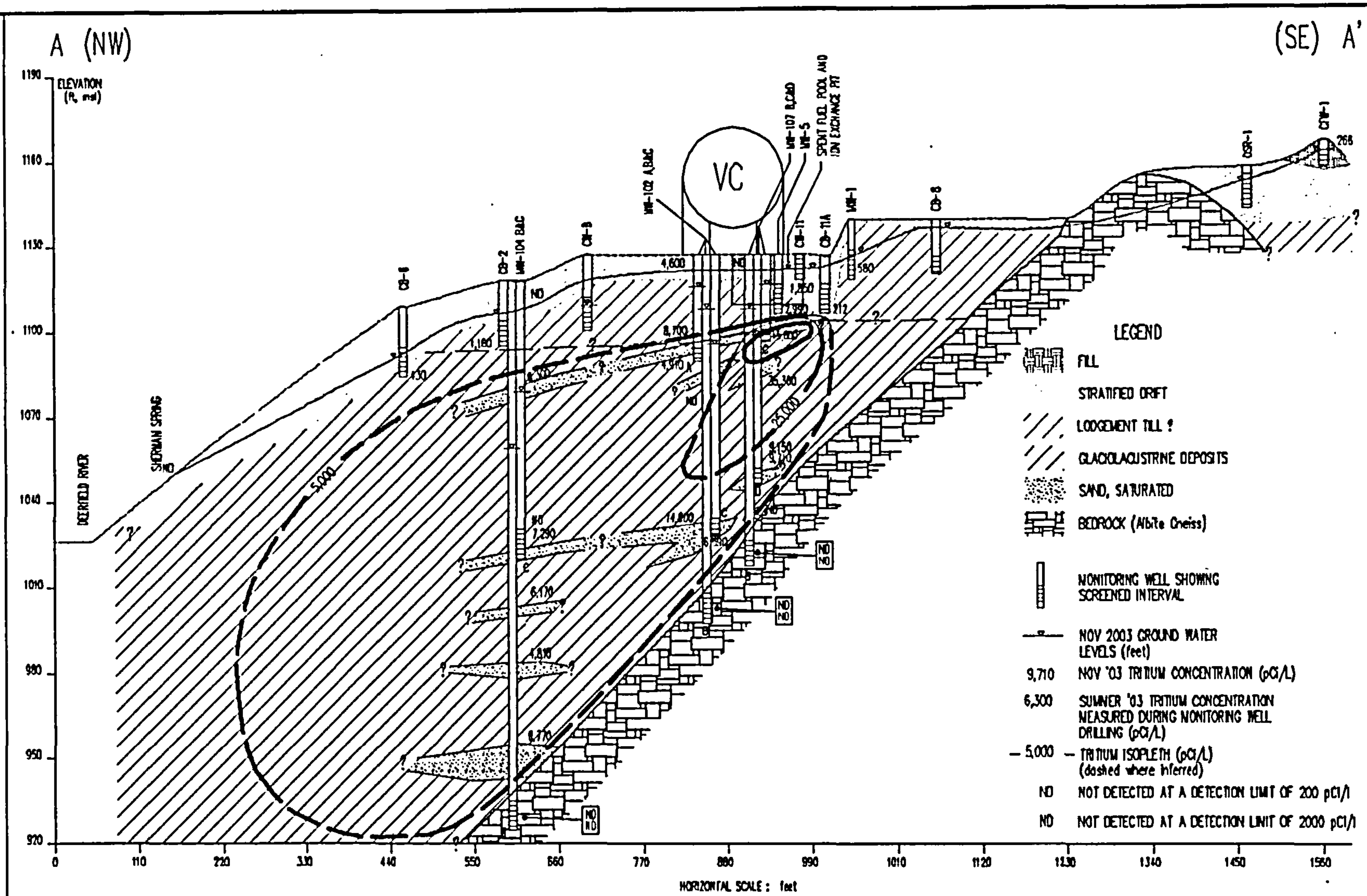
Date: August 2004

Revision: 1

Figure: 2-9d



	Yankee Nuclear Power Station *Location of Cross Section Lines*	YANKEE	*Date: August 2004* *Revision: 1*	*Figure: 2-10a*



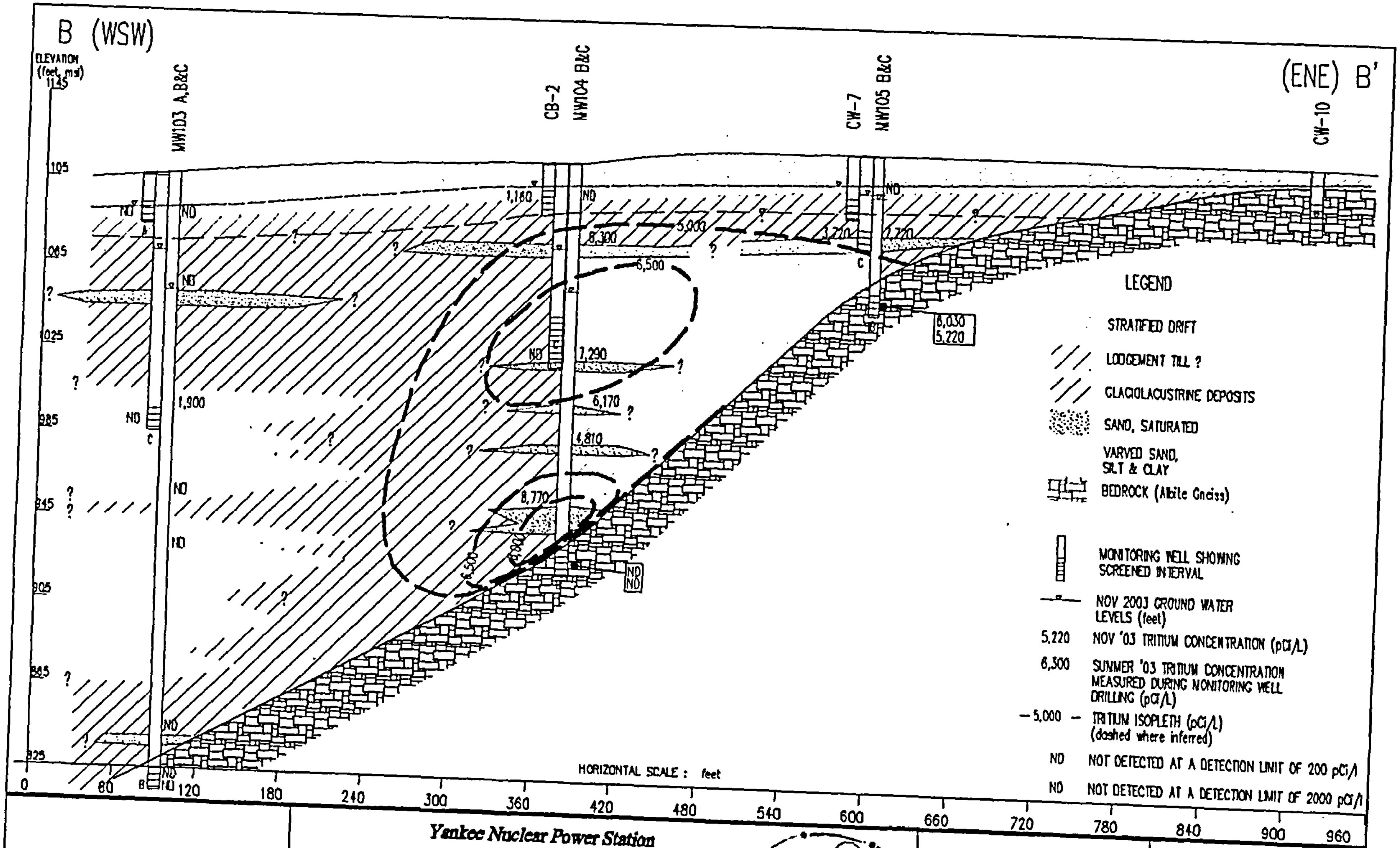
Yankee Nuclear Power Station

Hydrogeologic Cross Section A - A'

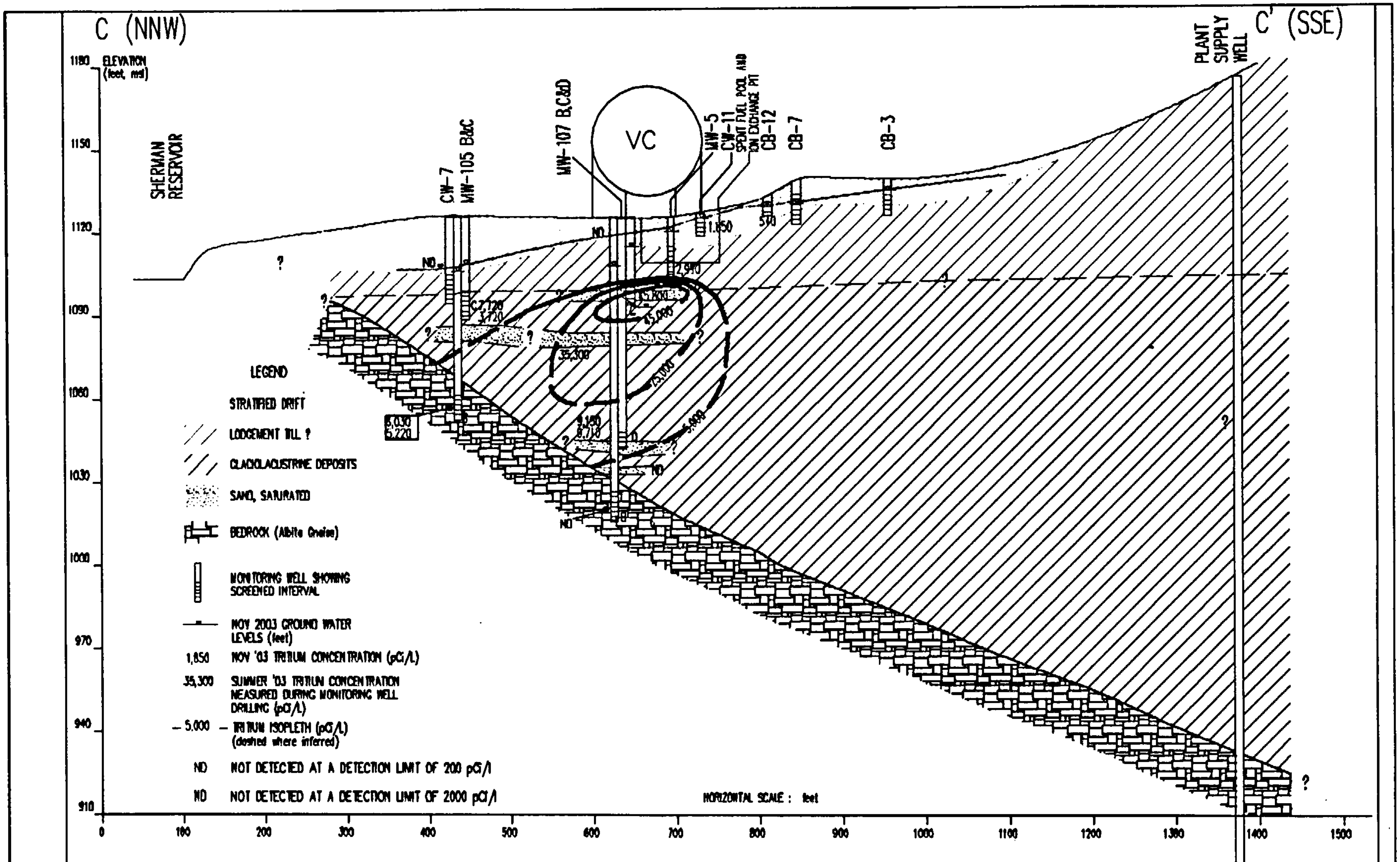
Date: August 2004

Revision: 1

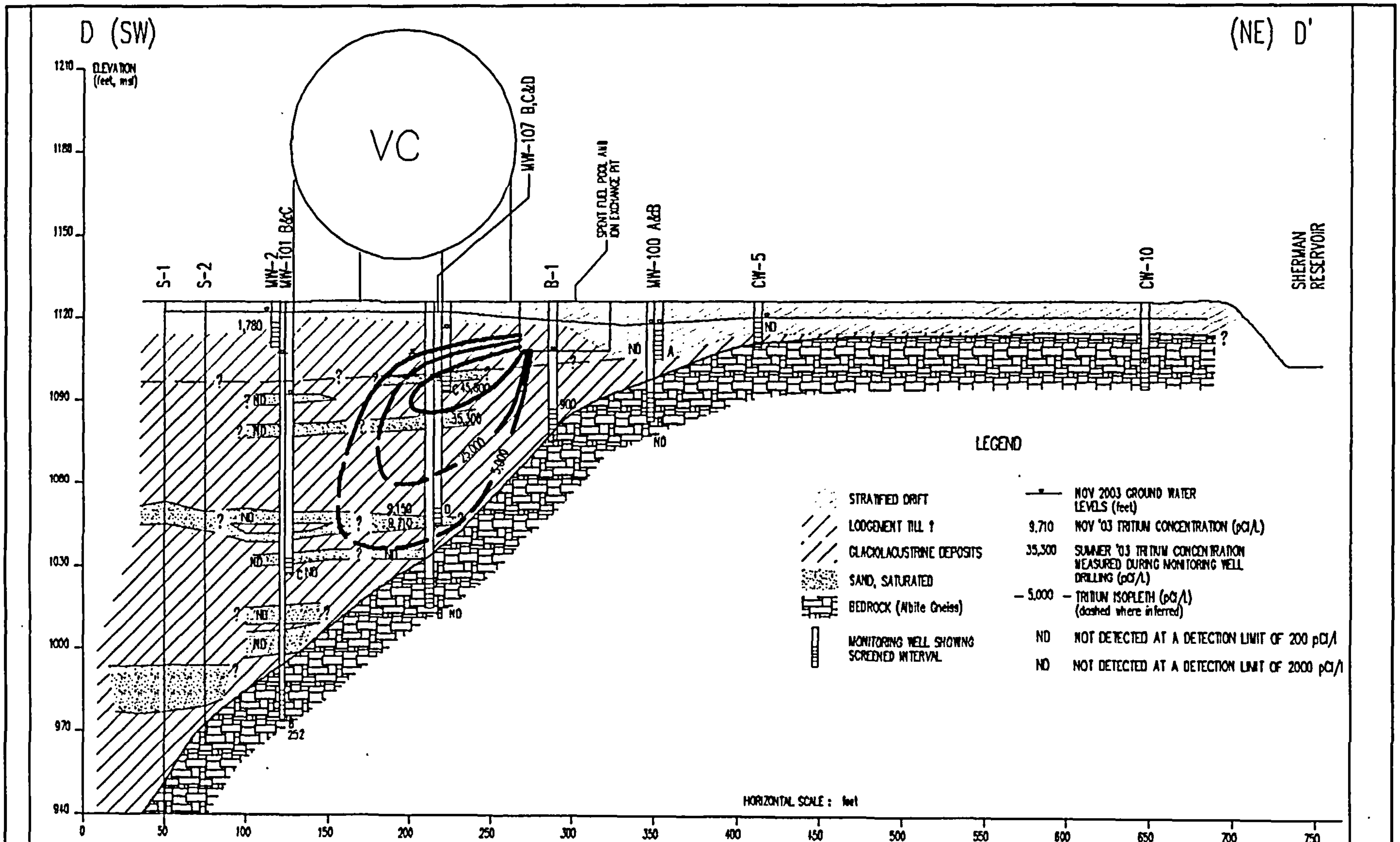
Figure: 2-10b



	Yankee Nuclear Power Station *Hydrogeologic Cross Section B - B'*	*Date: August 2004* *Revision: 1*	*Figure: 2-10c*



	Yankee Nuclear Power Station **Hydrogeologic Cross Section C - C'**	**Date: August 2004** **Revision: 1**	**Figure: 2-10d**



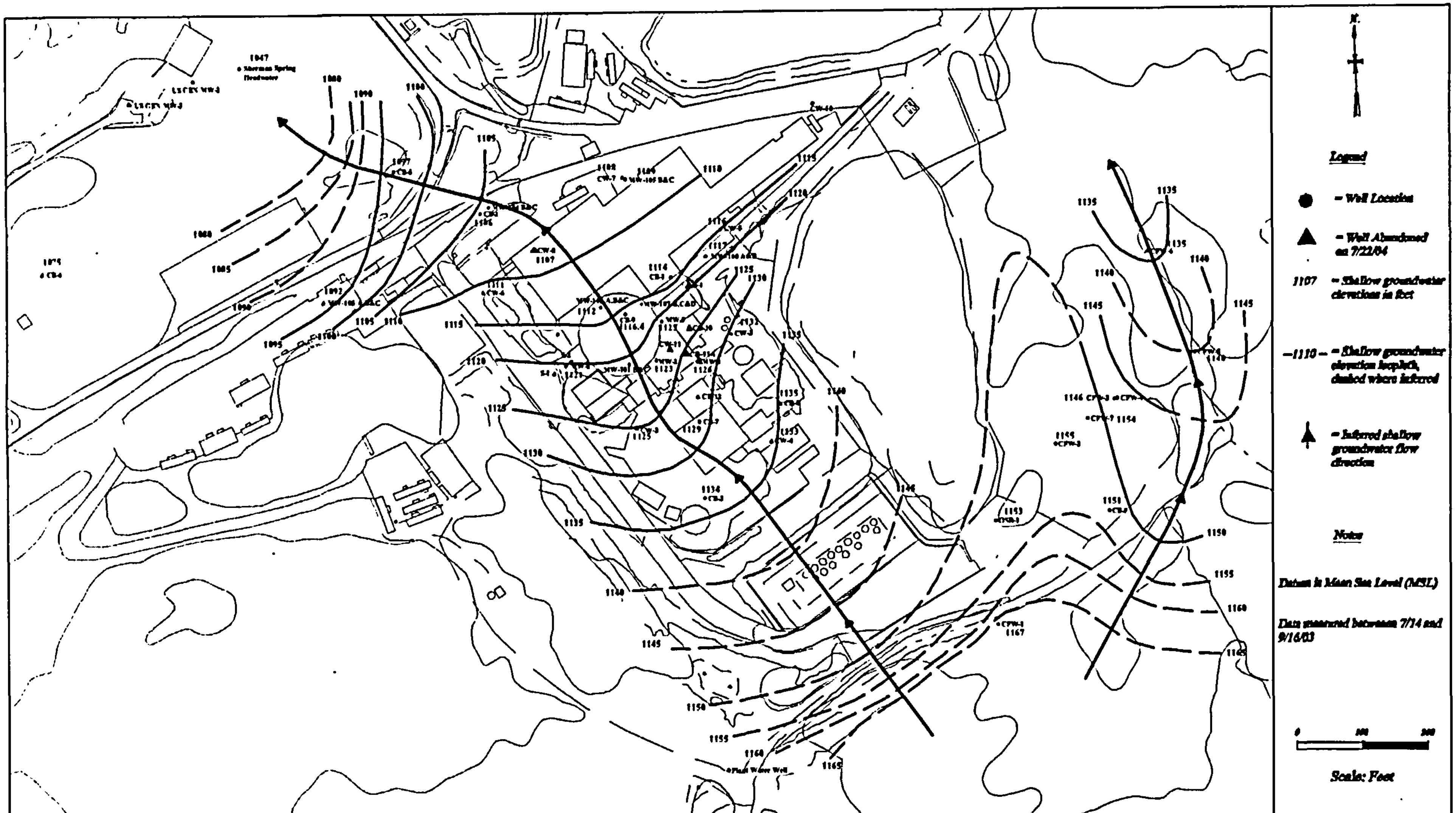
Yankee Nuclear Power Station

Hydrogeologic Cross Section D - D'

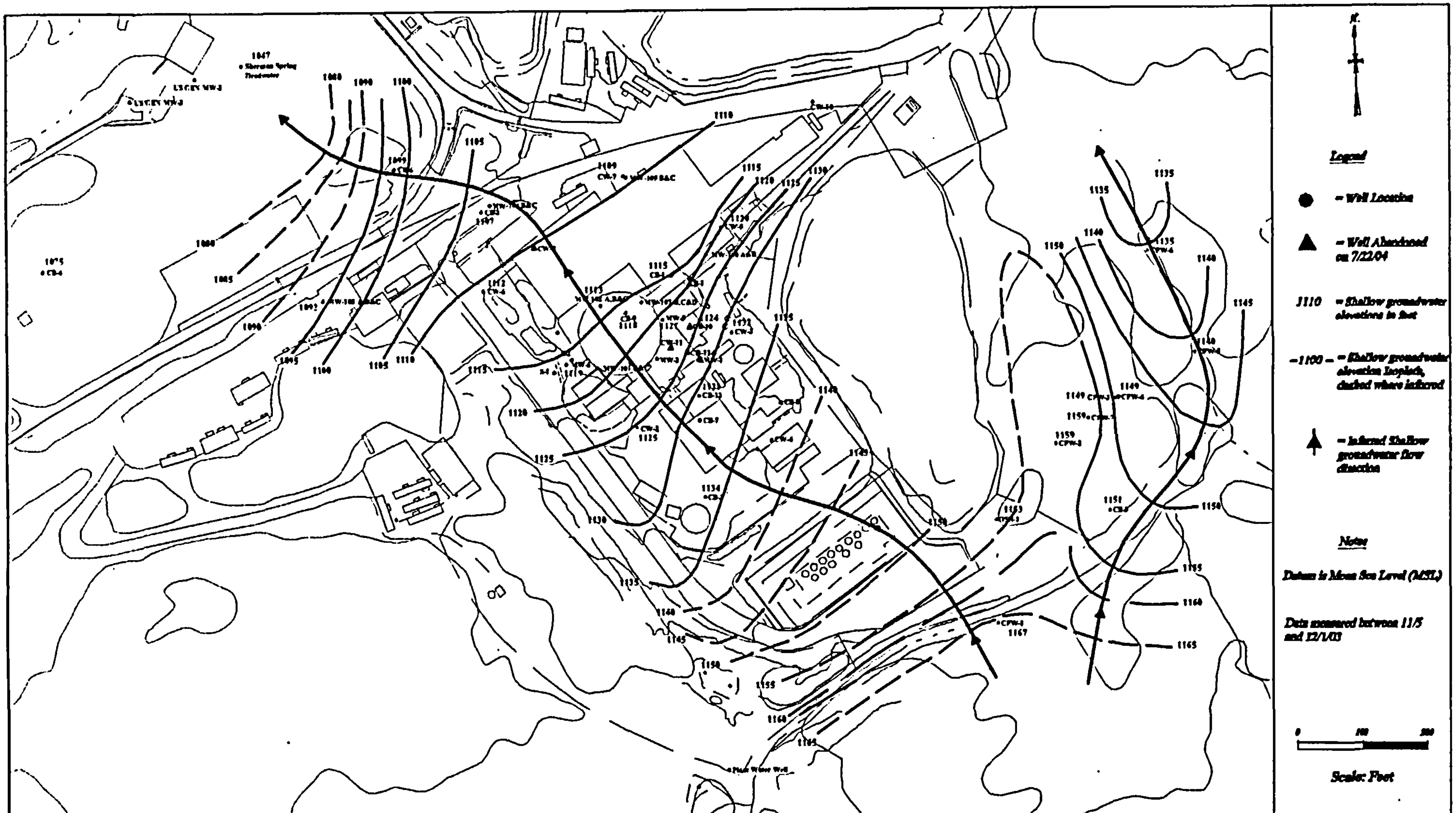
Date: August 2004

Revision: 1

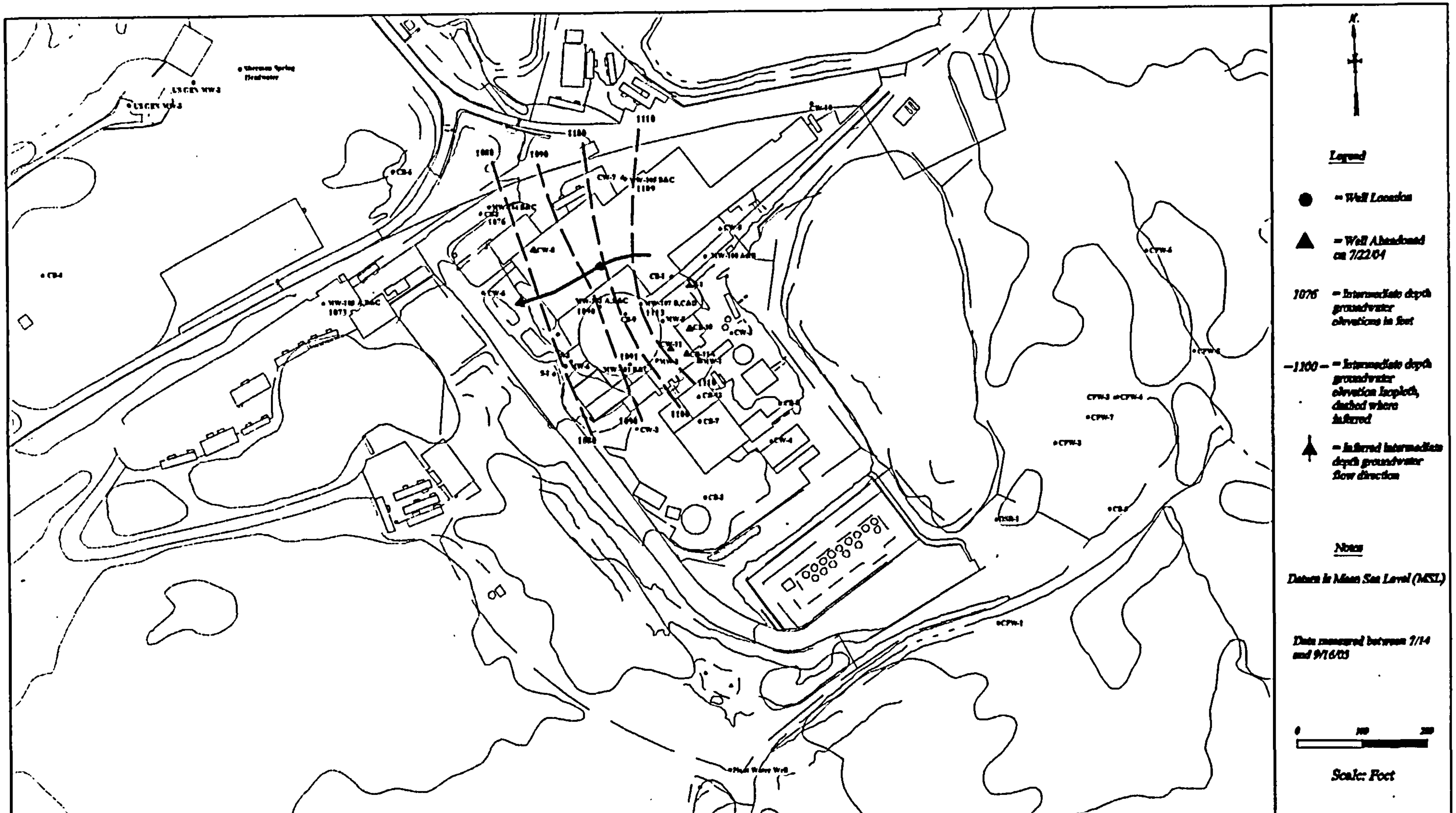
Figure: 2-10e



	Yankee Nuclear Power Station *Shallow Groundwater Elevation Contour Map* *for July 2003*	YANKEE	*Date: August 2004* *Revision: 1*	*Figure: 2-11*



Yankee Nuclear Power Station Shallow Groundwater Elevation Contour Map for November 2003	YANKEE	Date: August 2004 Revision: 1	Figure: 2-12



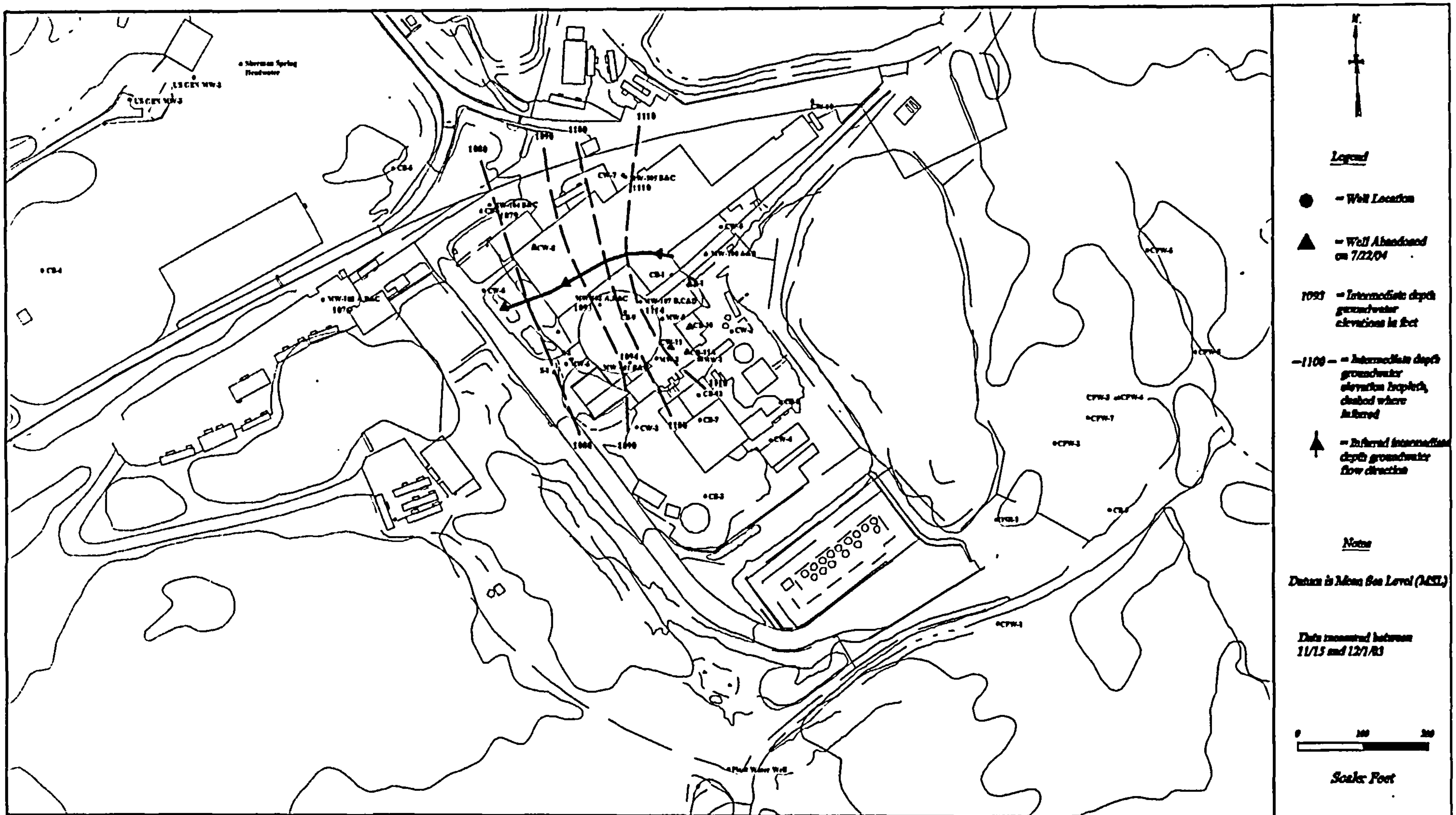
Yankee Nuclear Power Station
Intermediate Depth Groundwater Elevation Contour Map
for July 2003

YANKEE

Date: August 2004

Revision: 1

Figure: 2-13

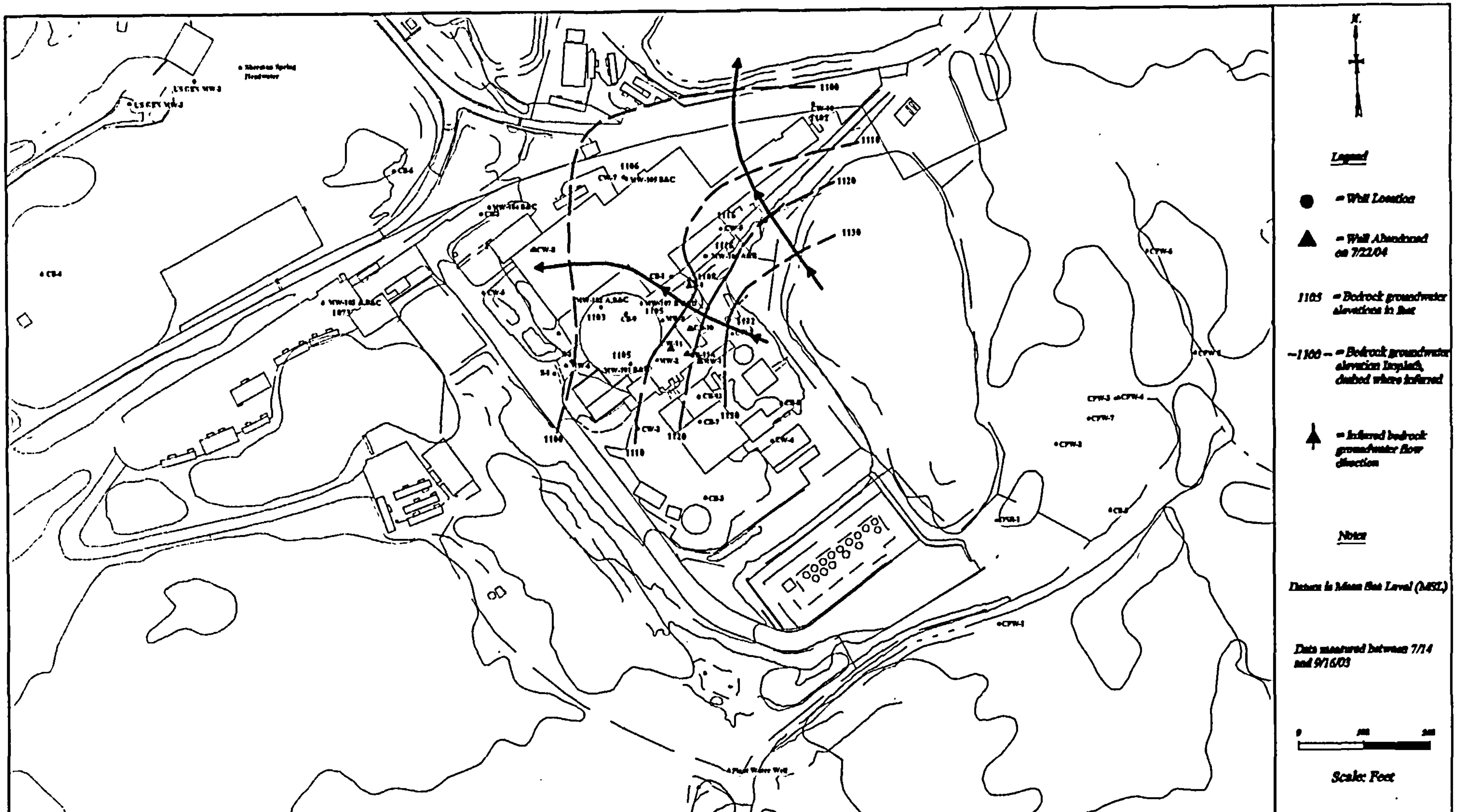


Yankee Nuclear Power Station
Intermediate Depth Groundwater Elevation Contour Map
for November 2003

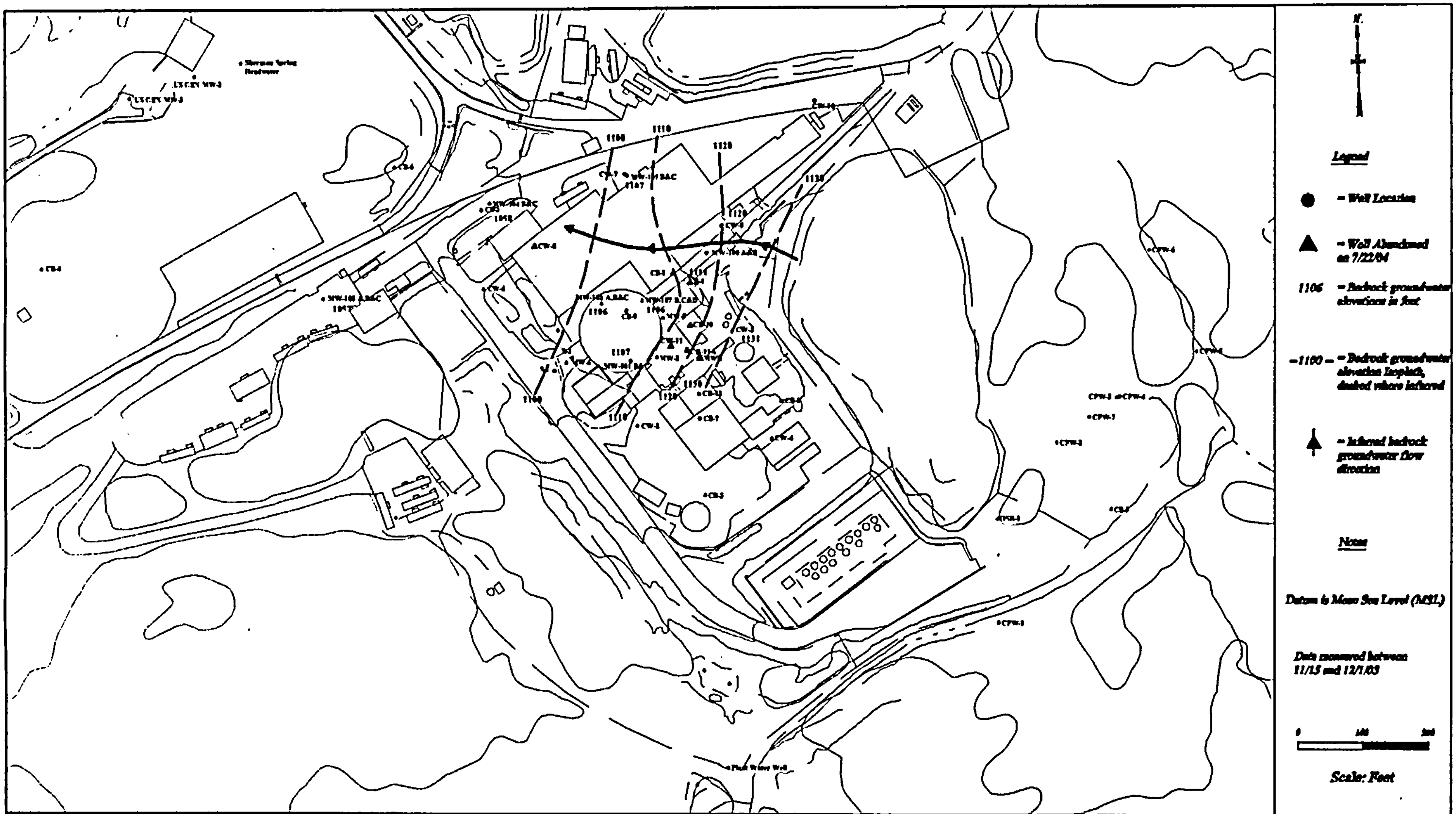
Date: August 2004

Revision: 1

Figure: 2-14



Yankee Nuclear Power Station Bedrock Groundwater Elevation Contour Map for July 2003	Date: August 2004 Revision: 1	Figure: 2-15



	Yankee Nuclear Power Station *Bedrock Groundwater Elevation Contour Map for November 2003*	*Date: August 2004* *Revision: 1*	*Figure: 2-16*

3.2 Decommissioning Approach

Decommissioning activities are being completed in three phases:

Phase 1: Mechanically/electrically isolate the Spent Fuel Pool, remove SSCs not supporting fuel storage, and remove fuel and GTCC waste from the SFP,

Phase 2: Dismantlement and disposition of remaining systems, structures, and components (SSCs), and

Phase 3: Termination of the Part 50 license.

As discussed herein, Phase 1 has been completed. Phase 2 activities are ongoing and their status is described in this section. Phase 3 is intended to occur following completion of all radiological decommissioning activities.

The following are general decontamination and dismantlement considerations that are being incorporated, as appropriate, into the activities for decommissioning the systems, components and structures at YNPS.

- Radiological characterization survey data has been used to identify the systems, structures, and components to be decontaminated and dismantled. The extent of contamination associated with the SSCs is presented in Table 3-1.

- Decommissioning work documents with sufficient detail are being developed, reviewed, and approved in accordance with project and plant programs and procedures.

- Plant tag-out procedures are being used to de-energize electrical and control equipment, isolate, and drain fluid systems, and isolate and depressurize pneumatic systems. Radiation Protection procedures will be used to ensure compliance with radiological requirements for contamination control and worker protection and ALARA programs. Occupation safety standards will be observed.

- Components are being identified prior to removal. The components are then removed using the techniques and methods as specified in the decommissioning work packages. There the components are either decontaminated or shipped to a low-level radioactive waste disposal facility or, if appropriate, shipped to an approved landfill.

- Contaminated structural steel components, on which a volume reduction process is being applied, may be moved to a processing area and packaged into containers for shipment to an off-site waste processing facility.

- Remaining portions of basements and slabs will be perforated to allow for groundwater and/or surface water infiltration.

- Remaining buried contaminated components (e.g., piping, drains, and conduit) are being excavated. After excavation, the components will be examined to ensure that they are physically sound prior to cutting and removal. Most buried contaminated piping is located in steel conduits (i.e., pipes enclosed in pipes). Contamination controls will be modified as necessary if the components are significantly degraded.

- Once decommissioning and/or remediation activities have been completed, and prior to final status survey, isolation and controls will be implemented as described in Section 5.4.5.

- A final status survey will be performed to verify removal of contamination to below release levels.

- Coatings will be removed, as required by local, state, and federal regulations. PCB paints will be removed from exposed concrete surfaces as required by the Alternate Method of Disposal Authorization (AMDA) requirements prior to demolition of the structure, as authorized by the EPA on October 8, 2002 (Reference 3-4) and subsequent changes thereto.

3.2.1 Phase 1 Activities

Since 1993 Yankee has removed and disposed of the steam generators, pressurizer, and the reactor vessel. The reactor vessel internals, which are greater-than-Class-C (GTCC) waste, remain onsite and are stored at the site's independent spent fuel storage installation (ISFSI).

The Spent Fuel Pit (SFP) and other systems associated with fuel storage were electrically and mechanically isolated to create a Spent Fuel "Island" that would not be adversely impacted by other decommissioning activities. The majority of systems and components not required to support the storage of spent fuel have been dismantled and disposed of in accordance with the YNPS Decommissioning Plan and Final Safety Analysis Report. The status of plant SSCs, as of July 2003 is provided in Table 3-2.

Once a Spent Fuel "Island" was established, the focus of site activities shifted to the removal of spent fuel and GTCC waste from the SFP, to the ISFSI. Movement of the fuel and the non-fuel GTCC waste from the SFP to the ISFSI was completed in June 2003.

3.2.2 Phase 2 Activities

After removing the spent fuel and GTCC waste from the SFP, the remaining components of the systems listed below are being dismantled and decontaminated.

- Temporary Waste Water Processing System,
- Radiation Monitoring System,
- Ventilation Systems (Including Vapor Container Ventilation and Purge System),
- Fuel Handling Equipment System,
- SFP Cooling and Purification System,

- Auxiliary Service Water System,
- Demineralized Water System,
- Compressed Air System,
- Electrical System,
- Heating System, and
- Fire Protection and Detection System

After removing systems and components from an area or building, contaminated concrete, steel, and other building materials are being decontaminated or removed. The structures listed below are being decontaminated and/or dismantled during the decommissioning of the SFP Island.

- Yard Area Crane and Support Structure,
- Vapor Container (VC),
- Reactor Support Structure,
- VC Polar Crane,
- Radiation Shielding,
- Pipe Chases,
- Fuel Transfer Chute,
- Ion Exchange Pit,
- Primary Vent Stack,
- Spent Fuel Pit and SFP Building,
- New Fuel Vault,
- Primary Auxiliary Building,
- Waste Disposal Building,
- Safe Shutdown System Building,
- Potentially Contaminated Area (PCA) Storage Buildings and Warehouse,
- Compactor Building
- Service Building and Fuel Transfer Enclosure,
- Miscellaneous Storage Tanks and
- Meteorological Tower.

Upon the completion of Phase 2 activities, all systems and components will have been removed from plant buildings and yard areas (with the exceptions of those supporting spent fuel and GTCC storage in the ISFSI) and disposed of at the appropriate facility. In general, above grade portions of site buildings will be remediated, if necessary, and demolished. Below-grade portions of site structures (elevation 1022'-8" and below) are being remediated to meet the site release criteria or are being removed. Building demolition debris that has been determined to contain "no detectable radioactivity" or has passed a final status survey may be used as backfill on site. Details concerning dismantlement and remediation efforts are provided in the subsections to follow.

Following submittal of the License Termination Plan, Final Status Surveys will be conducted to verify that structures and open land areas meet the release criteria. Independent verification of the results by the NRC will allow for the release of the individual surveyed structures and open land areas. In order to facilitate remediation, the facility superstructures may be demolished

3.2.2.2.2 Vapor Container

The Vapor Container (VC) is a spherical steel structure that surrounds the Reactor Support Structure. It is located about 23 feet above grade and is supported by 16 steel columns. The steel columns are supported by reinforced concrete pedestals.

The Vapor Container provides lateral support to the VC Service Elevator Tower and the PVS. Attachments are limited to minor platform framing, exterior stairs, and lightly loaded supports for pipes and cable trays.

The following considerations are specific to the dismantlement and decontamination of the VC:

- Piping penetrations should be cut off as close as practicable to the VC shell when the process system which passes through it is dismantled.

- Electrical penetrations should be cut off as close as practicable to the VC shell after all cables in the penetration have been disconnected and removed.

- Platforms, ladders, and stairs along with the supporting steel members should be removed in conjunction with area decontamination and dismantlement activities.

The VC is no longer needed for contamination isolation and will be demolished, decontaminated, and removed from the site.

3.2.2.2.3 Reactor Support Structure

The Reactor Support Structure is a reinforced concrete structure which supports the polar crane. The Reactor Support Structure consists of two concentric concrete cylinders. The cylinders are connected together with reinforced concrete radial walls which formed compartments for the Main Coolant Loops, pressurizer, and Equipment Hatch. The compartments are covered by a reinforced concrete charging floor. The charging floor is composed of removable concrete slabs which allow crane access to the compartments.

The Reactor Support Structure is supported on eight reinforced concrete steel encased columns which penetrate the VC shell. The VC penetrations are sealed by stainless steel expansion joints. An annular space is provided to permit the VC and internal concrete structure to move independently.

The following considerations are specific to the dismantlement and decontamination of the Reactor Support Structure:

- The steel casings of the support columns that form the shell to the expansion joint should be removed to permit access to the concrete columns.

- The concrete columns will be decontaminated, as required.

- All contaminated equipment was removed prior to decontamination or removal of concrete on the walls, floors, and ceilings.

- Concrete and reinforcing bar on the inner section of the inner support wall, which was behind the Neutron Shield Tank, was slightly activated and has been partially removed.

- The concrete and reinforcing around the Main Coolant Loop penetrations may also be slightly activated. The removal zone was determined using cored samples of the concrete reinforcing.

The RSS will be demolished. Debris meeting the "no detectable activity" criteria or passing a final status survey may be used as backfill on site.

3.2.2.2.4 VC Polar Crane

The VC Polar Crane was used to support refueling and maintenance-related activities inside the VC. The crane was originally designed for the installation of the Reactor Vessel and Steam Generators. However, crane capacity was reduced during plant operations by converting one hook to a smaller capacity to increase hook travel speed. The smaller hook was replaced with a larger hook as part of the Component Removal Project, returning the Polar Crane to its original capacity. After the project was completed, the larger hook was again replaced with the smaller hook.

The crane consists of a bridge which rides on a 75-foot diameter crane rail with a common trolley rigged with two hooks. The rated capacity of the bridge and common trolley is 150 tons. The installed hooks have rated capacities of 75 tons (Hook No. 1) and 15 tons (Hook No. 2). The VC Polar Crane may be used to support decontamination and dismantlement activities in the VC.

The following considerations are specific to the decontamination and dismantlement of the VC Polar Crane:

- The VC Polar Crane should be decontaminated at the time of decontamination of the VC shell or should be removed and decontaminated at a designated area/facility.

- The hoist, trolley, motors, and control cab should be removed from the girders.

The VC Polar Crane will be dismantled and disposed of as waste.

3.2.2.2.5 Radiation Shielding

Radiation shielding is installed for both personnel and equipment protection. The radiation shielding is comprised of several categories according to function:

- Primary Shielding

- Filling the annular space between the Fuel Transfer Chute pipe and the SFP penetration pipe with grout,
- Removing one section of the Fuel Transfer Chute pipe uphill of the Lower Lock Valve (LLV),
- Installing a blind flange cap on the LLV,
- Erecting permanent form work and placing a concrete barrier in the LLV pit, and
- Installing metal plates above and below the LLV pit to preclude personnel access to this area.

The Fuel Transfer Chute will be removed to elevation 1022'-8", and a temporary cover will be installed on the remaining lower chute segment. There are currently no additional decontamination or dismantlement considerations specific to the Fuel Transfer Chute. The Fuel Transfer Chute will be demolished and disposed of as radioactive waste. The remaining lower chute segment will be demolished with the Spent Fuel Pit.

3.2.2.2.8 Ion Exchange Pit

The Ion Exchange Pit (IX Pit) is a reinforced concrete structure that contained the ion exchange vessels used to purify the SFP and Main Coolant System. The IX Pit is no longer in service, and some decontamination and dismantlement activities have commenced.

The IX Pit shares a common wall with the SFP, and thus, no major dismantlement activities could be performed on this common wall until the SFP had been drained.

The IX Pit metal hatch covers will be removed and disposed of as waste. In general the IX Pit walls will be demolished to elevation 1022'-8", with the exception of the south wall and the east wall which will be removed to elevation 1035'-8". The remaining earth-retaining walls will be stabilized as required by engineering analysis. There are currently no additional decontamination or dismantlement considerations specific to the IX Pit. Debris from demolition of the IX Pit may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.9 Primary Vent Stack

The Primary Vent Stack is a steel stack that vents monitored airborne releases from the Ventilation System and the VC Ventilation and Purge System. The bottom of the stack is supported by a steel frame that is supported by the PAB. The Primary Vent Stack may be used during the dismantlement period to support decommissioning activities, and as needed to vent air processed by both the Ventilation System and VC Ventilation and Purge System. There are currently no additional decontamination or dismantlement considerations specific to the Primary Vent Stack. The Primary Vent Stack will be dismantled and disposed of as radioactive waste.

3.2.2.2.10 Spent Fuel Pit and SFP Building

The Spent Fuel Pit (SFP) is a reinforced concrete structure that provided underwater storage of irradiated fuel, control rods, and associated fuel transfer equipment. The SFP inside dimensions are approximately 16 feet by 34 feet by 37 feet deep, with a wall thickness that varies between 5 and 6 feet. A stainless-steel linear was later added to the SFP walls and floor to prevent leakage.

The SFP Building is a steel-braced frame, metal-sided structure that supports the superstructure to both the New Fuel Vault and the SFP. The building provides an enclosed work area and contains the Spent Fuel Manipulator Crane, the New Fuel Hoist, and the SFP Cooling System pumps. Roof hatches are provided for equipment and cask access using the Yard Area Crane, which is located directly above the building.

Components and systems will be removed from the SFP and SFP Building. The SFP walls will be demolished to elevation 1022'-8". The support columns will be removed to the top of the concrete foundation. The coatings from remaining interior and exterior surfaces of the SFP will be removed. The liner will be removed and disposed of as radioactive waste.

Decontamination and dismantlement considerations specific to the SFP Building are as follows:

- The SFP liner should be decontaminated before dismantlement.
- The SFP Handling Equipment should be dismantled into more easily managed sections.
- Soil under the SFP will be sampled as a part of site characterization.

The debris from demolition of the SFP and SFP Building may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.11 New Fuel Vault

The New Fuel Vault is a reinforced concrete and concrete masonry structure. The vault is contained within a lower section of the SFP Building. The west and south walls of the New Fuel Vault are common to the SFP and the IX Pit, respectively.

During decommissioning and dismantlement, all systems and components will be removed from the New Fuel Vault. In general the walls of the New Fuel Storage Vault are being removed to elevation 1022'-8", with the exception of the south wall which is being removed to elevation 1035'-8". There are currently no additional decontamination or dismantlement considerations specific to the New Fuel Vault. The debris from demolition of the New Fuel Vault may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.12 Primary Auxiliary Building

The Primary Auxiliary Building (PAB) is a concrete masonry building with two stories and a partial basement at the southeast corner. Systems and components within the PAB have been dismantled and will be removed (including those on the PAB roof slab). In general the PAB walls will be demolished to 1022'-8", with the exception of the south wall and east wall which will be demolished to elevation 1035'-8". The remaining earth retaining walls will be stabilized as required by engineering analysis. There are currently no additional decontamination or dismantlement considerations specific to the PAB. Debris from demolition of the PAB may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.13 Waste Disposal Building

The Waste Disposal Building contained system and structures for processing, packaging, and temporarily storing low-level radioactive waste, prior to shipment offsite. The structure is a steel-framed building with concrete masonry unit walls. Systems have been dismantled and the Waste Disposal Building has been decontaminated. The Waste Disposal building shares common walls with the Warehouse, Potentially Contaminated Area (PCA) Storage Building 1, and the Compactor Building.

Systems and components will be removed from the building. Hazardous materials will be removed. The building will be removed to the top of the floor at elevation 1035'-8". There are currently no additional decontamination or dismantlement considerations specific to the Waste Disposal Building. The debris from demolition of the Waste Disposal Building may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.14 Safe Shutdown System Building

The Safe Shutdown System Building contains the Fire Water Storage Tank (TK-55) Heating Boiler and associated components. The Safe Shutdown Building will be required during the dismantlement period to house the heating boiler and prevent the contents of TK-55 from freezing. The structure is constructed of reinforced concrete walls.

During dismantlement activities, building equipment will be removed and disposed of as waste. The building, itself, will be demolished to the top of floor elevation 1034'-0". There are currently no additional decontamination or dismantlement considerations specific to the Safe Shutdown System Building. The debris from demolition of the Safe Shutdown System Building may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.15 Potentially Contaminated Area (PCA) Storage Buildings and Warehouse

There are three major areas located on the plant site for the storage of radioactive/hazardous materials and waste awaiting shipment:

<u>PCA Storage Building 1:</u> PCA Storage Building 1 is used primarily for the storage of low-level radioactive material prior to shipment. The structure is comprised of concrete masonry walls.

PCA Storage Building 2: PCA Storage Building 2 is used for the storage of contaminated tools and equipment. The structure is constructed of un-insulated corrugated metal panels.

PCA Warehouse: The PCA Warehouse is used for storage of low-level radioactive waste, waste containers, and contaminated equipment prior to shipment. The structure is a steel-framed building, with reinforced concrete masonry unit walls.

These storage areas may be used during the site dismantlement period to support radioactive material processing and storage. These structures will be decontaminated after all radioactive/hazardous materials stored within these areas have been permanently removed.

Once these structures are no longer required, systems and components will be removed from the buildings and disposed of as radioactive waste. These buildings will be demolished to elevation 1035'-8". There are currently no additional decontamination or dismantlement considerations specific to the PCA Storage Buildings or Warehouse. Debris associated with demolition of these structures may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.16 Compactor Building

The Compactor Building contained two solid waste compactors and provides a packaging area for radioactive waste shipping containers. The structure's walls are constructed from reinforced concrete masonry units. The Compactor Building may be required during the dismantlement period to reduce exposure to radiation and the spread of contamination. The structure will be removed after contaminated material processing is no longer required.

The Compactor Building will be demolished to the top of the floor at elevation 1035'-8", after components and systems are removed. Hazardous materials will be removed from the remaining portions of the structure. There are currently no additional decontamination or dismantlement considerations specific to the Compactor Building. The debris associated with the demolition of the Compactor Building may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.17 Service Building and Fuel Transfer Enclosure

The Service Building is divided into two sections. One of these sections is located in the Radiation Control Area (RCA) of the plant. This section contains the primary side machine shops, control point, primary side chemistry laboratory, counting room, and decontamination showers. The structure's walls are constructed from reinforced concrete masonry units. The building may be required to support dismantlement and decommissioning activities.

The Fuel Transfer Enclosure (FTE) is a relatively new structure that served as the work area for the preparation of the fuel storage canisters, as a part of the overall fuel loading operation. The FTE is a southern extension of the Service Building, under the yard area crane, and immediately adjacent to the SFP Building. It is a steel building that includes the existing North Decon Area,

and the existing welding booth, which served as the access point to the FTE. Access to the FTE by the Yard Crane was provided by a roof hatch. The FTE may also be required to support dismantlement and decontamination activities.

The Service Building and FTE will be demolished to the top of the ground-level floor slab at elevation 1022'-8", after systems and components have been removed. Hazardous materials will be removed. There are currently no additional decontamination or dismantlement considerations for the Service Building and Fuel Transfer Enclosure. The debris associated with the demolition of the Service Building and Fuel Transfer Enclosure may be used as backfill onsite if it meets the "no detectable" criteria or passes a final status survey.

3.2.2.2.18 Miscellaneous Storage Tanks

The following tanks are contaminated, potentially contaminated, or are needed to support decommissioning activities:

- Primary Water Storage Tank,
- Temporary Waste Water Processing Island Tanks,
- Service Building Radioactive Sump Tanks,
- Propane Tanks,
- Fire Water Storage Tank,
- Fuel Oil Storage Tanks,

These tanks will remain in service, as required, throughout the dismantlement phase. When no longer required, the tanks will be emptied, cleaned and disposed of by an authorized and licensed contractor. The tanks will be removed to the top of the concrete foundations. There are currently no additional decontamination or dismantlement considerations specific to the miscellaneous storage tanks. Tanks that contained radioactive materials will be disposed of as radioactive waste.

3.2.2.2.19 Meteorological Tower

The Meteorological Tower provided real time capability to determine wind speed and direction for onsite emergency planning purposes. The Meteorological Tower will be removed to grade.

A meteorological tower exists at the ISFSI pad to provide real time capability to determine wind speed and direction for on-site emergency planning purposes. There are currently no decontamination or dismantlement considerations specific to the Meteorological Tower.

3.2.3 Phase 3 Activities

The final phase of decommissioning will take place after all spent fuel and GTCC waste is removed from the site and the dismantlement and decontamination of the ISFSI is complete. In the interim, spent fuel and GTCC will be stored in the ISFSI.

Decommissioning of the ISFSI consists primarily of the disposal of the concrete canister overpacks, provided they are not shipped with the spent fuel casks. The overpack design

3.4.1 Occupational Exposure

The total radiation exposure impact for decommissioning was estimated in the Decommissioning Plan, Reference 3-5, to be approximately 744 person-rem (see breakdown in Table 3-3). This estimate was re-evaluated in 1996, resulting in a lower value of 580 person-rem (see also Table 3-3). As discussed in the PSDAR, the actual exposure through December 31, 2002, is 555 person-rem.

Radiation exposure to off-site individuals for expected conditions, or from postulated accidents is bounded by the EPA's Protective Action Guidelines and NRC regulation. The public exposure due to radiological effluents will continue to remain well below the 10CFR20 limits and the ALARA dose objectives of 10CFR50, Appendix I. This conclusion is supported by the YNPS Annual Effluent Release Reports in which individual doses to members of the public are calculated for station liquid and gaseous effluents.

3.4.2 Radioactive Waste Projections

No significant impacts are expected from the disposal of low-level radioactive waste (LLW). The total volume of YNPS LLW for disposal was estimated in the Decommissioning Plan, Reference 3-5, to be approximately 132,000 ft^3. As of the end of 2002, over 144,184 ft^3 was shipped. The previous estimate has been subsequently re-evaluated to reflect the current scope of work, and the "to go" volume for disposal is estimated to be 480,512 ft^3 (Reference 3-7). A final estimate for waste volume will be developed based upon the results of further characterization and waste optimization techniques. The waste volume estimated to be generated by the YNPS decommissioning remains bounded by the FGEIS estimate for a reference PWR of 647,000 ft^3.

3.5 References

3-1 Title 10 to the Code of Federal Regulations, Part 50.82, "Termination of license."

3-2 YNPS Post-Shutdown Decommissioning Activities Report, dated June 2003.

3-3 Supplement 1 to NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated November 2002.

3-4 Letter from R.W. Varney, Region Administrator, EPA Region I, to J. Kay, Regulatory Affairs, Yankee, Extension of Amended (as of January 6, 1999) Alternative Method of Disposal Authorization for PCB Paint Removal, dated October 8, 2002.

3-5 YNPS Decommissioning Environmental Report, dated December 1993.

3-6 USNRC Atomic Safety and Licensing Board Docket No. 50-029-DCOM, Supplemental Affidavit of Russell A. Mellor, September 3, 1996.

3-7 Memorandum RP-03-045 from Greg Babineau to Jim Kay, dated November 19, 2003.

4 SITE REMEDIATION PLANS

4.1 Introduction

In accordance with 10CFR50.82 (a)(9)(ii)(C) (Reference 4-1), the LTP must provide the "plans for site remediation." These plans must include the provisions to meet the criteria from Subpart E of 10CFR20 (Reference 4-2) before the site may be released for unrestricted use:

- Annual total effective dose equivalent to the average member of the critical group not to exceed 25 mrem, and
- The dose to the public must be "as low as reasonably achievable," or ALARA.

Decontamination and dismantlement (D&D) activities are being conducted in accordance with the YNPS Radiation Protection, Safety and Waste Management Programs, which are well established and frequently inspected. Changes made to the programs for D&D activities are documented and processed in accordance with existing plant administrative procedures and 10CFR50.59, as appropriate.

This section describes the methodologies and criteria that will be used to perform activities to remove residual radioactivity and to demonstrate compliance with the ALARA criterion, required by 10CFR20. More specific detail regarding remediation activities may be found in Section 3.

4.2 Remediation Actions

Remediation actions may be required to reduce the radioactivity levels below the applicable cleanup criteria as provided in Sections 5 and 6. The specific remedial actions depend on the type of area under consideration. These area types are categorized as one of the following:

- Soils/sediment
- Structures (including building interiors and exteriors, major freestanding exterior structures, exterior surfaces of plant systems, and paved exterior ground surfaces)
- Groundwater and surface water

Potential remediation activities for each category are described below. Specific decommissioning and remediation activities will be performed in accordance with applicable site procedures. Post-remediation surveys will be used to confirm that the remediation target is achieved.

The selection of appropriate instrumentation for post-remediation surveys is important from a planning and financial risk management perspective. In some cases small handheld beta-gamma detectors may be used to determine if remedial actions have been successful; their use depends

upon the radionuclides present in the survey unit, the DCGL for that radionuclide and the MDC of the detector. In other cases, the actual final status survey instrumentation may be used to evaluate remedial actions.

4.2.1 Soils

Soils not meeting the criteria for license termination will be removed and disposed of as radioactive waste. Offsite fill may be used to replace the excavated materials. As discussed previously in Section 2, the site characterization process establishes the location, depth and extent of soil contamination. As needed, additional investigations will be performed to ensure that any soil contamination profiles that may change during the remediation actions are adequately identified and characterized. In cases where offsite fill is used to replace the excavated materials, a radiation survey of the material will be conducted. This will be done as a documented survey to ensure that the background radiation levels (from the presence of naturally occurring radioactive material) from this fill material is not significantly higher than that from the onsite material. Based upon the results of this survey, either background radiation levels will be accounted for in subsequent final status surveys or the material will be rejected for use.

Excavations will be surveyed (either to FSS criterion, as discussed in Section 5, or to the "no detectable radioactivity" criteria) following soil removal for radiological remediation. The NRC will be notified, through routine communications, of YAEC's intent to backfill excavations. RAI #42

4.2.2 Structures

Remaining concrete from structures will be remediated, as necessary, to a level meeting the radiological criteria for unrestricted release of the site, as discussed in Section 6, or to the "no detectable radioactivity" criteria. Methods for remediating structures may include a variety of techniques, and a number of factors determine the choice of the remediation method for a given area. These include: the size of the contaminated area, the extent of contamination, surface material, depth of contamination, and accessibility.

Remediation activities for an area may include wiping, vacuuming, and washing with low- or high-pressure applications. Surfaces may also be remediated using surface removal techniques such as scabbling or grinding. Use of surface removal techniques controls the removal depth, minimizing the waste volume produced.

For concrete surfaces, remediation methods may include core drilling, concrete sawing, or scabbling. Scabbling removes the concrete surface using roto-peen devices, flappers, or similar devices and is effective for removing contamination that resides close to the surface. Abrasive blasting may also be used as an effective technique for contamination removal from surfaces that are not necessarily smooth. Also, chipping, jack-hammering, and other similar aggressive methods may be needed for removal of concrete surfaces as deep as the first mat of reinforcing steel. Contamination control barriers will be used as appropriate during activities, such as these, that may result in airborne contamination. Strippable coatings can be used to remove contaminants from surfaces where more aggressive methods may not be appropriate or when other techniques are not successful.

represent the level, expressed as a percentage or fraction of the DCGL, for which the benefit of further clean-up of structures is greater than the associated costs.

As discussed in Section 3, some structural elements and embedded or buried piping and conduit will remain that have been surveyed to ensure that no detectable radioactivity is present. Per NUREG-1757, Volume 2, Appendix N, material may be left onsite without performing an ALARA evaluation, if it contains no residual radioactivity distinguishable from background. Accordingly, no ALARA analysis will be applied to structures or equipment that have been surveyed and found to have no detectable radioactivity present.

Upon completion of post-remediation surveys and satisfaction of the 25 mrem/yr TEDE criteria, the level of residual radioactivity in the survey area will be compared against the appropriate generic ALARA screening level (soil or building surface). Where the level of residual radioactivity is lower than the generic ALARA screening level, the residual radioactivity is clearly ALARA, no further action is required, and final status surveys can proceed. Where the level of residual radioactivity is greater than the generic ALARA screening level, one of two actions will be taken: (1) a survey-unit ALARA evaluation may be performed to determine the unit-specific ALARA action level for comparison with level of residual radioactivity, or (2) additional clean-up can be performed without further ALARA analyses.

4.3.2 Survey Unit-Specific ALARA Evaluations

In cases where levels of residual radioactivity are above the generic ALARA screening levels described above, YAEC may adopt the option to perform survey unit-specific ALARA evaluations using approved site procedures. These survey unit-specific ALARA evaluations will be performed using survey unit-specific data from post-remediation surveys in accordance with Appendix N to NUREG-1757, Volume 2, and will take into account:

- Radiation doses and environmental impacts for the decommissioning process and from the residual radioactivity remaining onsite following the decommissioning, and
- Other costs and risks associated with the decontamination and decommissioning of the site.

Once the total cost, $Cost_T$, for a survey-unit specific clean-up activity has been calculated, a remediation level, expressed as a fraction of a DCGL, can be determined and the ALARA evaluation can be performed using the process described in NUREG-1757, Volume 2.

The action levels represent the radioactivity concentrations at which a clean-up action is cost beneficial. The ALARA criterion is met by demonstrating that the residual radioactivity is already below the action level or by performing the action. An ALARA analysis ensures that the efforts to remove residual contamination are commensurate with the risk associated with leaving the residual contamination in place. However, the residual contamination must be low enough to assure the annual dose to the average member of the critical group does not exceed 25 mrem/yr TEDE. RAI #44

Table 4A-1

Parameter Values for Use in ALARA Analyses

Parameter	Acceptable Value	
	Building	Land
PD	0.09 person/m^2	0.0004 person/m^2
r	0.07 per year	0.03 per year
N	70 years	1000 years

RAI #45

The development of values for the equation parameters of total Cost ($Cost_T$), and removable fraction for remediation action being evaluated, F, are described in Sections 4.A.1.1 and 4.A.1.2. Where values other than those in the table above or in Section 4.2.3 are used, justification is provided.

4.A.1.1 Calculation of Total Cost

Calculations of total cost generally include the monetary costs of:

- The clean-up action being evaluated ($Cost_R$)
- Transportation and disposal of wastes generated ($Cost_{WD}$)
- Workplace accidents that occur because of the clean-up action ($Cost_{ACC}$)
- Traffic fatalities resulting from transporting the waste generated by the action ($Cost_{TF}$)
- Doses received by workers performing the clean-up action ($Cost_{WDose}$)
- Doses to the public from excavation, transportation, and disposal of the waste ($Cost_{PDose}$)

Thus,

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} \quad \text{(Equation B-2)}$$

Other monetary costs may be included as appropriate for the specific situation.

The cost of waste transport and disposal, $Cost_{WD}$, is calculated using the following equation:

$$Cost_{WD} = V_A \times Cost_V \quad \text{(Equation 4A-3)}$$

Where

V_A = volume of waste produced, m^3
$Cost_V$ = cost of waste disposal, \$/m^3

4.A.1.2 Determination of Clean-up Action Effectiveness

The clean-up action effectiveness, F, is the fraction of the residual radioactivity removed by the clean-up action. It is determined by collecting and analyzing pre- and post-clean-up measurements in the area in which the clean-up action is performed. A sufficient number of measurements are made to establish a consistent value.

4A.2 ALARA Evaluation

When dismantlement actions are completed, residual radioactivity may remain. 10CFR20.1402 requires assurance that residual radioactivity has been reduced to levels that are ALARA. For evaluations prior to additional clean-up actions, the ALARA analysis for data evaluation will be performed using data from operational Radiation Protection surveys in accordance with NUREG-1757 and will take into account:

- Radiation doses and environmental impacts for the decommissioning process and from the residual radiation remaining on site after the completion of decommissioning.
- Other costs and risks associated with the decontamination and decommissioning of the site.

Once the total cost, $Cost_T$, for a clean-up action has been calculated, an ALARA action level, expressed as a fraction of a $DCGL_W$, can be determined and the ALARA evaluation can be performed using the previously presented equations.

As discussed above this evaluation determines the point at which clean-up is cost beneficial and then compares existing residual radioactivity levels to that ALARA action level. When the residual radioactivity is in excess of the calculated ALARA action level, additional clean-up action is considered to be cost beneficial and should be taken. If residual activity is below the ALARA action level, the ALARA criterion is considered to be met already and no additional remedial action is required to be performed.

ALARA evaluations will be performed when justification is needed for not performing additional clean-up in an area. This is consistent with the recommendations provided in NUREG-1757. As appropriate, the final status survey report will appropriately document that all concentrations in the survey unit are below the ALARA action level. As previously discussed, if the decision to perform a given clean-up action has been made, then the activity does not require an ALARA justification.

As previously noted, the ALARA criteria is met by demonstrating that the residual radioactivity is already below the action level or by performing the clean-up action. An ALARA analysis ensures that the efforts to remove residual contamination are commensurate with the risk that exists with leaving the residual contamination in place. RAI #46

5 FINAL STATUS SURVEY PLAN

5.1 Introduction

The Final Status Survey (FSS) Plan describes the methods to be used in planning, designing, conducting, and evaluating final status surveys at the YNPS site. These surveys serve as key elements to demonstrate that the dose from residual radioactivity is less than the maximum annual dose criterion for license termination for unrestricted use specified in 10CFR20.1402 (Reference 5-1). The additional requirement of 10CFR20.1402, that residual radioactivity at the site be reduced to levels that are as low as reasonably achievable (ALARA), is addressed in Section 4. The Final Status Survey Plan was developed using the guidance of NUREG-1575, "The Multi-Agency Radiological Site Survey and Investigation Manual (MARSSIM)" (Reference 5-2); Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 5-3); NUREG-1727, "NMSS Decommissioning Standard Review Plan," (Reference 5-4); and NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance," (Reference 5-5).

The FSS process described in the survey plan adheres to the guidance of MARSSIM. However, advanced survey technologies may be used to conduct radiological surveys that can scan the surface and record the results. This survey plan allows for the use of these advanced technologies, where survey quality and efficiency can be increased, as long as the survey results are at least equivalent, in terms of their statistical significance, to those that would have been obtained using the non-parametric sampling methods of MARSSIM. In cases where advanced survey technologies are to be used, a technical evaluation will be developed to describe the technology to be used and to demonstrate how the technology meets the objectives of the survey. These technical evaluations will be referenced, as appropriate, in Final Status Survey Reports and will be available for NRC review. Notification will be made to the NRC prior to the use of advanced instruments or technologies.

RAI#1

RAI#2, 19, 27, 28, 29

5.2 Scope

The FSS Plan encompasses the radiological assessment of impacted structures, systems and land areas for meeting the dose rate criterion for unrestricted release specified in 10CFR20.1402. In addition, Section 5.6.3.2.4 addresses the plan for the assessment of groundwater.

for collecting the data, practical constraints and the scale of decision making. For the FSS, the target population is the set of samples or direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest (e.g., soil, water, concrete) is specified during the planning process. The spatial boundaries include the entire area of interest including soil depth, area dimensions, contained water bodies and natural boundaries, as needed. Temporal boundaries include those activities impacted by time-related events including weather conditions, seasons (i.e., more daylight available in the summer), operation of equipment under different environmental conditions, resource loading and work schedule.

- **Develop a Decision Rule**

  This step of the DQO process develops the binary statement that defines a logical process for choosing among alternative actions. The decision rule is a clear statement using the "If...then..." format and includes action level conditions and the statistical parameter of interest (e.g., mean of data). Decision statements can become complex depending on the objectives of the survey and the radiological character of the affected area.

- **Specify Tolerable Limits on Decision Errors**

  This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided for rejection.

  The primary consideration during FSS will be demonstrating compliance with the release criteria. The following statement will be used as the null hypothesis at YNPS: "The survey unit exceeds the release criteria." RAI#3

  Decision errors occur when the data set leads the decision-maker to make false rejections or false acceptances during hypothesis testing. The α error (Type I error) is set at 0.05 (5%), and a nominal value of 0.05 (5%) has been established for the β error (Type II error). Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors is considered acceptable. The upper bound corresponds to the release criteria. The Lower Bound of the Gray Region (LBGR) is determined in this step of the DQO process. LBGR is influenced by a parameter known as the relative shift. The relative shift is set between (and including) 1 and 3. If the relative shift is not between (or including) 1 and 3, then the LBGR is adjusted.

  Graphing the probability that a survey unit does not meet the release criteria may be used during FSS. This graph, known as a power curve, may be performed retrospectively (i.e., after FSS) using actual measurement data. This retrospective power curve may be

important when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria) to demonstrate that the DQOs have been met.

- **Optimize the Design for Obtaining Data**

  The first six steps are the DQOs that develop the performance goals of the survey. This final step in the DQO process leads to the development of an adequate survey design.

5.4.2 Classification of Survey Areas and Units

The adequacy of the final status survey process rests upon partitioning the site into properly classified survey units of appropriate physical area. Section 2 of the LTP discusses in detail the HSA for the YNPS site and the classifications assigned to all of the site structures and grounds. Characterization is an ongoing effort throughout the decommissioning process, and survey unit classifications may be modified on the basis of new characterization information or impacts from decommissioning activities. The process described in LTP Section 1.6 will be used to evaluate the modifications to unit classifications to determine whether prior notification to the NRC is required. Survey areas have been determined as described in Section 2.1.1 of this LTP. RAI#5

A survey area may consist of one or more survey units. A survey unit is a physical area consisting of structures or land areas of a specified size and shape which will be subject to a final status survey. Compliance with the applicable criteria will be demonstrated for each survey unit.

Survey units are limited in size based on classification, exposure pathway modeling assumptions, and site-specific conditions. The surface area limits, used in establishing the initial set of survey units for the YNPS Final Status Survey Plan, are provided in Table 5-1 for structures and land areas. The area limits for structures refer to floor area, and not the total surface area; which would include the walls and ceiling. This is consistent with the guidance in Table A.1 of Appendix A to NUREG-1757) and MARSSIM. The floor area limits given in Table 5-1 were also used to establish survey unit sizes for structures such as roofs or exterior walls of buildings. The limits given in Table 5-1 will also be used should the need arise to establish any new survey units beyond the initial set given in this plan.

As indicated in LTP Section 2, areas of YNPS that are classified as impacted have been divided into survey units to facilitate survey design. Each survey unit has been assigned an initial classification based on the site characterization process and the historical site assessment.

Table 5-1
YNPS Survey Unit Surface Area Limits

Survey Unit Classification	Surface Area Limit
Class 1:	
Structures (floor area)	≤ 100 m^2
Land areas	≤ 2,000 m^2
Class 2:	
Structures (floor area)	100 m^2 < area ≤ 1,000 m^2
Land areas	2,000 m^2 < area ≤ 10,000 m^2
Class 3:	
Structures (floor area)	no limit
Land areas	no limit

A survey unit can have only one classification. Thus, situations may arise where it is necessary to create new survey units by subdividing areas within an existing unit. For example, residual radioactivity may be found within a Class 3 survey unit, or residual radioactivity in excess of the $DCGL_W$ may be found in a Class 2 unit. In such cases, it may be appropriate to define a new survey unit within the original unit that has a lower (more restrictive) classification. Alternately, the classification of the entire unit can be made more restrictive. The NRC will be notified at least 14 days prior to subdividing and/or reclassifying a survey area. RAI#6, #24

5.4.3 Reference Coordinate Systems

Measurements and sample locations can be identified in one of two ways: using a benchmark location or a global positioning system (GPS). If benchmark is used, that benchmark (origin) will be provided on the map or plot included in the final status survey package. The GPS to be used at YNPS site has sub-meter accuracy. Sub-meter accuracy is sufficient to establish a reproducible reference coordinate system and to physically locate sample points determined by the final status survey plant for an area. A benchmark is being established for daily pre-operational checks of the systems. RAI#7, #20

Any coordinate systems used for surveys will typically take the form of a grid of intersecting, perpendicular lines; but other patterns (e.g., triangular and polar) may be used as convenient. Physical gridding of a survey unit will only be done in cases where it is beneficial and cost effective to do so. When physical gridding is used, benchmark locations will be designated by either marking a spot with surveyor's paint (or equivalent) for indoor areas or setting an iron pin (or equivalent) for outdoor areas. If needed, grid lines or measurement locations will be marked (e.g., with chalk lines, paint, surveyor's flags), as appropriate. Global positioning systems may also be used as practical.

5.4.4 Reference Areas and Materials

The DQO process will be used during the planning phase in the preparation of a final status survey plan to determine whether media specific backgrounds, ambient area background or no background will be applied to a survey area or unit. The approach used for a specific survey unit will be based on the survey unit classification and the DCGLs.

If applied, media specific backgrounds will be determined via measurements made in one or more reference areas and on various materials selected to represent the baseline radiological conditions for the site. The determination of media specific background will be controlled with a documented survey plan, which will include the DQO process. These data will be evaluated in a technical support document and available for inspection by the NRC. This process will ensure that the collected data will meet the needs of the final status survey. The collected data may be used as the reference area data set when using the Wilcoxon Rank Sum test, or, for survey units with multiple materials, background data may be subtracted from survey unit measurements (using paired observations) if the Sign Test is applied.

Depending on the values of the DCGLs, an alternative method to using material specific backgrounds may be used during final status surveys. This alternative method will involve the determination of the ambient area background in the survey unit and will only be applicable to beta-gamma detecting instruments. This determination will be made prior to performing a final status survey at a location within a survey area that is of sufficient distance (or attenuation) from the surfaces to eliminate beta particles originating from the surfaces from reaching the detector. At such a location, the ambient background radiation will be due only to ambient gamma radiation and will be a background component of surface measurements. The average background determined at this location can be used as a conservative estimate since it is expected to be less than the material specific background for the material in the room because it does not fully account for the naturally occurring radioactivity in the materials. Using this lower ambient background will result in conservative calculated residual radioactivity levels. If the average background reading exceeds a predetermined value, the survey would be terminated and an investigation performed to determine and eliminate the reason for the elevated reading. Each of the survey unit readings would subtract this average background value and the Sign Test applied. If this alternative method is to be used, the NRC will be notified of YAEC's intent at least 14 days prior to implementation. RAI #18

Whether or not they are radionuclide-specific, background measurements should account for both spatial variability over the area being assessed and the precision of the instrument or method being used to make the measurements. Thus, the same materials or areas may require more than one background assessment to provide the requisite background information for the various survey instruments or methods expected to be used for final status surveys. The result of these background assessments will provide the basis for determining the mean and its associated standard deviation.

The presence of the spent fuel stored at the Independent Spent Fuel Storage Installation (ISFSI) will increase gamma radiation levels at close distances to the storage pad. The specific region where this elevated gamma radiation will influence the final status surveys has not been precisely

5.4.5.2 Area Surveillance Following Final Status Surveys

Isolation and control measures will be implemented through approved plant procedures and will remain in force throughout final survey activities and until there is minimal risk of recontamination from decommissioning or the survey area has been released from the license. In the event that isolation and control measures established for a given survey unit are compromised, evaluations will be performed and documented to confirm that no radioactive material was introduced into the area that would affect the results of the Final Status Survey.

To provide additional assurance that land areas and structures that have undergone successful final status surveys (FSS) remain unchanged until final site release, these areas will be surveyed periodically. The strategy for performing these surveys depends on the following:

- the type of area (land or building),
- the area classification of the survey areas as well as that of the adjacent survey areas,
- the potential for re-contamination of the area from remediation activities in adjacent areas,
- the proximity to operational events involving radioactive contamination.

For FSS areas adjacent to areas where either remediation activities (as required to meet the site release criteria) or operational events may have impacted the FSS area, a re-survey of the FSS area will be conducted. This re-survey will involve judgmental sampling of boundary and/or potential access points to the FSS area. If the results of the re-surveys indicate that any measurement (DCGL fraction for land areas and bulk materials and static measurement for surfaces) is statistically greater than the initial FSS results (that is, measurement is > 2 standard deviations from the initial FSS mean), then an investigation survey will be conducted of the area. RAI#9 The investigation survey will include a larger physical area than the re-survey. If the results of the investigation survey are statistically different than the FSS survey results, then a full FSS survey of the affected units will be performed in accordance with the LTP. The results of re-surveys and investigation surveys will be documented and maintained in the FSS files for the affected survey units. Additionally, for any area that has completed FSS activities, any soil, sediment, or equipment relocated to that area will require demonstration that the material introduced does not result in residual radioactivity that is statistically different than that in the FSS.

Periodic surveys will be performed on a random sample basis for 5% of those survey areas for which FSS activities have been completed. If the results of these surveys exceed specific radiological contamination levels (i.e., measurements > 2 standard deviations from the initial FSS mean), an investigation survey will be conducted. This investigation survey will be more RAI#10 extensive than the scope of the routine survey to define the magnitude and extent of the contamination. If the results of the investigation survey indicate contamination that is statistically different than the FSS survey results (as described above), then full FSS of the affected survey areas will be performed in accordance with the LTP. The results of re-surveys and investigation surveys will be documented and maintained in the FSS files for the affected

5.4.6.1 Gross Activity DCGLs

For alpha or beta surface activity measurements, field measurements will typically consist of gross activity assessments rather than radionuclide-specific techniques. Gross activity DCGLs will be established, based on the representative radionuclide mix, as follows:

$$DCGL_{GA} = \frac{1}{\sum_{1}^{n} \frac{f_i}{DCGL_i}} \qquad \text{(Equation 5-1)}$$

where

f_i = fraction of the total activity contributed by radionuclide i
n = the number of radionuclides
$DCGL_i$ = DCGL for measurable radionuclide i

Gross activity DCGLs can be developed for gross beta measurements, or a gross beta DCGL can be scaled so that it acts as a surrogate for gross alpha (see Section 5.4.6.2). RAI#11

5.4.6.2 Surrogate Ratio DCGLs

In order to address the potential for contamination with difficult-to-detect radionuclides for gross surface contamination measurements, one of two processes will be employed: (1) the use of a surrogate relationship to contamination or (2) direct measurement of alpha contamination. It is acceptable industry practice to assay a hard-to-detect (HTD) radionuclide by using an easy-to-detect (ETD) radionuclide as a surrogate. A common example would be to use a beta measurement to assay for a hard-to-detect alpha emitting radionuclide. Another example would be to relate a specific radionuclide, such as Cesium-137, to one or more radionuclides of similar characteristics. In such cases, to demonstrate compliance with the release criteria for the survey unit, the DCGL for the surrogate radionuclide or mix of radionuclides must be scaled to account for the fact that it is being used as an indicator for an additional radionuclide or mix of radionuclides. The result is referred to as the surrogate DCGL. RAI#12

The following process will be applied to assess the need to use surrogate ratios for final status surveys (FSS).

- Determine whether HTD radionuclides (e.g., TRU, Sr-90, H-3) are likely to be present in the survey unit based on process knowledge and historical data or characterization.

- When HTD radionuclides are likely to be present, establish a relationship using a representative number of samples (typically six or more). The samples may come from another survey unit if the source of the contamination and expected concentrations are reasonably the same. These samples will be analyzed for ETD and HTD radionuclides

using gross alpha, alpha spectroscopy, gross beta analysis, or gamma spectroscopy techniques.

Surrogate relationships will be determined using one of methods described below.

- Develop a surrogate relationship for each HTD radionuclide.

$$DCGL_{surrogate} = DCGL_{ETD} \times \frac{DCGL_{HTD}}{(f_{HTD:ETD} \times DCGL_{ETD}) + DCGL_{HTD}} \quad \textit{(Equation 5-2)}$$ RAI#12

- Determine the average surrogate DCGL and the standard deviation from the surrogate relationships.

If the %CV (coefficient of variation) of the average surrogate DCGL is within 25% then the average surrogate DCGL will be applied to the survey area. The %CV is the percent ratio of the standard deviation to the average surrogate DCGL. If this criterion is not met, the following steps will be applied.

- After a more detailed spatial analysis of the radionuclide mix distribution, the unit may be subdivided into separate survey units.
- The lowest surrogate DCGL from the observed radionuclide mix may be applied to the entire survey unit.
- A DCGL, specific to the survey unit, may be used. This DCGL would be determined by collecting and analyzing additional samples and documenting the evaluation of the resulting radionuclide distribution.

- The surrogate DCGL may be computed from a simple recurrence formula :

$$\frac{C_{ETD}}{DCGL_{Surrogate}} = \frac{C_{ETD}}{DCGL_{ETD}} + \frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \ldots + \frac{C_i}{DCGL_i} \quad \text{(Equation 5-3)}$$

or, for simplification

$$\frac{C_E}{D_{Surrogate}} = \frac{C_E}{D_E} + \frac{C_1}{D_1} + \frac{C_2}{D_2} + \ldots + \frac{C_i}{D_i} \quad \text{(Equation 5-4)}$$

where:

D_E	=	the DCGL for the easy-to-detect radionuclide
D_1	=	the DCGL for the first hard-to-detect radionuclide
D_2	=	the DCGL for the second hard-to-detect radionuclide

D_i = the DCGL for the i[th] hard-to-detect radionuclide
f1 = the activity ratio of the first hard-to-detect radionuclide to the easy-to-detect radionuclide
f2 = the activity ratio of the second hard-to-detect radionuclide to the easy-to-detect radionuclide
fi = the activity ratio of the ith hard-to-detect radionuclide to the easy-to-detect radionuclide RAI#12, 13

Consider the case of three HTD radionuclides from which a surrogate will be calculated.

$$DCGL_{Surrogate} = \frac{(D_E D_1 D_2 D_3)}{(D_1 D_2 D_3)+(f_1 D_E D_2 D_3)+(f_2 D_E D_1 D_3)+(f_3 D_E D_1 D_2)} \quad \text{(Equation 5-5)}$$

A general expression for the surrogate equation based on recursive relationships is provided by Equation 5-6 for n HTD radionuclides.

$$DCGL_{Surrogate} = \frac{1}{1/D_E + \sum_{i=1}^{n} f_i / D_i} \quad \text{(Equation 5-6)}$$

5.4.6.3 Elevated Measurement Comparison (EMC) DCGLs

The DCGL established for the average residual contamination in a survey unit is $DCGL_W$. Values of the $DCGL_W$ may be scaled through the use of area factors to obtain a DCGL that represents the same dose to an individual from residual contamination over a smaller area within a survey unit. Such a value is called $DCGL_{EMC}$, where the subscript EMC stands for elevated measurement comparison. The $DCGL_{EMC}$ is defined as the product of the applicable $DCGL_W$ and a correction factor know as the area factor.

The area factor is equal to the ratio of the dose from the base-case contaminated area to the dose from a smaller contaminated area with the same radioactive source concentration. Area factors are required for both the resident farmer and the building occupancy scenarios. Area factors for both the resident farmer and building occupancy scenarios are being calculated for the radionuclides of concern at the YNPS site considering all applicable potential pathways of exposure.

For the resident farmer scenario, RESRAD (Version 6.21) is being used to determine area factors. For the building occupancy scenario, RESRAD-BUILD (Version 3.21) is being used to determine area factors. Area factors are not being computed for areas smaller than 1 m^2 for either the resident farmer or the building occupancy scenarios. Area factors are being provided in an appendix to Section 6 of the LTP.

- For Class 1 survey units, 100% of the surface will be scanned;
- For Class 2 survey units, between 10% and 100% of the surface will be scanned in a combination of systematic and judgmental measurements for outdoor units and for floor and lower walls of structures; and 10% to 50% of the surface will be covered for upper walls and ceilings;
- Scanning will be done on a judgmental basis for Class 3 survey units.

The considerations used in determining the scanning coverage to be applied to survey unit/area include: RAI #14

- the potential for suspect areas based upon historical information and walkdown,
- the potential for residual radioactivity relative to the DCGL, and
- any other indication of the potential for elevated activity below the DCGL.

Though the emphasis of the document is on conducting final status surveys through a combination of fixed measurements and scans, MARSSIM also allows for use of advanced survey technologies as long as these techniques meet the applicable requirements for data quality and quantity. "Advanced technologies" in this context refers to survey techniques where the instrument is capable of recording data as an area is surveyed and the measurement sensitivity is an acceptable fraction of the applicable $DCGL_W$ (see Section 5.6.1.3). Such methods are desirable for final status surveys since they allow survey units to be assessed with a single measurement rather than separate fixed measurements and scans.

Advanced survey techniques may be used alone or in combination with fixed measurements and scans to assess a survey unit. For Class 1 and Class 2 units, two conditions must be met for advanced technologies to be employed as the only survey technique: an acceptable fraction of the survey unit surface area must be scanned; and the minimum detectable concentration (MDC) for the measurements must be an acceptable fraction of the $DCGL_W$. For Class 1 units, 100% of the area must be covered. For Class 2 units, the coverage requirements for advanced technologies to be used alone are from 50% to 100% of the area for outdoor survey units or for floors and lower walls; and from 10% to 50% of the area for upper walls and ceilings. In cases where these coverage requirements cannot be achieved by an advanced survey technology or where the MDC is too large relative to the applicable $DCGL_W$ (see below), the survey will be augmented with fixed measurements and traditional scans as necessary in accordance with Section 5.5.1 and subsequent subsections of this plan. Advanced technologies may be used for judgmental assessments in Class 3 areas as long as the following MDC requirements are met.

The number of scan areas will be greater than 15, which corresponds to the minimum number of samples for α=0.05 and β=0.05. The location of the scan area will be determined by using the guidance in Section 5.5.1.6. The size of the scan area will be determined by the size of the survey area, the percent survey coverage, and the number of scan areas. RAI #15, #16

For fixed measurements, MARSSIM states that MDCs should be as far below the $DCGL_W$ as possible, with values less than 10% of the $DCGL_W$ being preferred, and up to 50% of the

result in excessive costs in that survey units that meet the release criterion could be subjected to superfluous remediation efforts. Under the current regulatory models, an α value that is too large equates to greater risk to the public in that there is a greater chance of releasing a survey unit that does not meet the release criterion.

Section A.7.2 of Appendix A to NUREG-1757 recommends that the α decision error rate be set to 0.05 (5%) and that "any value of β is acceptable to the NRC." Thus, decision error rates for final status surveys designed for the YNPS site will be set as follows:

- the α value will always be set to 0.05 unless prior NRC approval is granted for using a less restrictive value;

- the β value is nominally set to 0.05, but may be changed if it is found that more fixed measurements than necessary are being made to demonstrate compliance with the release criterion.

5.5.1.2 Determining the Relative Shift

Another input to the process of selecting the required number of measurements that is somewhat independent of the statistical test to be employed is the determination of what is called the relative shift. The relative shift is a parameter that quantifies the concentrations to be measured in a survey unit relative to the variability in these measurements. The relative shift is a function of the $DCGL_W$, a parameter called the "lower bound of the gray region" (LBGR), and either the expected standard deviation of the measurements to be made in the survey unit (σ_s) or the standard deviation established for the corresponding reference area (σ_r). The choice of σ_s or σ_r depends on whether the survey data are to be evaluated against a reference area(s). Reference areas are used if the WRS test is applied or, where gross measurements are to be background subtracted, the Sign test may be used. The σ_s values will be selected by:

- using existing characterization or remediation support survey data or
- making preliminary measurements.

Values of σ_r will be computed using data collected from measurements in reference areas or from reference materials (typically outside of the survey area or unit), as appropriate. RAI #18

Given that σ_s and σ_r values should reflect a combination of the spatial variability in the concentration and the precision in the method of measurement, these values will be selected based on existing survey data only when the existing measurements were made using techniques equivalent to those to be used during the final status survey.

The LBGR represents the concentration to which the survey unit must be decontaminated in order to have an acceptable probability of passing the statistical test. The difference between the $DCGL_W$ and the LBGR, known as the shift, can be thought of as a measure of the resolution of the measurements that will be made in a survey unit. The shift is denoted as Δ.

The relative shift (Δ/σ) is computed as the quotient of the shift and the appropriate standard deviation values. If no reference area data are needed to evaluate the survey results, the expected standard deviation of the measurements (σ_s) is used. If a reference area is required, the larger of the values of σ_s or σ_r is used.

To compute the relative shift, the appropriate sigma value and an initial LBGR are selected. The initial value for LBGR will be based upon site specific information, if available; otherwise, per MARSSIM, and Section A.7.1 of Appendix A to NUREG-1757, the initial value for the LBGR will be set to one-half of the $DCGL_W$. If the resulting relative shift is not in the range of 1.0 and 3.0, the LBGR is adjusted until it is. If the relative shift is too low, the LBGR is decreased; and if the relative shift is too high, the LBGR is increased.

5.5.1.3 Selecting the Required Number of Measurements for the WRS Test

The minimum number of fixed measurements required when the WRS is computed by the following equation:

$$N = \frac{1}{2} \times \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{3(P_r - 0.5)^2} \qquad \text{(Equation 5-7)}$$

where

N = the minimum number of measurements required for each survey area or reference area;

$Z_{1-\alpha}$ = the percentile represented by the α decision error;

$Z_{1-\beta}$ = the percentile represented by the β decision error; and

P_r = the probability that a random measurement from the survey unit exceeds a random measurement from the reference area by less than the $DCGL_W$ when the survey unit median is equal to the LBGR concentration above background.

Values of P_r, $Z_{1-\alpha}$ and $Z_{1-\beta}$ will be taken from Tables 5.1 and 5.2 of MARSSIM. P_r is a function of the relative shift, and $Z_{1-\alpha}$ and $Z_{1-\beta}$ depend on the selected values for α and β.

The value of N computed for the WRS test applies for both the survey unit and the reference area (i.e., at least N measurements should be performed in both areas). To ensure against lost or unusable data, the value of N will be increased by at least a factor of 1.2 when assigning the number of measurements to be made.

5.5.1.4 Selecting the Required Number of Measurements for the Sign Test

The minimum number of fixed measurements required when the Sign test is computed by the following equation:

$$N = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(Sign\ p - 0.5)^2}$$ (Equation 5-8)

where

N = the minimum number of measurements required;
$Z_{1-\alpha}$ = the percentile represented by the α decision error;
$Z_{1-\beta}$ = the percentile represented by the β decision error; and
Sign p = the probability that a random measurement from the survey unit will be less than the $DCGL_W$ when the survey unit median concentration is equal to the LBGR.

Values for Sign p will be taken from Table 5-4 of MARSSIM.

To ensure against lost or unusable data, the number of date points will be will be increased by 20%, and rounded up, over the value, N, calculated in Equation 5-7 and 5-8.

5.5.1.5 Assessing the Need for Additional Measurements in Class 1 Survey Units

Given the potential for small areas of elevated activity in Class 1 survey units, evaluations must be performed to assess the potential for missing such areas while scanning in locations not covered by fixed measurements. This evaluation, referred to as the Elevated Measurement Comparison (EMC), is performed by comparing the MDC of the scanning technique to the $DCGL_{EMC}$ for the survey unit of interest. If the scanning MDC is larger than the $DCGL_{EMC}$, additional measurements may be required beyond the minimum number computed via Equation 5-7 or 5-8. The effect of these additional measurement points is to tighten the grid spacing for the fixed measurements, thus reducing the probability of missing a small area of elevated activity to an acceptable level.

The adequacy of the scanning technique will be evaluated by calculating a scanning MDC, expressed as a fraction of the $DCGL_{EMC}$ as shown below.

As described in Section 5.4.6.3, the relationship between the $DCGL_{EMC}$ and the $DCGL_W$ using the area factor for nuclide i is:

$$DCGL^i_{EMC} = AF^i DCGL^i_W$$ (Equation 5-9)

Where, AF^i is the area factor for radionuclide i.

For soil, the relationship between a scanning minimum detectable count rate (MDCR) and the minimum detectable soil concentration is:

$$MDC^i(pCi/g) = \frac{MDCR(cpm)}{E^i(cpm/pCi/g)} \quad \text{(Equation 5-10)}$$

Where, E^i is the conversion factor (in cpm/pCi/g) for the radionuclide i (instrument efficiency for scanning).

The soil scanning MDC expressed as a fraction of the $DCGL_{EMC}$ is calculated by the following equation:

$$MDC(fDCGL_{EMC}) = MDCR\sum\frac{f^i}{E^i DCGL^i_{EMC}} \quad \text{(Equation 5-11a)}$$

Or

$$MDC(fDCGL_{EMC}) = MDCR\sum\frac{f^i}{E^i AF^i DCGL^i_W} \quad \text{(Equation 5-11b)}$$

Where f^i is the decimal fraction of the radionuclide mix comprised by ETD radionuclide i and is based upon characterization data, as a part of the Final Status Survey. If characterization data indicates the presence of HTD radionuclide, then a surrogate $DCGL_{EMC}$ will be calculated for an ETD radionuclide using equation 5-6 where $DCGL_{EMC}$ is substituted for $DCGL_W$ and equation 5-11a applied.

An example calculation to determine the soil scanning MDC expressed as a fraction of the $DCGL_{EMC}$ when multiple radionuclides are present is shown below:

Assumptions:
Two radionuclides are present; Cs-137 and Co-60
Cs-137 fraction in mix (f) = 0.75
Co-60 fraction in mix (f) = 0.25
Cs-137 efficiency (E) = 228 cpm/pCi/g
Co-60 efficiency (E) = 882 cpm/pCi/g
Elevated area = 100 m^2
Example Cs-137 area factor (AF) = 2.93
Example Co-60 area factor (AF) = 1.41
Example Cs-137 $DCGL_W$ = 7.91 pCi/g
Example Co-60 $DCGL_W$ = 3.81 pCi/g
MDCR = 2,000 cpm

$$MDC(fDCGL_{EMC}) = 2{,}000\left[\frac{0.75}{(228)(2.93)(7.91)} + \frac{0.25}{(882)(1.41)(3.81)}\right] = 0.4$$

For scanning building surfaces, the following equation from MARSSIM provides the method to calculate the MDC for beta-gamma measurements. It has been repeated here below for clarity:

$$MDC(dpm/100cm^2) = \frac{1.38\sqrt{B}}{\sqrt{p}\varepsilon_i\varepsilon_s\left(\frac{A}{100}\right)t} \qquad \text{(Equation 5-12)}$$

1.38 = sensitivity index,
B = number of background counts in time interval t,
p = surveyor efficiency,
ε_i = instrument efficiency for the emitted radiation (counts per emission),
ε_s = source efficiency (intensity) in emissions per disintegration,
A = sensitive area of the detector (cm^2),
t = time interval of the observation while the probe passes over the source (min)

With t is the time the detector spends over a source of radionuclide i which can be related to the travel velocity of the probe, V(cm/min), and the minimum dimension of the detector, L (cm), as:

$$t(\min) = \frac{L(cm)}{V(cm/\min)} \qquad \text{(Equation 5-13)}$$

Equation 5-12 can be rewritten as follows:

$$MDC^i(dpm/100cm^2) = \frac{1.38\sqrt{\frac{B}{t^2}}}{\sqrt{p}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)} = \frac{1.38\sqrt{\frac{R_b}{t}}}{\sqrt{p}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)} = \frac{1.38\sqrt{R_b}}{\sqrt{p}\sqrt{t}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)} \qquad \text{(Equation 5-14)}$$

Substituting Equation 5-13 into 5-14 gives:

$$MDC^i(dpm/100cm^2) = \frac{1.38\sqrt{R_b}}{\sqrt{p}\varepsilon_i^i\varepsilon_s^i\left(\frac{A}{100}\right)\sqrt{\frac{L}{V}}} \qquad \text{(Equation 5-15)}$$

The MDCR for an analog detector with an audible signal can be expressed as:

$$MDCR(cpm) = \frac{1.38\sqrt{B}}{t} = \frac{1.38\sqrt{R_b}}{\sqrt{t}} = \frac{1.38\sqrt{R_b}}{\sqrt{\frac{L}{V}}} \qquad \text{(Equation 5-16)}$$

Using this, Equation 5-15 is re-written as:

$$MDC^i(dpm/100cm^2) = \frac{MDCR}{\varepsilon_i^i \varepsilon_s^i \left(\frac{A}{100}\right)\sqrt{p}} \qquad \text{(Equation 5-17)}$$

To allow for multiple ETD radionuclides, the scan MDC expressed as a fraction of the $DCGL_{EMC}$ is:

$$MDC(fDCGL_{EMC}) = \frac{MDCR}{\left(\frac{A}{100}\right)\sqrt{p}} \sum \frac{f^i}{\varepsilon_i^i \varepsilon_s^i DCGL_{EMC}^i} \qquad \text{(Equation 5-18)}$$

Hard-to-detect radionuclides are included by using the surrogate ratio in determining the $DCGL_{EMC}$.

By substituting $DCGL_{EMC}^i = AF^i DCGL_W^i$ into Equation 5-18 yields the building surface scanning MDC equation expressed as a fraction of the $DCGL_{EMC}$:

$$MDC(fDCGL_{EMC}) = \frac{MDCR}{\left(\frac{A}{100}\right)\sqrt{p}} \sum \frac{f^i}{\varepsilon_i^i \varepsilon_s^i AF^i DCGL_W^i} \qquad \text{(Equation 5-19)}$$

If YAEC intends to use a method of calculating MDC, different than that in MARSSIM as presented above, a technical evaluation of the method will be written. This evaluation will be available for NRC inspection in support of final status survey activities. RAI #19

An example calculation to determine the building surface scanning MDC expressed as a fraction of the $DCGL_{EMC}$ when multiple radionuclides are present is shown below:

Assumptions:
Two radionuclides are present; Cs-137 and Co-60
Cs-137 fraction in mix (f) = 0.75
Co-60 fraction in mix (f) = 0.25
Probe width (L) = 10.2 cm (4 inches)
Scan rate (V) = 305 cm/min (2 inches/sec)
Background count rate (R_b) = 200 cpm
p = 0.5
$\varepsilon_i = 0.3$ for Co-60
$\varepsilon_i = 0.38$ for Cs-137
$\varepsilon_s = 0.25$ for Co-60
$\varepsilon_s = 0.5$ for Cs-137
Probe area (A) = 100 cm^2
MDCR = 27.6 cpm
Elevated area = 10 m^2
Example Cs-137 area factor (AF) = 2.6
Example Co-60 area factor (AF) = 2.5
Example Cs-137 $DCGL_W$ = 4.30E+04 dpm/100 cm^2
Example Co-60 $DCGL_W$ = 1.11E+04 dpm/100 cm^2

$$MDC(fDCGL_{EMC}) = \frac{27.6}{\left(\frac{100}{100}\right)\sqrt{\frac{10.2}{305}}}\left[\frac{0.75}{(0.38)(0.5)(2.6)(4.30E4)} + \frac{0.25}{(0.3)(0.25)(2.5)(1.11E4)}\right] = 0.02$$

As shown in these two examples, the fraction of $DCGL_{EMC}$ is less than one. Therefore no additional measurements are required.

If the value of MDC ($fDCGL_{EMC}$) is greater than one, additional measurements may need to be taken in the survey unit as determined by taking the following steps.

Determine the size of the elevated area from the area factors corresponding to the highest $fDCGL_{EMC}$ which is still less than one. That area is denoted as A_{EMC}.

The number of measurements (N_{EMC}) required to detect an area of elevated concentration equal to A_{EMC} is then computed as

$$N_{EMC} = \frac{A}{A_{EMC}} \qquad \text{(Equation 5-20)}$$

where A is the total area of the survey unit. N_{EMC} (computed via Equation 5-20) is then compared to N, the number of fixed measurement points computed via Equation 5-7 or 5-8. The larger of N_{EMC} or N is then used as the requisite number of fixed measurement locations and to compute the grid spacing.

5.5.1.6 Determining Measurement Locations

For Class 1 and Class 2 survey units, fixed measurements will be performed over a systematic measurement pattern consisting of a grid having either a triangular or a square pitch. The pitch (grid spacing) will be determined based on the number of measurement required and whether the desired grid is triangular or square.

Systematic grids will not be used for surveys involving fixed measurements for Class 3 units. Instead, fixed measurement locations will be selected at random throughout the survey unit area by generating pairs of random numbers between zero and one. One pair of random numbers will be generated for each fixed measurement to be made. The random number pairs, representing (x, y) coordinates, will be multiplied by the maximum length and width dimensions of the survey unit to yield the location for each fixed measurement. For odd-shaped survey units, a rectangular area encompassing the survey unit will be used to establish the maximum length and width. A new pair of random numbers will be generated if any of them give locations that are not actually within the survey unit boundaries. New pairs of numbers will also be generated in cases where a measurement cannot be made at a specific location because of an obstruction, inaccessibility, etc.

The spacing to be used in setting up the systematic grid used to establish fixed measurement locations for Class 1 and Class 2 areas will be computed as

$$L = \sqrt{\frac{A}{0.866N}} \quad \text{for a triangular grid, or} \qquad \text{(Equation 5-21)}$$

$$L = \sqrt{\frac{A}{N}} \quad \text{for a square grid} \qquad \text{(Equation 5-22)}$$

where L = grid spacing (dimension is square root of the area),
A = the total area of the survey unit, and
N = the desired number of measurements.

In the case of Class 1 units, the value used for N in Equations 5-21 and 5-22 should be the larger of that from Equations 5-7 or 5-8 (if the scan MDC is sufficient to see small areas of elevated activity) or Equation 5-20. The value of N should include additional measurements required to ensure against losses or unusable data.

Once the grid spacing is established, a random starting point will be established for the survey pattern using the same method as described above for selecting random locations for Class 3 units. Starting from this randomly-selected location, a row of points will then be established parallel to one of the survey unit axes at intervals of L. Additional rows will then be added parallel to the first row. For a triangular grid, additional rows will be added at a spacing of 0.866L from the first row, with points on alternate rows spaced mid-way between the points from the previous row. For a square grid, points and rows will be spaced at intervals of L. Section 5.5.2.5 of MARSSIM describes the process to be used for selecting fixed measurement locations and provides examples of how to establish both a systematic grid and random measurement locations.

Software tools that accomplish the necessary grid spacing, including random starting points and triangular or square pitch, may be employed during Final Status Survey. When available, this software will be used with suitable mapping programs to determine coordinates for a global positioning system (GPS). The use of these tools will provide a reliable process for determining, locating and mapping measurement locations in open land areas separated by large distances and will be helpful during independent verification.

5.5.2 Judgmental Assessments

For those Class 2 and Class 3 survey units for which 100% of the area is not surveyed, it is important to consider performing judgmental assessments to augment any regimented measurements made in accordance with the above guidance. Such assessments may consist of biased sampling or measurements performed in locations selected on the basis of site knowledge and professional judgment. Judgmental assessments serve to provide added assurance that residual contamination at the site has been adequately located and characterized.

In addition to any judgmental measurements deemed necessary to provide comprehensive survey coverage for a given survey unit, the survey process should include an isotopic mix evaluation in cases where measurable activity still exists. Doing so will allow an assessment of the adequacy of the $DCGL_W$ selected for the survey unit in question to be made during the subsequent data assessment phase. For gross count measurements (i.e., not radionuclide specific), radionuclide mix information will also allow for an evaluation of the suitability of the efficiencies applied in converting raw count data to activity. FSS procedures specify the percentage and/or number of samples that need to be analyzed when evaluating a radionuclide mix, consistent with Section 5.4.6.2. The process relies on a graded approach that depends upon the activity levels present. This procedure will be available onsite for NRC review. RAI #21

The basis for judgmental assessments will be documented in the Final Status Survey Plan.

by the elevated measurement comparison, any indication of residual radioactivity in excess of the $DCGL_W$ during the scan of a Class 2 unit will warrant further investigation. For Class 3 units, any scan measurement that shows a positive indication over background will be investigated.

In cases where an advanced survey method is used instead of fixed measurements or samples, the investigation levels given in Table 5-2 for fixed measurements or samples will be applied with the exception of the statistical outlier test for measurements in Class 1 survey units. In cases where advanced survey methods are used as a means of traditional scanning, the investigation levels for scan measurements in Table 5-2 will be used.

5.5.3.2 Investigations

Locations where initial measurements give results that exceed an applicable investigation level will be identified for confirmatory measurements. If it is confirmed that residual activity exists in excess of the investigation level, additional measurements will be made to determine the extent of the area of elevated activity and to provide reasonable assurance that other areas of elevated activity do not exist. Potential sources of the elevated activity will be postulated and evaluated against the original classification of the survey unit and its associated characterization data. The possibility of the source of the elevated activity having affected other adjacent or nearby survey units will also be evaluated. Documentation will be compiled containing the results from the investigation surveys and showing any areas where residual activity was confirmed to be in excess of the investigation level. If residual activity in excess of the applicable investigation level is confirmed, the documentation will also address the potential source(s) of the activity and the impact this has on the original classification assigned to the survey unit. A decision will then be made regarding re-classification of the unit in whole or in part.

5.5.3.3 Remediation

"Remediation" in the context of the LTP is intended to mean activities performed to meet the criteria of 10CFR20, Subpart E. Activities to remove materials may be performed for other reasons (such as removal of materials associated with decommissioning activities, removal of soils for use as fill in a different area of the site, removal of materials for worker ALARA considerations, or removal of materials for non-radiological remediation), and thus are not considered to be "remediation." If during the time of the Final Status Survey, the survey area is found not to "pass" or any areas of residual activity of residual activity are found to be in excess of the $DCGL_{EMC}$ remediation will be performed. Areas of residual activity may also need to be remediated to meet the ALARA criterion. Remediation actions are discussed in Section 4 and documented as described in Section 5.8.

RAI #22

RAI #23

5.5.3.4 Re-classification

The decision to re-classify an area, or part of an area, is made following a review of the basis for the original classification, considering the evaluation process outlined in Section 5.5.3.2

(consistent with MARSSIM). This process includes sufficient additional measurements to confirm the residual contamination, determine the nature and extent of the contamination present, provide assurance that other areas of elevated activity do not exist within the survey unit, and evaluate the impact (if any) of the affected area on nearby survey units. The results of these measurements will be evaluated, and the area, or part of the area, will be re-classified and re-surveyed per Section 5.5.3.5 in a manner that is consistent with the process described in MARSSIM. Additionally, if required remediation actions are taken in the area, it will be re-surveyed per Section 5.5.3.5 in a manner that is consistent with the process described in MARSSIM. Re-classification of areas from a less to a more restrictive classification may be done without prior NRC approval; however, re-classification to a less restrictive classification would require NRC notification at least 14 days prior to implementation, consistent with the guidance in Appendix 2 to NUREG-1700, Revision 1.

5.5.3.5 Re-survey

If a survey unit is re-classified (in whole or in part), or if remediation is performed within a unit, then the areas affected are subject to re-survey. Any re-surveys will be designed and performed as specified in this plan based on the appropriate classification of the survey unit. That is, if a survey unit is re-classified or a new survey unit is created, the survey design will be based on the new classification.

For example, a Class 3 area with unexpected radioactivity will be subdivided into at least two areas. One of these may remain as a Class 3 area while the other may now be a Class 2 area. For the Class 3 area, either a new survey will be designed and implemented or the Type I and Type II errors will be adjusted and additional measurements made until the required number of measurements is met (see Section 5.5.1). NRC will be notified prior to subdividing a survey area. The Type I and Type II decision error rates will be documented in the final status survey report. RAI#4, #17, #25

A Class 2 area that is subdivided due to the levels of radioactivity identified will be divided into at least two areas as well. In this case if the original survey design criteria has been satisfied, no additional action is required, otherwise the remaining Class 2 survey unit will be redesigned. The new sub-divided survey unit will be surveyed against a new survey design.

If an area has passed the WRS or Sign Test and additional clean-up is required in only a small area of a Class 1 survey unit (e.g., for ALARA purposes), any replacement measurements or samples required will be made within the remediated area at randomly selected locations following verification that the remediation activities did not affect the remainder of the unit. Re-survey will be required in any area of a survey unit affected by subsequent remediation activities. RAI#26

5.6 Final Status Survey Implementation and Data Collection

The requirements and objectives outlined in this plan and the project QA plan will be incorporated into Standard Operating Procedures (SOPs). Procedures will govern the survey design process, survey performance and data assessment (decision making). The final status

maximum residual activity concentrations can be quantified over any area desired, allowing one to assess compliance with the applicable criteria ($DCGL_W$ or $DCGL_{EMC}$) by inspection.

If advanced technology instrumentation is selected for use, a technical support document will be developed which describes the technology to be used and how the technology meets the objectives of the survey. This document will be available for NRC inspection in support of final status survey activities.

5.6.1.4 Bulk Spectroscopy Monitor

The bulk spectroscopy monitor consists of eight coaxial high purity Germanium detectors (each with approximately 40% relative efficiency) which are configured for use with specially-designed computer software. The software supports mathematically determined detector efficiency calibration, which is particularly important in field applications where source-based calibrations are not practical. The monitoring system also includes software to permit simultaneous spectra acquisition from all eight detectors and subsequent summing of the spectra for analysis, including application of an efficiency appropriate for the summed spectra and for the geometry of the measured container and its contents.

It is anticipated that the sensitivity of the detection system will be capable of achieving approximately 10% of the applicable DCGLs (e.g., soil or concrete debris) and the volumetric environmental "free-release" criteria for solid materials. The location of the monitoring system will be such that licensed radioactive material remaining on site (e.g., ISFSI and material storage areas) will have minimal impact on the sample count time necessary to achieve the desired detection limits.

5.6.1.5 Other Advanced Survey Technologies

Other instruments and methods that may be used for final status surveys include, but are not limited to, in situ gamma spectrometry, in situ object counting systems, and systems capable of traversing ducting or piping. Like the advanced technologies discussed above, these other methods may in some cases provide sufficient areal coverage so that augmenting the measurement with scanning is not necessary.

In situ gamma spectrometry is an established technique for assaying the average radionuclide concentration in large volumes of material. It has the advantage of being able to assess large areas with a single measurement. If desired, the detector's field of view can be reduced through collimation to allow assay of smaller areas.

In situ object counting refers to gamma spectrometry systems that include software capable of modeling photon transport in complex geometries for the purpose of estimating detector efficiencies. This eliminates the need for a calibration geometry representing the object to be counted.

5.6.1.6 Samples

Sampling is the process of collecting a portion (typically 1 liter) of a medium as a representation of the locally remaining medium. Extraneous materials such as undesired vegetation, debris, and rocks are removed during sampling. Then the collected portion of the medium is then analyzed to determine the radionuclide concentration. Examples of materials that may be sampled include soil, sediments, concrete, paint, and groundwater. RAI #30

Section 5.9, "Final Status Survey Quality Assurance and Quality Control Measures" addresses QA requirements for final status survey activities that apply to onsite and offsite laboratories employed to analyze samples as a part of the final status survey process. Performance of laboratories will be verified periodically by QA auditors. This verification will include reviews of personnel training, procedures and equipment operation.
Trained and qualified individuals will collect and control samples. Sampling activities will be performed under approved procedures. YAEC will use a sample tracking and control system to ensure sample integrity.

5.6.2 Survey Instrumentation

5.6.2.1 Instrument Selection

The selection and proper use of appropriate instruments for both fixed measurements and laboratory analyses is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The survey plan design must establish acceptable measurement techniques for scanning and direct measurements. The DQO process must include consideration as to the type of radiation, energy spectrum and spatial distribution of radioactivity as well as the characteristics of the medium to be surveyed (e.g.; painted, scabbled, chemically decontaminated).

The particular capabilities of a radiation detector establish its potential for being used in conducting a specific type of survey based on the factors discussed above. Radiation survey parameters that will be needed for final survey purposes include surface activities and radionuclide concentrations in soil. To determine these parameters, both field measurements and laboratory analyses will be necessary. For certain radionuclides or radionuclide mixtures, both alpha and beta radiation may have to be measured. In addition to assessing average radiological conditions, the survey objectives must address identifying small areas of elevated activity.

The DQO process must consider the field conditions the instrument will be used in to determine the affect and magnitude of variation from conditions established during calibration. These conditions might include source to detector geometry (including distance and solid angle), size and distribution of the source relative to the detector, and composition and condition of surface to be assessed. Most of these factors should have been determined during the instrument selection process. In some cases, instrument efficiencies may require modifications to account for surface conditions or coverings. Such modifications, if necessary, will be established using the information in Section 5 of NUREG-1507 and pertinent site characterization data. This will be performed during the planning process and documented by a technical support document and referenced in the survey plan. This technical support document will include the evaluation supporting instrument selection.

5.6.2.3 Response Checks

The DQO process determines the frequency of response checks, typically before issue and after an instrument has been used (typically at the end of the work day but in some cases this may be performed during an established break in activity, e.g., lunch). This additional check will expedite the identification of a potential problem before continued use in the field. Instrumentation will be response checked in accordance with plant procedures. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that data are discarded, the affected area will be re-surveyed. FSS procedures require that all adjustments to data be documented in the FSS reports. RAI #31

5.6.2.4 MDC Calculations

The DQO process evaluates the ability of the instrument to measure radioactivity at levels below the applicable DCGL. This evaluation will be performed and documented by a technical support document and referenced by the survey plan. This evaluation may also be included with the technical support document discussed in Section 5.6.2.1 above.

Instrument detection limits are typically quantified in terms of their minimum detectable concentration, or MDC. The MDC is the concentration that a given instrument and measurement technique can be expected to detect 95% of the time under actual conditions of use.

Instruments and methods used for field measurements will be capable of meeting the investigation level in Table 5-2. RAI #32

Before any measurements are performed, the instruments and techniques to be used must be shown to have sufficient detection capability relative to the applicable DCGLs. The detection capability of a given instrument and measurement technique is quantified by its MDC.

5.6.2.4.1 MDCs for Fixed Measurements

Per NUREG-1507, MDCs for fixed measurements are computed as

$$MDC_{fixed} = \frac{3 + 4.65\sqrt{B}}{Kt} \qquad \text{(Equation 5-23)}$$

where 3 and 4.65 = constants as described in NUREG-1507;
B = background counts during the measurement time interval (t);
t = counting time; and
K = a proportionality constant that relates the detector response to the activity level in the sample being measured.

The proportionality constant K typically encompasses the detector efficiency, self-absorption factors and probe area corrections, as required. The dimensions of the counting interval "t" are consistent with those for the MDC and the proportionality constant K. Thus, "t" would be in minutes to compute an MDC in dpm/100 cm^2.

An example calculation to determine the MDC_{fixed} for the detection of Co-60 with a 100 cm^2 gas proportional detector is shown below.

Assumptions:
Background count rate = 200 cpm
t = 1 minute
B = 200 counts in the measurement time interval (t)
$K = \varepsilon_i \varepsilon_s(A/100)$, where A = area of the detector in cm^2
ε_i = 0.38 counts per emission
ε_s = 0.25 (from ISO 7503-1) emissions per disintegration
A = 100 cm^2

$$MDC_{fixed} = \frac{3 + 4.65\sqrt{200}}{(0.38)(0.25)(100/100)(1)} = 724 dpm/100cm^2$$

Actual values for ε_s will be selected from ISO 7503-1 or NUREG-1507 or empirically determined and documented prior to performing the final status survey.

5.6.2.4.2 MDCs for Beta-Gamma Scan Surveys for Structure Surfaces

As recommended in Section 5.1 of Appendix E to NUREG-1727, MDCs for surface scans for structure surfaces for beta and gamma emitters will be computed via

$$MDC_{structure,scan} = \frac{1.38\sqrt{B}}{\sqrt{p}\varepsilon_i\varepsilon_s\left(\frac{A}{100}\right)t}\ \text{dpm/100cm}^2 \qquad \text{(Equation 5-24)}$$

where 1.38 = sensitivity index,
B = number of background counts in time interval t,
p = surveyor efficiency,
ε_i = instrument efficiency for the emitted radiation (counts per emission),
ε_s = source efficiency (intensity) in emissions per disintegration,
A = sensitive area of the detector (cm^2),
t = time interval of the observation while the probe passes over the source (minutes).

The value of 1.38 used for the sensitivity index corresponds to a 95% confidence level for detection of a concentration at the scanning MDC with a false positive rate of 60%. The numerator in Equation 5-24 represents the minimum detectable count rate that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) will be taken to be 0.5, as recommended in Section A.5.1 of Appendix A to NUREG-1757. The factor of 100 corrects for probe areas that are not 100 cm^2. In the case of a scan measurement, the counting interval is the time the probe is actually over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. The source efficiency term (ε_s) in Equation 5-24 may be adjusted to account for effects such as self-absorption, as appropriate.

An example calculation to determine the $MDC_{structure,\ scan}$ for the detection of Co-60 with a 100 cm^2 gas proportional detector follows.

Assumptions:

Probe width = 4 inches
Scan rate = 2 inches/sec
Background count rate = 200 cpm
t = 2 seconds = 0.033 minute
B = 6.7 counts in the measurement time interval (t)
p = 0.5
ε_i = 0.38 counts per emission
ε_s = 0.25 (from ISO 7503-1) emissions per disintegration
A = 100 cm^2

$$MDC_{structure,scan} = \frac{1.38\sqrt{6.7}}{\sqrt{0.5}(0.38)(0.25)\left(\frac{100}{100}\right)(0.033)} = 1611\,\text{dpm}/100\text{cm}^2$$

Actual values for ε_s will be selected from ISO 7503-1 or NUREG-1507 or empirically determined and documented prior to performing the final status survey.

5.6.2.4.3 MDCs for Alpha Scan Surveys for Structure Surfaces

In cases where alpha scan surveys are required, MDCs must be quantified differently than those for beta-gamma surveys because the background count rate from a typical alpha survey instrument is nearly zero (1 to 3 counts per minute typically). Since the time that an area of alpha activity is under the probe varies and the background count rates of alpha survey instruments is so low, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates. In general, it is expected that separate alpha and beta surface activity measurements will not be necessary at the YNPS and that surrogate measurements will instead be used for alpha surface activity assessments (see Section 5.4.6.2).

For alpha survey instrumentation with a background around one to three counts per minute, a single count will give a surveyor sufficient cause to stop and investigate further. Thus, the probability of detecting given levels of alpha emitting radionuclides can be calculated by use of Poisson summation statistics. Doing so (see MARRSIM Section 6.7.2.2 and Appendix J for details), one finds that the probability of detecting an area of alpha activity of 300 dpm/100cm^2 at a scan rate of 3 cm per second (roughly 1 inch per second) is 90% if the probe dimension in the direction of the scan is 10 cm. If the probe dimension in the scan direction is halved to 5 cm, the detection probability is still 70%. Choosing appropriate values for surveyor efficiency, instrument and surface efficiencies will yield MDCs for alpha surveys for structure surfaces. If for some reason lower MDCs are desired, then scan speeds can be adjusted, within practical limits, via the methods of Section 6.7.2.2 and Appendix J of the MARSSIM.

5.6.2.4.4 MDCs for Gamma Scans of Land Areas

Section A.5.1 of Appendix A to NUREG-1757, the values given in Table 6.7 of MARSSIM may be adopted for gamma scans of land areas if NaI detectors of the dimensions considered in the table are used. If larger NaI detectors (e.g., 3 inch by 3 inch) or other detector types (e.g., plastic scintillator) are used, then the scan MDC will be computed using the methods of Section 6.7.2.1 of MARSSIM and documented. This is the same method as was used to derive the values given in MARSSIM Table 6.7. As an alternative, a specific technical study may be performed and documented to establish efficiency to a soil standard consistent with MARSSIM guidance. RAI #33

The radionuclides represented in MARSSIM Table 6.7 encompass those expected to be encountered in gamma scans for land areas at the YNPS. If desired, the methods of Sections 5.4.6.1 and 5.4.6.2 of this plan may be used to establish scan MDCs based on radionuclide mix ratios. Alternatively, the most limiting value for the radionuclide mix may be used, with most limiting in this case meaning the radionuclide for which the MDC is the largest fraction of its

$DCGL_W$ for soil, while still meeting the criteria of 5.5.3.1. Thus, selecting the highest MDC of the radionuclide constituents will result in a more rigorous final status survey design, and therefore, is more conservative. RAI #34

An example calculation to determine the $MDC_{land\ scan}$ for the detection of Cs-137 with a 2"x2" NaI detector is shown below.

The minimum detectable count rate (MDCR) for a surveyor must be calculated prior to determining the scan MDC. The MDCR is dependent upon the background counts expected during time, t, at which the detector is located over the localized contamination. The minimum detectable count rate (MDCR) for a surveyor is calculated using the following expression:

$$MDCR_{surveyor} = \frac{1.38\sqrt{b}}{\sqrt{p}\ t} \qquad \text{(Equation 5-25)}$$

where b = the background counts expected during time, t
t = the time the detector is located above the localized contamination
p = the surveyor efficiency

Assumptions:

Scan speed = 0.5 meters/sec
Localized contamination diameter = 56 cm
Background count rate = 7000 cpm
b = 130.67 counts in the measurement time interval (t)
t = 0.0187 minute
p = 0.5

$$MDCR_{surveyor} = \frac{1.38\sqrt{130.67}}{\sqrt{0.5}\ (0.019)} = 1195cpm$$

Next, the minimum detectable exposure rate (MDER) is calculated by dividing the $MDCR_{surveyor}$ by the response to exposure rate factor for Cs-137 of 900 cpm/μR/h from MARSSIM Table 6.7 as follows:

$$MDER = \frac{1195cpm}{900cpm/\mu R/h} = 1.33\mu R/h$$

The Microshield™ modeling code is used to calculate the exposure rate from the localized contamination. Assuming a localized contamination depth of 15 cm, a density of 1.6 g/cm^3, a dose point of 10 cm above the surface and an initial concentration of 5 pCi/g of Cs-137, results

Table 5-4
Available Instruments and Nominal Detection Sensitivities

Instrument	Application	Nominal Efficiency (Not Media Specific)	Nominal Background	Nominal MDC (fixed measurement)	Nominal Scan MDC
pancake GM probe (20 cm^2)	beta-gamma scans or fixed measurements for structure surfaces	17% (Tc-99)	50 cpm	1,050 dpm/100 cm^2 (1 minute count)	3140 dpm/100 cm^2
gas proportional counter (100 cm^2)	alpha or beta scans or fixed measurements for structure surfaces	β plateau: 16% (Tc-99); α plateau: 23% (Am-241)	350 cpm (β plateau); 15 cpm (α plateau)	560 dpm/100 cm^2 (β plateau) 90 dpm/100 cm^2 (α plateau); 1 minute counts	1770 dpm/100 cm^2 (β plateau); 400 dpm/100 cm^2 (α plateau)
plastic scintillator (100 cm^2)	beta-gamma scans or fixed measurements for structure surfaces	30% (Co-60)	600 cpm	390 dpm/100 cm^2 (1 minute count)	1230 dpm/100 cm^2
dual-phosphor scintillator (100 cm^2)	scans or fixed measurements; α and β, independently or simultaneously	20% (Co-60) 18% (Am-241)	300 cpm (β mode); 6 cpm (α mode)	420 dpm/100 cm^2 (β mode); 80 dpm/100 cm^2 (α mode)	1300 dpm/100 cm^2 (β mode); 400 dpm/100 cm^2 (α mode)
ZnS scintillator (100 cm^2)	alpha scans or fixed measurements on structure surfaces	19% (Pu-239)	2 cpm	50 dpm/100 cm^2 (1 minute count time)	400 dpm/100 cm^2
HPGe	in-situ gamma spectroscopy – soil	Varies with energy and geometry	Varies with energy and geometry	0.05 pCi/g Co-60 0.05 pCi/g Cs-137 (10 minute counts)	N/A
NaI(Tl)	Soil Gamma Scan	.12%	10,000 cpm	N/A	1.6 pCi/g Co-60* 6.3 pCi/g Cs-137
position-sensitive proportional counter	scan-and-record surveys	Co-60 (β): 18% Am-241 (α): 23%	350 cpm/100 cm^2 beta 15 cpm/100 cm^2 alpha	Typical values are 1,925 dpm/100 cm^2 β and 200 dpm/100 cm^2 α	
Bulk spectroscopy monitor (HPGe)	soils and volumetric debris	N/A	N/A	N/A	

*Assumes a 56 cm diameter by 15 cm deep soil contamination volume.

5.6.3.1.3 Buried Piping, Storm Drains, Sewer Systems, Plumbing and Floor Drains

Buried piping, storm drains, plumbing and floor drains are being removed or free-released in accordance with existing plant procedures.

Non-RCA sanitary systems at the YNPS Plant drain to on-site leach fields. These systems are independent of other plant systems and surface water or storm drains. If any residual radioactivity is suspected in portions of the sanitary plumbing systems, evaluations for both the leach fields and the associated system piping may be required. Evaluations required for any affected leach fields will be made as described in Section 5.6.3.2.2 of this plan, for sub-surface activity.

5.6.3.1.4 Concrete Debris

Standing concrete structures will be surveyed and survey results evaluated against ALARA constraints and ability to pass concrete debris DCGL. Additional remediation or segregation of elevated waste for disposal will be performed as indicated by the evaluations.

Concrete debris considered acceptable for meeting the concrete debris DCGL will be processed to appropriate sizes and loaded into containers for volumetric monitoring. Monitoring of the loaded containers will be through use of a multiple intrinsic germanium gamma spectroscopy system (referred to as the "bulk spectroscopy monitor") capable of detection to minor fractions of the concrete debris DCGL. Containers that indicate volumetric activity less than the concrete debris DCGL will be unloaded on site for later use as backfill. Containers that indicate greater than DCGL levels of activity will be removed from site and disposed of in appropriately licensed facilities.

5.6.3.2 Survey Considerations for Outdoor Areas

5.6.3.2.1 Residual Radioactivity in Surface Soils

In this context, surface soil refers to outdoor areas where the soil is considered to be uniformly contaminated from the surface down to 15 centimeters. These areas will be surveyed through combinations of sampling, scanning, in-situ measurements and bulk monitoring, as appropriate. A minimum of 5% of composite surface soil samples will be analyzed for hard-to-detect radionuclides. RAI #36

5.6.3.2.2 Residual Radioactivity in Subsurface Soils

Residual radioactivity in subsurface soils refers to residual radioactivity residing under the top 15 centimeters of soil or underneath structures such as building floors/foundations. Such areas include, but are not limited to, areas under buildings, building floors/foundations, or components where leakage was known or suspected to have occurred in the past and on-site storage areas where radioactive materials have been identified. However, the assessment of subsurface soil contamination is not currently complete. Soil in difficult to access areas such as under buildings

Note that "measurements" in Step 1 above refers to the net result in cases where background-subtracted gross activity measurements (using the paired observation methodology) are being evaluated.

Though it is not anticipated, if any of the data collected from a final status survey are reported as "less than MDC" or as background, actual values (obtained from the laboratory) will be assigned, even if negative, for purposes of applying the Sign test. RAI#39

5.7.3 Elevated Measurement Comparison

The Elevated Measurement Comparison (EMC) consists of comparing each measurement from the survey unit with the investigation levels discussed in Section 5.5.3. The EMC is performed for both measurements obtained on the systematic-sampling grid and for locations flagged by scanning measurements. Any measurement from the survey unit that is equal to or greater than an investigation level indicates an area of relatively high concentrations that should be investigated, regardless of the outcome of the nonparametric statistical tests. Thus, the use of the EMC against the investigation levels may be viewed as assurance that unusually large measurements will receive proper attention regardless of the outcome of those tests and that any area having the potential for significant dose contributions will be identified. The EMC is intended to flag potential failures in the remediation process. It should not be used as the primary means to identify whether or not a unit meets the release criterion.

If residual radioactivity exists in an isolated area of elevated activity in addition to residual radioactivity distributed relatively uniformly across a survey unit, the unity rule will be used to ensure that the total dose is within the release criterion, i.e.,

$$\frac{\delta}{DCGL_W} + \frac{\overline{C}_{elevated} - \delta}{(AreaFactor) \times DCGL_W} < 1 \qquad \text{(Equation 5-26)}$$

where: δ = average concentration outside the elevated area,
$\overline{C}_{elevated}$ = average concentration in the elevated area.

A separate term will be used in Equation 5-26 for each elevated area identified in a survey unit.

Note that EMC considerations generally apply only to Class 1 survey units, since areas of elevated activity should not exist in Class 2 or Class 3 survey units.

5.7.4 Unity Rule

When radionuclide specific measurements are made in survey units having multiple radionuclides, compliance with the radiological release criterion will be assessed through use of

the unity rule, also known as the sum of fractions. The unity rule, represented in the expression below, is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one, i.e.:

$$\frac{C_1}{DCGL_1}+\frac{C_2}{DCGL_2}+\ldots+\frac{C_n}{DCGL_n}\leq 1 \qquad \text{(Equation 5-27)}$$

where:

C_n = Concentration of radionuclide n
$DCGL_n$ = DCGL for radionuclide n

5.7.5 Data Assessment Conclusions

The result of the data assessment is the decision to reject or not to reject the null hypothesis. Provided that the results of investigations triggered by the EMC were resolved, a rejection of the null hypothesis leads to the decision that the survey unit meets the release criterion. If the data assessment concludes that the null hypothesis cannot be rejected, this may be due to one of two things: 1) the average residual concentration in the survey unit exceeds the $DCGL_W$; or 2) the analysis did not have adequate statistical power. "Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. Quantitatively, the power is $1 - \beta$, where β is the Type II error rate (the probability of accepting the null hypothesis when it is actually false). A retrospective power analysis can be used in the event that a survey unit is found not to meet the release criterion to determine if this is indeed due to excess residual activity or if it is due to an inadequate sample size.

Retrospective power analyses will be performed, if necessary, following the methods of MARSSIM Sections I.9.1 and I.9.2 for the Sign test and WRS test, respectively. If the retrospective power analysis indicates insufficient power, then an assessment will be performed to determine whether the observed median concentration and/or observed standard deviation are significantly different from the estimated values used during the DQO process. The assessment may identify and propose alternative actions to meet the objectives of the DQOs. These alternative actions may include failing the unit and starting the DQO process over, remediating some or all of the survey unit and starting the DQO process over and adjusting the LBGR to increase sample size. For example, the assessment determines that the median residual concentration in the survey unit exceeds the $DCGL_W$ or is higher than was estimated and planned for during the DQO process. A likely cause of action might be to fail the unit or remediate and resurvey using a new sample design. As another example, the assessment determines that additional samples are necessary to provide sufficient power. One course of action might be to determine the number of additional samples and collect them at random locations. Note, this method may increase the Type I error, and therefore agreement with the regulator will be necessary prior to implementation. As another example, an assessment determines that additional samples are necessary to provide sufficient power or to resample the survey unit using

RAI #40

RAI #41

a new survey design. This situation may increase the Type I error, and therefore agreement with the NRC will be necessary prior to implementation. RAI #41

There may be cases where the team chooses to accept a lower power as a part of the planning process.. For instance, during the DQO process the calculated relative shift was found to be less than 1. The planning team would adjust the LBGR, evaluates the impact on power and accepts the lower power. In this case, the DQA process would require the planning team to compare the prospective power analysis with the retrospective power analysis and determine whether the lower power is still justified and the DQOs satisfied.

5.8 Final Status Survey Reports

Consistent with Section 4.5.2 of NUREG-1757, the documentation describing the final status survey for a given survey unit will include:

- An overview of the results of the final status survey;
- A discussion of any changes that were made in the final status survey from that described in the LTP;
- A description of the method by which the number of samples was determined for each survey unit;
- A summary of the values used to determine the numbers of sample and a justification for these values;
- The survey results for each survey unit including:
  - The number of samples taken for the survey unit;
  - A map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units, and random locations shown for Class 3 survey units and reference areas;
  - The measured sample concentrations;
  - The statistical evaluation of the measured concentrations, when applicable;
  - Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation;
  - A discussion of anomalous data including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of $DCGL_w$;
  - Discussion of ALARA evaluations performed and conclusions from those evaluations.
  - A statement that a given survey unit satisfied the $DCGL_w$ and the elevated measurement comparison if any sample points exceeded the $DCGL_w$;

scenario definition. Metabolic parameters represent the metabolic characteristics of the receptor and are independent of the scenario definition. Physical parameters are the parameters that would not change if a different group of receptors were considered.

6.2.5.2 Prioritization

The parameters were prioritized in order of importance consistent with NUREG/CR-6697. Prioritization was based on:

- The relevance of the parameter in dose calculations,
- The variability of the dose as a result of changes in the parameter value,
- The parameter type and
- The availability of parameter-specific data.

Priority 1 parameters are considered to be high priority; priority 2 parameters are considered to be medium priority; and priority 3 parameters are considered to be low priority.

6.2.5.3 Treatment

The parameters were treated as either "deterministic" or "stochastic" depending on parameter type, priority, availability of site-specific data and the relevance of the parameter in dose calculations. The "deterministic" modules of the code use a single value for input parameters and generate a single value for dose. The "probabilistic" modules of the code use probability distributions for stochastic input parameters and generate a range of doses.

The behavioral and metabolic parameters are treated as deterministic and were assigned values from NUREG/CR-5512, Volume 3, NUREG/CR-6697, or the applicable code's default library. Physical parameters for which site-specific data are available are also treated as deterministic.

The remaining physical parameters, for which no site-specific data are available to quantify, are classified as either Priority 1, 2, or 3. Priority 1 and 2 parameters are treated as stochastic and are assigned a probability distribution from NUREG/CR-6697. The priority 3 physical parameters are treated as deterministic and are assigned values from NUREG/CR-5512, Volume 3, NUREG/CR-6697, or the applicable code's default library.

6.2.5.4 Sensitivity Analyses

In order to determine the values for those parameters, not already assigned a value as discussed in Section 6.2.5.3, a sensitivity analysis was performed to determine which of the stochastic parameters have an influence on the resulting dose and associated DCGLs. The analysis was performed using the probabilistic modules of RESRAD, Version 6.21, and RESRAD-BUILD, Version 3.21.

The stochastic parameters, as identified in the preceding paragraphs, were generally assigned distribution types and corresponding distribution statistical parameters from NUREG/CR-6697,

Attachment C. Sensitivity analyses were performed on the stochastic parameters using the assigned distributions. To perform the sensitivity analysis, the following information was required:

Sample Specifications: The analyses were run using 2000 observations for soils, 300 observations for building occupancy and 1 repetition for both scenarios. The Latin Hypercube Sampling (LHS) technique was used to sample the probability distributions for each of the stochastic input parameters. The correlated or uncorrelated grouping option was used to preserve the prescribed correlations

Input Rank Correlations: Correlation coefficients were assigned between correlated parameters.

Output Specifications: All of the output options were specified.

Sensitivity analyses were performed for each of the radionuclides. The Partial Rank Correlation Coefficient (PRCC) for the peak of the mean dose was used as a measure of the sensitivity of each parameter.

For the resident farmer scenario, a parameter was identified as sensitive if the absolute value of its PRCC (|PRCC|) was greater than or equal to 0.25 and non-sensitive if the |PRCC| value was less than 0.25. For the building occupancy scenario, a parameter was identified as sensitive if the |PRCC| value was greater than or equal to 0.10 and non-sensitive if the |PRCC| value was less than 0.10. These thresholds (So) were selected based on the guidance included in NUREG/CR-6676 and -6692.

6.2.5.5 Parameter Value Assignment for DCGL Determination

As previously discussed, behavioral and metabolic parameters were assigned values from NUREG/CR-5512 Volume 3, NUREG/CR-6697, or NUREG/CR-6755. If site data was available for physical parameters, that information was used. For Priority 3 physical parameters for which no site data was available, values from NUREG/CR-5512 Volume 3, or NUREG/CR-6697 were used.

Priority 1 and 2 physical parameters were assigned values as follows:

- Priority 1 and 2 physical parameters shown to be sensitive (|PRCC| ≥ So) were assigned conservative values:
  - A site-specific value, or
  - Depending on whether the parameter was positively or negatively correlated with dose, the 75% or 25% quantile value of the distribution was used, respectively.
  - For distributions where the mean value is greater than the 75% value, the mean value was used.

- Priority 1 and 2 physical parameters shown to be non-sensitive ($|PRCC| < So$) were assigned:
  - a distribution or site-specific value, or
  - the median value of the distribution

6.2.6 Code Output and Calculation of DCGL

RESRAD determines an annual peak of the mean dose in mrem/yr, and RESRAD-BUILD determines an average annual dose at the time of the peak dose in mrem/yr. Specifying a unit radionuclide concentration (i.e., 1 pCi/g in RESRAD or 1 pCi/m^2 in RESRAD-BUILD), to be used in conjunction with the parameters selected by the process described previously, a dose conversion factor (DCF) is calculated by the code (in mrem/yr per pCi/g for RESRAD and mrem/yr per pCi/m^2 for RESRAD-BUILD). As suggested in NUREG-1757, DCFs, based upon the peak of the mean dose, were used to calculate the corresponding derived concentration guideline levels (DCGLs) in pCi/g or dpm/100cm^2, representing an annual dose of 25 mrem/yr, using the following equations:

$$\text{DCGL (pCi/g)} = \frac{\text{25 mrem/yr}}{\text{DCF (mrem/yr / pCi/g)}} \qquad \text{(Equation 6-1)}$$

or

$$\text{DCGL (pCi/m}^2\text{)} = \frac{\text{25 mrem/yr}}{\text{DCF (mrem/yr / pCi/m}^2\text{)}} \qquad \text{(Equation 6-2)}$$

$$\text{DCGL (dpm/cm}^2\text{)} = \text{DCGL (pCi/m}^2\text{)} \times \text{(0.037 dps/pCi)} \times \text{(60 sec/min)} \times \text{(m/100cm)}^2 \qquad \text{(Equation 6-3)}$$

$$\text{DCGL (dpm/100cm}^2\text{)} = \text{DCGL (pCi/m}^2\text{)} \times \text{(0.037 dps/pCi)} \times \text{(60 sec/min)} \times \text{(m/100cm)}^2 \times 100 \qquad \text{(Equation 6-4)}$$

6.3 Calculation of DCGLs for Soil

6.3.1 Dose Model

The DCGLs for soil were calculated using the resident farmer scenario. The residual radioactive materials were assumed to be contained in a soil layer on the property that can be used for residential and light farming activities. The average member of the critical group is the resident farmer that lives on the plant site, grows all of his/her diet onsite and drinks water from a groundwater source onsite. The pathways used in this analysis are identified in Section 6.2.2.3.

6.3.2 Conceptual Model

The conceptual model used in the code was based on the site characteristics expected at the time of release of the site. The model is comprised of a contaminated zone underlain by an unsaturated zone underlain by a saturated zone. The contaminated zone is assumed to be at the ground surface with no cover material and the ground water is initially uncontaminated. The model as described is consistent with that described by Yu et al (Reference 6-10). The parameters used to quantify the conceptual model are listed in Appendix 6A.

6.3.3 Parameter Value Assignment

The process described in Section 6.2.5 was used to determine the parameter input values or distributions. The evaluation of site/regional data and the justification of values assigned to the site-specific parameters that comprise the conceptual model are provided in Appendix 6A. The values/distributions assigned to all parameters for the sensitivity analyses and the basis for assigning such values/distributions are summarized in Appendix 6B.

6.3.4 DCGL Determination

The input values assigned to sensitive and non-sensitive parameters for the DCGL runs were based on the process described in Section 6.2.5.5 in conjunction with the sensitivity analysis results presented in Appendix 6C. The DCGL determination was performed using RESRAD Version 6.21 analyses with the input values summarized in Appendix 6D.

The resulting DCFs, based upon the peak of the mean dose, are provided in Appendix 6E. The DCGLs, representing a dose of 25 mrem/yr, determined using Equation 6-1 are also provided in Appendix 6E.

6.4 Calculation of DCGL for Structures

6.4.1 Structure Surface DCGL

6.4.1.1 Dose Model

The dose model used to calculate the surface DCGLs is based upon the building occupancy scenario as defined in NUREG/CR-5512, Volumes 1, 2, and 3 and NUREG-1757. The scenario assumes that the critical group consists of light industrial workers working in the building following license termination. The pathways used in this analysis are those identified in Section 6.2.3.3.

6.4.1.2 Conceptual Model

The conceptual model was developed based on site characteristics expected at the time of license termination. The model is comprised of a room, with dimensions representing the average wall size expected to remain at the site. The four walls and floor of this room are assumed to be contaminated uniformly and to equal levels. This is considered to be a conservative assumption

as normally the amount of contamination on room walls is less than that on the floor and decreases as the distance from the floor increases. No contaminated ceiling is included in the model, as partial rooms/rooms remaining at the time of license termination will either have no ceiling or will be covered with a ceiling constructed of new (uncontaminated) materials. Appendix 6F provides the details for the determination of the room dimensions.

6.4.1.3 Parameter Value Assignment

The process described in Section 6.2.5 was used to determine the parameter input values or distributions. The evaluation of site/regional data and the justification of values assigned to the site-specific parameters that comprise the conceptual model are provided in Appendix 6F. The values/distributions assigned to all parameters for the sensitivity analyses and the basis for assigning such values/distributions is summarized in Appendix 6G. Preliminary runs were performed prior to the sensitivity analyses to determine the time in which the maximum dose occurred.

6.4.1.4 DCGL Determination

The input values assigned to sensitive and non-sensitive parameters for the DCGL runs were based on the process described in Section 6.2.5.5 in conjunction with the sensitivity analysis results presented in Appendix 6H. The DCGL determination was performed using RESRAD-BUILD Version 3.21 analyses with the input values summarized in Appendix 6I.

The resulting DCFs, based upon the average dose during the year that the maximum dose occurs, are provided in Appendix 6J. The DCGLs, representing a dose of 25 mrem/yr, determined using Equation 6-2 through 6-4 are also provided in Appendix 6J.

6.4.2 Structure Volumetric DCGL

Two methodologies have been used in calculating volumetric DCGLs for contamination in concrete:

- a modified resident farmer scenario using RESRAD, which uses a diffusion based release rate of radionuclides from the concrete, has been used to determine DCGLs for subsurface partial structures, and

- a modified resident farmer scenario using RESRAD, assuming an instantaneous release of radionuclides from the concrete, has been used to determine DCGLs for concrete debris from building demolition.

6.4.3 Calculation of DCGLs for Subsurface Partial Structures

6.4.3.1 Dose Model

The dose model used to calculate the volumetric DCGLs for subsurface partial surfaces is based upon the resident farmer as defined in NUREG/CR-5512, Volumes 1, 2, and 3 and NUREG-1757. The average member of the critical group is the resident farmer that lives on the plant site, grows all of his/her diet onsite and drinks water from a groundwater source onsite. The pathways used in this analysis are identified in Section 6.2.2.3.

6.4.3.2 Conceptual Model

The conceptual model used in the code was based on the site characteristics expected at the time of release of the site. YNPS has modeled five structures as remaining at the time of license termination:

- Primary Auxiliary Building (PAB) Primary Drain Collection Tank (PDCT) Cubicle
- PAB Gravity Drain Tank (GDT) Cubicle
- Spent Fuel Pit (SFP)*
- Waste Disposal Building (WDB) Cubicle
- Elevator Pit

The model was applied to a set of radionuclides determined to exist in samples of concrete from the IX Pit/SFP complex (Reference 6-11)

The following approach was taken: (1) to determine the source term from the concrete to the groundwater and (2) to determine the dose from this source term.

Two mechanisms were considered in determining the source term: diffusive release from the concrete and sorption onto the backfill and soil that surround the facilities. Diffusive release was found to be the rate-limiting step for the radionuclides in the analyses (for the six radionuclides identified in concrete samples).

Additional analyses were performed to determine the impact that contaminant distribution in the walls has on release rates. These analyses showed that for every radionuclide except H-3 (that is, C-14 Co-60, Ni-63, Sr.-90, and Cs-137), the peak release rate was affected by the concentration within only the first inch of the wall. Therefore, the effect of having a non-uniform distribution in concentration through the thickness of the wall is minimal for these radionuclides. However, H-3 has a higher concrete diffusion coefficient than the other radionuclides addressed. Accordingly, release of H-3 from concrete is influenced by concentrations deeper within the wall (i.e., a few inches from the surface).

Using a concentration of 1 pCi/g and a concrete density of 2.5 g/cm^3, the total release to the subsurface was estimated for each radionuclide. Values for RESRAD input parameters were

* YAEC's current plan is to completely demolish the Spent Fuel Pit.

selected to match the release rate calculated. RESRAD was then used to calculate the water pathway dose, using the same assumptions in the soil DCGL calculations.

6.4.3.3 Parameter Value Assignment

The total release from the subsurface structures was estimated for each radionuclide, using a concentration of 1 pCi/g and a concrete density of 2.5 g/cm^3. Input parameter values for RESRAD were selected to match the release rate calculated by DUST-MS (Reference 6-12). Using the same assumptions as used in the soil DCGL calculations, RESRAD was used to calculate the dose from the water pathway.

6.4.3.4 DCGL Determination

The doses determined from the assumed concentrations of 1 pCi/g were scaled to 0.5 mrem/yr and are provided in Reference 6-13. The DCGLs representing a dose of 0.5 mrem/yr are provided in Appendix 6K.

6.4.4 Calculation of DCGLs for Concrete Debris

6.4.4.1 Dose Model

The DCGLs for concrete debris were calculated using the resident farmer scenario. The residual radioactive materials were assumed to be contained in a layer of concrete debris located on the property that can be used for residential and light farming activities. The average member of the critical group is the resident farmer that lives on the plant site, grows all of his/her diet onsite and drinks water from a groundwater source onsite. The pathways used in this analysis are identified in Section 6.2.2.3. Note that the intruder scenario from NUREG-1757, Appendix J, has been incorporated into this model by the very conservative assumption that no cover exists over the debris on the site.

6.4.4.2 Conceptual Model

6.4.4.2.1 General Model

The conceptual model is based on the site characteristics expected at the time of license termination. The model includes the use of concrete debris for filling cellar holes and site grading. It also assumes the presence of a potential intruder who removes all of the clean material that will cover the concrete debris. The use of the resident farmer scenario in RESRAD assumes that normal farm activities will take place on the concrete debris including the growing of food crops and the raising of livestock.

Key assumptions of the conceptual model:

The concrete debris contains residual radioactivity. This concrete is used to fill cellar holes and grade the site and is identified as the contaminated zone. The model uses the very conservative assumption that the entire contaminated zone extends into the water table. Although the

Massachusetts Department of Environmental Protection requires 3 feet of uncontaminated cover over the concrete fill, an intruder scenario has been incorporated into the conceptual model, consistent with NUREG-1757, and thus no cover is assumed.

The on-site well for drinking, crop irrigation and livestock is drilled within the concrete debris field as part of the Mass Balance water transport model.

The RESRAD code is designed to estimate doses from a contaminated zone above the water table. Because the conceptual model includes a contaminated zone that extends above and into the water table the following RESRAD parameters have been modified to develop a dose model consistent with the conceptual model of the site:

- the Mass Balance model (MB) used for water transport
- no unsaturated zones
- no dilution of groundwater by using a well pumping rate equal to 250 m^3/y (RESRAD default)

The basis for the parameters used to define the conceptual model are provided in Appendix 6L.

6.4.4.2.2 Tritium Model

For H-3, two separate conceptual models are developed based on more realistic site assumptions: primarily that the cellar hole area is potentially in contact with ground water and that the larger site area to be graded is above the water table. Two RESRAD dose models are applied to obtain separate H-3 DCGL values for each case.

The first model, described in Section 6.4.4.2.1, modifies the RESRAD parameters to reflect a contaminated zone within the saturated zone, in this case, the combined area of the cellar hole spaces. For the H-3 cellar hole scenario, all the other key elements discussed previously are maintained with the exception of the contamination fractions. RESRAD was allowed to calculate the fraction based on the smaller area of the cellar holes, because this small area cannot realistically support the production of all the food products (plant, meat, milk) used by the resident farmer.

The second model reflects the site grading scenario where the larger site grading area comprises the contaminated zone and is located above the water table. Key parameters for the H-3 site grading scenario that differ from the cellar hole scenario are as follows:

- the Nondispersion model (ND) is used for water transport
- one unsaturated zone
- well pumping rate value determined for the soil-resident farmer scenario

The basis for the parameters used to define the conceptual model are provided in Appendix 6L.

6.4.4.3 Parameter Value Assignment

The process described in Section 6.2.5 was used to determine the parameter input values or distributions. The values/distributions assigned to all parameters for the sensitivity analyses and the basis for assigning such values/distributions are summarized in Appendix 6M.

6.4.4.4 DCGL Determination

The input values assigned to sensitive and non-sensitive parameters for the DCGL runs were based on the process described in Section 6.2.5.5 in conjunction with the sensitivity analysis results presented in Appendix 6N. The DCGL determination was performed using RESRAD Version 6.21 analyses with the input values summarized in Appendix 6N.

The resulting DCFs are provided in Appendix 6O. The DCGLs, representing a dose of 25 mrem/yr, determined using Equation 6-1 are also provided in Appendix 6O.

6.5 Residual Radioactivity in Groundwater

LTP Section 5.6.3.2.4 requires that the concentration of well water available (based upon the well supply requirements assumed in Section 6 for the resident farmer) be below the EPA MCLs at the time of license termination. A calculation of the dose contribution from groundwater at the EPA MCLs was performed (Reference 6-15). This calculation used the approved groundwater DCGL from the Connecticut Yankee LTP for H-3 of 6.52E+05 pCi/l, representing a dose of 25 mrem/yr (Reference 6-16). The dose due to H-3 (the only plant-related radionuclide positively identified in groundwater) was determined to be 0.77 mrem/yr, when the concentration was at the EPA MCL for H-3 (20,000 pC/l).

6.6 Combining Dose Contributions from Different Media

YNPS considers the following media concurrently, when calculating the total dose from the site, in accordance with 10CFR20.1402:

- soils,
- subsurface partial structures,
- concrete debris, and
- groundwater.

The DCGLs for subsurface partial structures and groundwater represent a dose of 0.5 mrem/yr and 0.77 mrem/yr respectively. The sum of the dose contributions from subsurface partial structures and groundwater (1.27 mrem/yr) will be subtracted from the 25 mrem/yr total, leaving 23.73 mrem/yr for the dose contribution from soil and concrete debris.

DCGLs for soil and concrete debris, representing 23.73 mrem/yr, are provided in Table 6-1. In areas where soil and concrete debris used as backfill are present, the lower radionuclide-specific DCGL for the two media will be applied to soils and concrete debris. In areas where only soil is present (i.e., concrete debris backfill is not present), the soil radionuclide-specific DCGLs will be applied to soil.

Table 6-1

Summary of DCGLs for Different Media Types

Radionuclide	Soil (pCi/g)†	Building Surface (dpm/100 cm^2)‡	Subsurface Partial Structures (pCi/g)§	Concrete Debris† (pCi/g)
H-3	3.5E+02	3.4E+08	1.35E+02	9.5E+01 (cellar holes) 2.8E+02 (grading)
C-14	5.2E+00	1.0E+07	2.34E+03	7.2E+00
Fe-55	2.8E+04	4.0E+07	-	1.4E+02
Co-60	3.8E+00	1.8E+04	3.45E+03	4.3E+00
Ni-63	7.7E+02	3.7E+07	6.16E+04	1.0E+02
Sr-90	1.6E+00	1.4E+05	1.39E+01	7.6E-01
Nb-94	6.8E+00	2.6E+04	-	7.0E+00
Tc-99	1.3E+01	1.4E+07	-	6.1E+01
Ag-108m	6.9E+00	2.5E+04	-	7.0E+00
Sb-125	3.0E+01	1.0E+05	-	3.1E+01
Cs-134	4.7E+00	2.9E+04	-	4.7E+00
Cs-137	8.2E+00	6.3E+04	1..45E+03	6.7E+00
Eu-152	9.5E+00	3.7E+04	-	9.5E+00
Eu-154	9.0E+00	3.4E+04	-	9.1E+00
Eu-155	3.8E+02	6.5E+05	-	3.8E+02
Pu-238	3.1E+01	5.7E+03	-	9.5E+00
Pu-239	2.8E+01	5.1E+03	-	8.8E+00
Pu-241	9.3E+02	2.5E+05	-	1.4E+02
Am-241	2.8E+01	5.0E+03	-	4.1E+00
Cm-243	3.0E+01	7.2E+03	-	4.7E+00

† Represents a dose of 23.73 mrem/yr
‡ Represents a dose of 25 mrem/yr
§ Represents a dose of 0.5 mrem/yr, radionuclides based upon those found in concrete samples as discussed in Reference 6-11

6.7 Application of Decay

Because of the presence of spent fuel on site and the delay in availability of a central repository, portions of the YNPS site must remain licensed by the NRC well after decommissioning is complete. These portions include the ISFSI and areas surrounding the ISFSI. It is anticipated that fuel will remain onsite at YNPS until approximately 2022.

For this reason, YAEC intends to account for the reduction in dose due to decay for those areas of the site that are being final status surveyed, well in advance of their release from the NRC license (i.e., the industrial area). The DCGLs provided herein will be adjusted (using the half-life information in Table 2-6), such that the dose at the time of anticipated release of the area from the license is no greater than 23.73 mrem/yr, as discussed above. H-3 will not be decay adjusted, as its movement through soil/concrete into groundwater is likely more rapid than its decay, and, thus, would have the potential to contribute an excessive groundwater dose. The mobility of the other radionuclides is retarded such that decay would occur before their movement through soil/concrete into groundwater. Thus, adjustment due to decay will be performed for all radionuclides, with the exception of H-3.

6.8 Calculation of Area Factors

Area factors are required for both soil DCGLs and building surface DCGLs. First, the total doses from all pathways are calculated for each radionuclide and for each area of contamination. Doses relative to the base case contaminated area are then calculated. Finally, area factors are calculated for each radionuclide, which are the reciprocals of the relative doses.

6.8.1 Calculation of Area Factors for the Soils

Area factors for the resident farmer are calculated using the RESRAD 6.21 computer code with input parameters used in the original soils analysis and a unit activity of 1 pCi/g. As the area decreases, the set of ingestion pathway input parameters referred to as Contamination Fractions also decreases, using the equation in Reference 6-10. A Contamination Fraction indicates the fraction of a person's total diet that is obtained from the contaminated area. As the contaminated area decreases below a certain size, it is reasonable to assume that the fraction of the person's total diet from the contaminated area will also decrease proportionately. The RESRAD Contamination Fractions are listed below:

- Fraction of Drinking Water from the Site (FDW)
- Fraction of Household Water from the Site (FHHW)
- Fraction of Livestock Water from the Site (FLW)
- Fraction of Irrigation Water from the Site (FIRW)
- Fraction of Aquatic Food from the Site (FR9)
- Fraction of Plant Food from the Site (FPLANT)
- Fraction of Meat from the Site (FMEAT)
- Fraction of Milk from the Site (FMILK)

Equation D.5 of the RESRAD User's Manual varies the Contamination Fraction for plant food as follows:

$FA = A/2000$, where $0 \leq A \leq 1000\ m^2$
$FA = 0.5$, where $A > 1000\ m^2$

Since the $DCGL_w$s were calculated using a conservative value for FA of 1.0, Equation D.5 is multiplied by a factor of 2.0 to yield the contamination fraction of 1.0 at an area of 1000 m^2 (or larger) for plants. Values of the multiplier are listed in Appendix 6P as a function of the size of the contaminated zone. The same values are conservatively assigned to the contaminated fractions for drinking water, livestock water, irrigation water, and aquatic food.

The values for meat and milk are smaller and are derived below:

$FA = A/20{,}000\ m^2$, where $0 \leq A \leq 20{,}000\ m^2$
$FA = 1$, where $A > 20{,}000\ m^2$

Since the $DCGL_w$s were calculated using a conservative value for FA of 1.0, Equation D.5 is adjusted upward by applying the ratio of 20,000 m^2/13022 m^2 (the area assumed for the contaminated area in the soils analyses) or 1.54. Values are listed in Appendix 6P as a function of the area of the contaminated zone.

The fraction of household water remains set at 1.0 for all sizes of contaminated zones, which is consistent with the RESRAD code input screen that does not allow deviation from the default value of 1.0.

The total doses corresponding to the various areas of the contaminated zone are calculated using the input parameter values listed in Appendix 6P. Appendix 6Q summarizes the total dose by radionuclide and area.

6.8.2 Calculation of Area Factors for the Building Surfaces

For the building occupancy scenario, a somewhat different approach is used to compute the area factors used to establish the $DCGL_{EMC}$. While the $DCGL_w$ is the average concentration over the entire survey unit, the $DCGL_{EMC}$ should reflect the exposure an occupant would receive from an area of elevated activity having dimensions that are much smaller than the total interior area of the room. The total surface area of contaminated sources for the base case is 82.03 m^2, which includes the floor and four walls. For areas that are comparable to that for the room as a whole, evaluation against the $DCGL_w$ is appropriate.

The total doses for various areas of the contaminated source are calculated using RESRAD-BUILD. The model used in RESRAD-BUILD is similar to that used in the model for calculating building occupancy $DCGL_w$s. However, only one source is modeled herein, instead of the five sources considered in calculating the building occupancy $DCGL_w$s. The receptor is located at the source midpoint at a distance of 1 m away. All other input parameters are the same as in the building occupancy $DCGL_w$ calculation and are presented in Appendix 6R.

Appendix 6S presents the radionuclide-specific area factors.

6.9 References

6-1 Code of Federal Regulations, Title 10, Section 20.1402, "Radiological Criteria for Unrestricted Uses."

6-2 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," dated December 1997.

6-3 NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination," dated July 1998.

6-4 NUREG/CR-5512, "Residual Radioactivity from Contamination"
Volume 1: "Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," dated October 1992.
Volume 2: "User's Manual DandD Version 2.1;" dated April 2001
Volume 3: "Parameter Analysis, Draft Report for Comment," dated October 1999.

6-5 NUREG-1757, "Consolidated NMSS Decommissioning Guidance," dated September 2003.

6-6 NUREG/CR-6676, "Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes," dated May 2000.

6-7 NUREG/CR-6692, "Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes," dated November 2000.

6-8 NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes, dated November 2000.

6-9 NUREG/CR-6755, "Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code," February 2002.

6-10 ANL/EAD-4, "Users Manual for RESRAD Version 6.0," Yu, C. et al., dated July 2001.

6-11 YAEC Internal Communication, J. Darman to AP-0831 File, "IX-Pit Sample Plan Close-Out," January 7, 2004 (also filed as RP 04-008., Darman to Heath, dated March 4, 2004).

6-12 Sullivan, T.M., "DUST-MS - Disposal Unit Source Term- Multiple Species: Data Input Guide.", Brookhaven National Laboratory, 1997

6-13 YA-CALC-00-003-04, "Assessment of radionuclide release from contaminated concrete at the Yankee Rowe Nuclear Power Plant, dated August 2004.

6-14 YA-CALC-00-002-04, "RESRAD 6.21 Sensitivity Analysis and Derived Concentration Guideline Levels (DCGLs) for Concrete Debris," dated August 2004.

6-15 YA-REPT-00-003-04, "Estimate of Dose Due to Tritium in Groundwater at EPA's Maximum Contaminant Levels," dated April 2004.

6-15 Haddam Neck Plant License Termination Plan, Rev. 1a, dated October 2002.

Appendix 6K

DCGLs for Subsurface Partial Structures

Table 6K-1

Peak Dose for Initial Concentrations of 1 pCi/g

with Assumed Clean Concrete Backfill

Radionuclide	Dose (mrem/yr)
H-3	3.70E-03
C-14	2.14E-04
Co-60	1.45E-04
Ni-63	8.12E-06
Sr-90	3.60E-02
Cs-137	3.46E-04

Table 6K-2

DCGLs for Partially Intact Structures

Representing 0.5 mrem/yr Dose

Radionuclide	DCGL (pCi/g)
H-3	1.35E+02
C-14	2.34E+03
Co-60	3.45E+03
Ni-63	6.16E+04
Sr-90	1.39E+01
Cs-137	1.45E+03

Appendix 6L
Parameters Used to Quantify Conceptual Model

A. Buildings identified as potentially having subsurface spaces at the completion of the DEMCO Phase 1 Demolition Plan and/or the email communication with J. Lynch [5]

Table 1-1 Vertical Extension of Remaining Below-Grade Structures

Building	YR drawing reference	Wall elevations msl, ft (wrt plant grade)	Vertical Extension of Structure, meters (wrt plant grade)	Area m^2
PAB TK-30	PAB 9699-FM-57A	1022'8"-1004'2" = 18'6"	5.6	18
PAB, TK-27	PAB 9699-FM-57A	1022'8"-1004'2" = 18'6"	5.6	14.6
Spent Fuel Pool	Fuel Pit 9699-FC-45B	1022'8"-1008'0" = 14'8"	4.5	51.6
Waste Vault	PAB 9699-FC-43A	1020'6"-1010'8" = 9'10"	3.0	11.7
Elevator Pit	PAB 9699-FC-43A	1022'8"-1016'2" = 6'6"	1.9	6.5
IX Pit	PAB 9699-FC-40A, 40K,40L	1022' 8' - 1012' 6" = 6' 6"	3.1	67.5

B. Reference 5: Correspondence between J. Lynch and P. Littlefield, "RE. Concrete Debris," August 4, 2004

----- Original Message -----
From: Joe Lynch
To: 'Pete Littlefield'
Sent: Monday, July 12, 2004 10:38 AM
Subject: RE: Concrete Debris

Pete:

I sent you the Site Grading Plan under a separate message.

To address your questions, the building the subject of fill are the PAB (south wall towards the VC), the Fuel Pool excavation and the Ion Exchnage Pit excavation.

Concrete debris will be 8" in size or less.....uniformly distributed.

The majority of the fill will be used in the area extending from the southern end of the diesel generator building north to the northern end of the turbine building. In the east-west direction the fill zone would be from the east edge of the diesel generator/fuel storage building to the west edge of that building. This area is approximately 300 feet in the north-south direction and 180 feet in the east-west direction. The fill area will be approximately triangular in cross-section and will vary from 10 feet deep at the southern edge to approximately zero depth at the northern end (an average of 5 feet of depth). As a volume calculation this would equate to 300x180x5/27=10,000cy. This is an approximate number at this stage, but there is some science behind it. The fill area could potentially extend easterly along the ledge cut line approximately 200 feet. However, if we can dispose of the entire volume of ABC fill within the area described above, it may be better to keep it confined to a smaller footprint.

If you need any further information or clarification please let me know.

Regards,

Joe

I have listed the contact inforamtion for the designers of the Site Grading Plan if you have more questions or need clarification.

Kevin Cooley, P.E.
Civil Engineer
Kleinschmidt
Energy & Water Resource Consultants
75 Main St.
Pittsfield, ME 04967
Phone: (207) 487-3328
Fax: (207) 487-3124
Kevin.Cooley@KleinschmidtUSA.com
www.KleinschmidtUSA.com

C. Telecon: Joseph Lynch and Peter Littlefield, July 15, 2004, regarding "Preliminary Estimate of Concrete and Soil Borrow and Fill Volumes."

Preliminary Estimates of Concrete and Soil Borrow and Fill Volumes
Yankee Nuclear Power Station
Rowe, MA

WORKING DRAFT - FOR DISCUSSION PURPOSES ONLY

Category	Type	Source	Compacted Volume	Subtotal	Total	Source	Comments
			(cubic yards)				
Borrow	Concrete	Structures to Grade	13,105	17,705	69,905	1	
		Structures 0 - 18 inches below grade	4,600			2	Not all may need to be removed if greater than 3 feet of fill planned.
	Pavement	Not quantified.	1,200	1,200			
	Soil	ISFSI soils on SCTA	10,000	50,900		3	
		ISFSI soils in mid-parking lot	11,200			4	
		SCTA Removal	29,200			5	
		Detention Basin Excavation	900			5	
Fill	ABC	Building Voids	4,800	18,300	59,850	1	Based on information provided by Yankee.
		Shaping material (<3 feet) for H&A Plan	10,800	[illegible]		5	
		Shallow Foundations	2,700			2	Assumes all structures removed to 18 inches below grade.
	ABC?	Screenwell Foundation	3,400	10,500		1	Use of ABC will require USGen approval
		Circulating Water Pipes	600			1	Method of placing ABC will need to be evaluated.
		SCTA Below 3 feet	6,500			5	Does YAEC want to place ABC in SCTA?
	Soil	Service Building Foundation	350	24,050		2	
		Cap for H&A Plan	18,700			5	
		SCTA Upper 2 feet	5,000			5	
	Engineered Soils	Dam Extension	6,000	6,000		5	

Notes:
Volume estimates are preliminary and for discussion purposes only - not intended for contracting or design purposes.
Assumes 3 feet of soil will be required over ABC fill or structures left in-place
Volume of pavement has not been quantified
Assumes that no net fill or borrow for shoreline activities.
Assumes volume of soil in borrow area will be same as in fill area (fluff factor would be negligible).
Assumes soil on-site is suitable for use as topsoil.

Sources:
1 - Yankee Waste Optimization Estimates
2 - Concrete volume estimates prepared by Joe McCormack, April 2004
3 - Volume estimate reported in SCTA CSA
4 - Estimate prepared by Ken Dow
5 - ERM preliminary volume estimates

ABC - asphalt, brick, concrete 6" or less

D. The calculation of the plant transfer factor (ptf) for concrete is based on the correlation of the Kd and the root uptake factor (CR) defined in Reference 12 Equation 3.9-2, as shown below

$$\mathrm{Ln(Kd)} = 4.62 + \mathrm{stex} - 0.56[\ln(\mathrm{CR})] \qquad \text{Equation 1}$$

Where:

Kd = distribution coefficient for concrete

stex = -2.52 for sand soil (coarsest medium in Reference 12 and site soil type)

CR = Root Uptake Transfer Factor (pCi/g plant per pCi/g medium) or the RESRAD soil/plant transfer coefficient (Reference 15, Section H, p. H-13).

Rearranging and solving equation 1 for CR results in the following equation to calculate CR for given values of Kd:

$$\ln(\mathrm{CR}) = \frac{\ln(\mathrm{Kd}) - 4.62 - (\mathrm{stex})}{-0.56}$$

$$\ln(\mathrm{CR}) = \frac{\ln(\mathrm{Kd})}{-0.56} + 3.75$$

$$\mathrm{CR} = 42.52\ (\mathrm{EXP}(\ln(\mathrm{Kd})/-0.56)) \qquad \text{Equation 2}$$

Specifically:

a. A Uniform Distribution is assigned to Ag, Cm, Co, Cs, Fe, Ni, Sr and Tc. The minimum and maximum Kd values are substituted into Equation 2.

b. A Loguniform Distribution is assigned to Ac, Am, C, Eu, Gd, Nb, Np, Pa, Pu and Th. The minimum and maximum Kd values are substituted into Equation 2.

c. A Lognormal Distribution is assigned to Pb, Sb, and U. The mean and standard deviation of the lognormal distribution were determined following the calculation of CR using equation 2 and the natural log transformation of CR.

d. A Truncated Lognormal Distribution from Reference 12 is assigned to H-3 and Ra-226 to allow stochastic treatment of this parameter for the sensitivity analysis.

E. Equilibrium Groundwater Concentration

RESRAD uses the linear relationship in Equation 3, taken from Reference 15, Section H, to estimate the ground water concentration resulting from concentrations in concrete (soil) particles.

$$S = \mathrm{Kd}^{*}C \qquad \text{Equation 3}$$

Equation 4 expresses the ground water concentration under equilibrium conditions in a saturated environment based on the relationships defined by Equation 3. This equation is used to compare the RESRAD well water concentration to the equilibrium ground water concentration.

$$C = \frac{1000\ S_{o}\rho_{b}}{\left[1 + (k_{d}\rho_{b}/n)\right] n} \qquad \text{Equation 4}$$

where:

C = Equilibrium groundwater concentration (pCi/L)
So = Initial principal radionuclide concentration in the concrete (pCi/gm)
ρ_b = Bulk density of the contaminated zone (gm/cm^3)
Kd = Distribution coefficient of the contaminated zone (cm^3/gm)
n = Total porosity of the contaminated zone
1000 cm^3 per liter

Appendix 6M

Table 6M-1 - Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Table 6M-2 - Input Parameters for Sensitivity Analysis, H-3 Graded Concrete Debris

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Soil Concentrations										
Basic radiation dose limit (mrem/yr)		3	D	25	10 CFR 20.1402 [1]	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR	
Distribution Coefficient										
Ac-227+D	P	1	S	Loguniform	Chemical analogy to Am [3]	200	5000	NR	NR	1.00E+03
Ag-108m	P	1	S	Uniform	Chemical analogy to Cu [3]	3000	10000	NR	NR	6.5E+03
Am-241	P	1	S	Loguniform	[3]	200	5000	NR	NR	1.00E+03
Am-243+D	P	1	S	Loguniform	[3]	200	5000	NR	NR	1.00E+03
C-14	P	1	S	Loguniform	[3]	10	500	NR	NR	7.07E+01
Cm-243	P	1	S	Uniform	[3]	200	1000	NR	NR	6.00E+02
Co-60	P	1	S	Uniform	[3]	181	383	NR	NR	2.82E+02
Cs-134	P	1	S	Uniform	[3]	34	240	NR	NR	1.37E+02
Cs-137+D	P	1	S	Uniform	[3]	34	240	NR	NR	1.37E+02
Eu-152	P	1	S	Loguniform	Chemical analogy to Am [3]	200	5000	NR	NR	1.00E+03
Eu-154	P	1	S	Loguniform	Chemical analogy to Am [3]	200	5000	NR	NR	1.00E+03
Eu-155	P	1	S	Loguniform	Chemical analogy to Am [3]	200	5000	NR	NR	1.00E+03
Fe-55	P	1	S	Uniform	[3]	7	18	NR	NR	1.25E+01
Gd-152	P	1	S	Loguniform	Chemical analogy to Am [3]	200	5000	NR	NR	1.00E+03
H-3	P	1	D	0.00	[3]			NR	NR	
Nb-94	P	1	S	Loguniform	[3]	100	1000	NR	NR	3.16E+02
Ni-63	P	1	S	Uniform	[3]	10	61	NR	NR	3.55E+01
Np-237+D	P	1	S	Loguniform	[3]	100	5000	NR	NR	7.07E+02

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Pa-231	P	1	S	Loguniform	Chemical analogy to Nb [3]	100	1000	NR	NR	3.16E+02
Pb-210+D	P	1	S	Lognormal-n	[3]	10.77	0.88	NR	NR	4.76E+04
Pu-238	P	1	S	Loguniform	[3]	500	5000	NR	NR	1.58E+03
Pu-239	P	1	S	Loguniform	[3]	500	5000	NR	NR	1.58E+03
Pu-241+D	P	1	S	Loguniform	[3]	500	5000	NR	NR	1.58E+03
Ra-226+D	P	1	D	100	[3]			NR	NR	
Sb-125	P	1	S	Lognormal-n	[3]	7.35	1.11	NR	NR	1.55E+03
Sr-90+D	P	1	S	Uniform	[3]	10	11	NR	NR	1.05E+01
Tc-99	P	1	S	Uniform	[3]	6	21	NR	NR	1.35E+01
Th-229+D	P	1	S	Loguniform	[3]	500	5000	NR	NR	1.58E+03
Th-230	P	1	S	Loguniform	[3]	500	5000	NR	NR	1.58E+03
U-233	P	1	S	Lognormal-n	[3]	4.99	2.37	NR	NR	1.47E+02
U-234	P	1	S	Lognormal-n	[3]	4.99	2.37	NR	NR	1.47E+02
U-235+D	P	1	S	Lognormal-n	[3]	4.99	2.37	NR	NR	1.47E+02
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	Ground water uncontaminated	NR	NR	NR	NR	
Calculation Times										
Time since placement of material (yr)	P	3	D	0		NR	NR	NR	NR	
Time for calculations (yr)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD Default	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Contaminate Zone										
Area of contaminated zone (m**2)	P	2	D	5020 170	Area of site to be graded with concrete [5] Combined area of the cellar holes used for H-3	NR	NR	NR	NR	
Thickness of contaminated zone (m)	P	2	D	3.8	Corresponds to maximum depth to groundwater [6]					
Length parallel to aquifer flow (m)	P	2	D	80 14.7	Length corresponds to area of 5020m^2 Based on area of cellar holes used for H-3	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	0	NUREG-1757 Intruder Scenario conservative assumption that required MA State DEP cover is removed [7]	NR	NR	NR	NR	
Density of Cover material (g/cm^3)	P	1	S	NA	No cover					
Cover erosion rate (m/yr)	P	2	D	NA	No cover					
Density of contaminated zone (g/cm^3)	P	1	S	Uniform	Distribution derived using total porosity range for coarse gravel [4] & concrete particle density of 2.2 g/cm^3 [4, equation 2.3 p 16]	1.41	1.67	NR	NR	1.54
Contaminated zone erosion rate (m/yr)	P	2	D	8.5E-04	Calculated value based on site-specific slope of 2.9% [8]	NR	NR	NR	NR	
Contaminated zone total porosity	P	2	S	Uniform	Range for coarse gravel [4]	0.24	0.36	NR	NR	0.3

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Contaminated zone field capacity	P	3	D	0.07	Calculated using Equation 4.4 [4] and arithmetic means for SZ total and effective porosity [8]	NR	NR	NR	NR	
Contaminated zone hydraulic conductivity (m/yr)	P	2	S	Loguniform	Range for gravel [8]	1.E+04	1.E+07	NR	NR	3.16E05
Contaminated zone b parameter	P	2	S	Bounded Lognormal n	NUREG 6697 dist for site soil type - sand [2] Coarsest media listed	- 0.0253	0.216	0.501	1.90	0.975
Humidity in air (g/m**3)	P	3	D	6.1	Regional value [8]	NR	NR	NR	NR	
Evapotranspiration coefficient	P	2	S	Uniform	NUREG/CR-6697 Att. C [2]	0.5	0.75	NR	NR	0.625
Average annual wind speed (m/sec)	P	2	D	2.03	Site-specific value calc. from site meteorological data [8]	NR	NR	NR	NR	
Precipitation (m/yr)	P	2	D	1.2	Site-specific value calculated from site geographical area ppt. [8]	NR	NR	NR	NR	
Irrigation (m/yr)	B	3	S	Uniform	NUREG/CR-6697, Att C methodology [2, 8]	0.252	0.618	NR	NR	0.435
Irrigation mode	B	3	D	Overhead	Site-specific - overhead vs. ditch irrigation is standard practice in Eastern U. S.	NR	NR	NR	NR	
Runoff coefficient	P	2	D	0.6	NUREG/CR-6697, Att. C section 4.2 methodology [2, 8]	NR	NR	NR	NR	
Watershed area for nearby stream or pond (m**2)	P	3	D	7.77E+05	Site-specific- drainage area [8]	NR	NR	NR	NR	
Accuracy for water/soil computations	-	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Saturated Zone Hydrological Data										
Density of saturated zone (g/cm^3)	P	1	D	1.54	Value derived using total porosity range for coarse gravel [4] & concrete particle density of 2.2 g/cm3 [4, Eqn 2.3 p 16]	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Saturated zone total porosity	P	1	D	0.28	Arithmetic mean for coarse gravel [4, Section3]	NR	NR	NR	NR	
Saturated zone effective porosity	P	1	D	0.21	Arithmetic mean for coarse gravel [4, Section 3]I	NR	NR	NR	NR	
Saturated zone field capacity	P	3	D	0.07	Calculated using equation 4.4 and porosity values from [4	NR	NR	NR	NR	
Saturated zone hydraulic conductivity (m/yr)	P	1	D	3.16E5	Median value for gravel [4]	NR	NR	NR	NR	
Saturated zone hydraulic gradient	P	2	D	0.1	Site gradient [8]	NR	NR	NR	NR	
Saturated zone b parameter	P	2	D	0.975	Median from NUREG-6697 distrbution for sand [2]	NR	NR	NR	NR	
Water table drop rate (m/yr)	P	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Well pump intake depth (m below water table)	P	2	D	10	RESRAD Default (not used with MB model)	NR	NR	NR	NR	
Model: Nondispersion (ND) or Mass-Balance (MB)	P	3	D	MB	MB model selected to minimize dilution in saturated zone	NR	NR	NR	NR	
Well pumping rate (m**3/yr)	P	2	D	250 50	RESRAD Default selected to ensure no dilution in saturated zone in MB model Assures no dilution in saturated zone in MB model for H-3	NR	NR	NR	NR	
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	3	D	0	Contaminated zone extends below the water table	NR	NR	NR	NR	
Occupancy										
Inhalation rate (m**3/yr)	B	3	D	8400	NUREG/CR-6697, Att C [2]	NR	NR	NR	NR	
Mass loading for inhalation (g/m**3)	P	2	S	Continuous linear	NUREG/CR-6697, Att. C [2]					2.33E-05
Exposure duration	B	3	D	30	RESRAD Default	NR	NR	NR	NR	
Indoor dust filtration factor	P	2	S	Uniform	NUREG/CR-6697, Att. C [2]	0.15	0.95	NR	NR	0.55

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Shielding factor, external gamma	P	2	S	Bounded lognormal-n	NUREG/CR-6697, Att. C [2]	-1.3	0.59	0.044	1	0.2725
Fraction of time spent indoors	B	3	D	0.6571	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Fraction of time spent outdoors (on site)	B	3	D	0.1181	NUREG/CR-5512, Vol. 3 Table 6.87 (outdoors + gardening) [9]	NR	NR	NR	NR	
Shape factor flag, external gamma	P	3	D	Circular	RESRAD Default - Circular contaminated zone assumed	NR	NR	NR	NR	
Ingestion, Dietary										
Fruits, vegetables, grain consumption (kg/yr)	B	2	D	112	NUREG/CR-5512, Vol. 3 (other vegetables + fruits + grain) [9]	NR	NR	NR	NR	
Leafy vegetable consumption (kg/yr)	B	3	D	21.4	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	
Milk consumption (L/yr)	B	2	D	233	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	
Meat and poultry consumption (kg/yr)	B	3	D	65.1	NUREG/CR5512, Vol. 3 (beef + poultry) [9]	NR	NR	NR	NR	
Fish consumption (kg/yr)	B	3	D	20.6	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	
Other seafood consumption (kg/yr)	B	3	D	0.9	RESRAD Default	NR	NR	NR	NR	
Soil ingestion rate (g/yr)	B	2	D	18.26	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	
Drinking water intake (L/yr)	B	2	D	478.5	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Contamination fraction of drinking water	P	3	D	1	RESRAD Default - all water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of household water (if used)	P	3		NA						
Contamination fraction of livestock water	P	3	D	1	RESRAD Default - all water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of irrigation water	P	3	D	1	RESRAD Default - all water assumed contaminate	NR	NR	NR	NR	
Contamination fraction of aquatic food	P	2	D	1	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Contamination fraction of plant food	P	3	D	1 -1	Used w/ NUREG/CR-5512, Vol. 3 [9] regional homegrown consumption rate RESRAD calculates fraction based on cellar hole area for H-3	NR	NR	NR	NR	
Contamination fraction of meat	P	3	D	1 -1	Used w/ NUREG/CR-5512, Vol. 3 [9] regional homegrown consumption rate RESRAD calculates fraction based on cellar hole area for H-3	NR	NR	NR	NR	
Contamination fraction of milk	P	3	D	1 -1	Used w/ NUREG/CR-5512, Vol. 3 [9] regional homegrown consumption rate RESRAD calculates fraction based on cellar hole area for H-3	NR	NR	NR	NR	
Ingestion, Non-dietary										
Livestock fodder intake for meat (kg/day)	M	3	D	27.1	NUREG/CR5512, Vol. 3 Table 6.87, beef cattle + poultry + layer hen [9]	NR	NR	NR	NR	
Livestock fodder intake for milk (kg/day)	M	3	D	63.2	NUREG/CR5512, Vol. 3 Table 6.87, forage + grain + hay [9]	NR	NR	NR	NR	
Livestock water intake for meat (L/day)	M	3	D	50.6	NUREG/CR5512, Vol. 3 Table 6.87, beef cattle + poultry + layer hen [9]	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Livestock soil intake (kg/day)	M	3	D	0.5	RESRAD Default	NR	NR	NR	NR	
Mass loading for foliar deposition (g/m**3)	P	3	D	4.00E-04	NUREG/CR-5512, Vol. 3 Table 6.87, gardening [9]	NR	NR	NR	NR	
Depth of soil mixing layer (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C [2]	0	0.15	0.6	NR	0.23
Depth of roots (m)	P	1	S	Uniform	Min. from NUREG/CR-6697, Att. C [2] Max. is site specific depth to water table [6]	0.3	3.8	NR	NR	2.05
Drinking water fraction from ground water	P	3	D	1	RESRAD Default - all water assumed to be supplied from groundwater	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Household water fraction from ground water (if used)	P	3		NA						
Livestock water fraction from ground water	P	3	D	1	RESRAD Default - all water assumed to be supplied from groundwater	NR	NR	NR	NR	
Irrigation fraction from ground water	P	3	D	1	RESRAD Default - all water assumed to be supplied from groundwater	NR	NR	NR	NR	
Wet weight crop yield for Non-Leafy (kg/m**2)	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	0.56	0.48	0.001	0.999	1.75
Wet weight crop yield for Leafy (kg/m**2)	P	3	D	2.88921	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Wet weight crop yield for Fodder (kg/m**2)	P	3	D	1.8868	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Growing Season for Non-Leafy (years)	P	3	D	0.246	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Growing Season for Leafy (years)	P	3	D	0.123	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Growing Season for Fodder (years)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/yr)	P	2	S	Triangular	NUREG/CR-6697, Att. C [2]	5.1	18	84	NR	33
Wet Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	S	Triangular	NUREG/CR-6697, Att. C [9]	0.06	0.67	0.95	NR	0.58
Wet Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 [9]	NR	NR	NR	NR	
Storage Times of contaminated Foodstuffs (days)										
Fruits, non-leafy vegetables, and grain	B	3	D	14	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 [9]	NR	NR	NR	NR	
Meat and poultry	B	3	D	20	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef) [9]	NR	NR	NR	NR	
Fish	B	3	D	7	RESRAD Default	NR	NR	NR	NR	
Crustacea and mollusks	B	3	D	7	RESRAD Default	NR	NR	NR	NR	
Well water	B	3	D	1	RESRAD Default	NR	NR	NR	NR	
Surface water	B	3	D	1	RESRAD Default	NR	NR	NR	NR	
Livestock fodder	B	3	D	45	RESRAD Default	NR	NR	NR	NR	
Special Radionuclides (C-14)										
C-12 concentration in water (g/cm^3)	P	3	D	2.00E-05	RESRAD Default	NR	NR	NR	NR	
C-12 concentration in contaminated soil (g/g)	P	3	D	3.00E-02	RESRAD Default	NR	NR	NR	NR	
Fraction of vegetation carbon from soil	P	3	D	2.00E-02	RESRAD Default	NR	NR	NR	NR	
Fraction of vegetation carbon from air	P	3	D	9.80E-01	RESRAD Default	NR	NR	NR	NR	
C-14 evasion layer thickness in soil (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C [2]	0.2	0.3	0.6	NR	0.3

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
C-14 evasion flux rate from soil (1/sec)	P	3	D	7.00E-07	RESRAD Default	NR	NR	NR	NR	
C-12 evasion flux rate from soil (1/sec)	P	3	D	1.00E-10	RESRAD Default	NR	NR	NR	NR	
Fraction of grain in beef cattle feed	B	3	D	0.2500	NUREG/CR-6697, Att. B [2]	NR	NR	NR	NR	
Fraction of grain in milk cow feed	B	3	D	0.1000	NUREG/CR-6697, Att. B [2]	NR	NR	NR	NR	
Dose Conversion Factors (Inhalation mrem/pCi)										
Ac-227+D	M	3	D	6.72E+00	FGR11 (RESRAD Dose Conversion Library)	NR	NR	NR	NR	
Ag-108m	M	3	D	2.83E-04	FGR11	NR	NR	NR	NR	
Am-241	M	3	D	4.44E-01	FGR11	NR	NR	NR	NR	
Am-243+D	M	3	D	4.40E-01	FGR11	NR	NR	NR	NR	
C-14	M	3	D	2.09E-06	FGR11	NR	NR	NR	NR	
Cm-243	M	3	D	3.07E-01	FGR11	NR	NR	NR	NR	
Co-60	M	3	D	2.19E-04	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	4.63E-05	FGR11	NR	NR	NR	NR	
Cs-137+D	M	3	D	3.19E-05	FGR11	NR	NR	NR	NR	
Eu-152	M	3	D	2.21E-04	FGR11	NR	NR	NR	NR	
Eu-154	M	3	D	2.86E-04	FGR11	NR	NR	NR	NR	
Eu-155	M	3	D	4.14E-05	FGR11	NR	NR	NR	NR	
Fe-55	M	3	D	2.69E-06	FGR11	NR	NR	NR	NR	
Gd-152	M	3	D	2.43E-01	FGR11	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR11	NR	NR	NR	NR	
Nb-94	M	3	D	4.14E-04	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	6.29E-06	FGR11	NR	NR	NR	NR	
Np-237+D	M	3	D	5.40E-01	FGR11	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Pa-231	M	3	D	1.28E+00	FGR11	NR	NR	NR	NR	
Pb-210+D	M	3	D	1.38E-02	FGR11	NR	NR	NR	NR	
Pu-238	M	3	D	3.92E-01	FGR11	NR	NR	NR	NR	
Pu-239	M	3	D	4.29E-01	FGR11	NR	NR	NR	NR	
Pu-241+D	M	3	D	8.25E-03	FGR11	NR	NR	NR	NR	
Ra-226+D	M	3	D	8.60E-03	FGR11	NR	NR	NR	NR	
Sb-125	M	3	D	1.22E-05	FGR11	NR	NR	NR	NR	
Sr-90+D	M	3	D	1.31E-03	FGR11	NR	NR	NR	NR	
Tc-99	M	3	D	8.33E-06	FGR11	NR	NR	NR	NR	
Th-229+D	M	3	D	2.16E+00	FGR11	NR	NR	NR	NR	
Th-230	M	3	D	3.26E-01	FGR11	NR	NR	NR	NR	
U-233	M	3	D	1.35E-01	FGR11	NR	NR	NR	NR	
U-234	M	3	D	1.32E-01	FGR11	NR	NR	NR	NR	
U-235+D	M	3	D	1.23E-01	FGR11	NR	NR	NR	NR	
Dose Conversion Factors (Ingestion mrem/pCi)										
Ac-227+D	M	3	D	1.48E-02	FGR11 (RESRAD Dose Conversion Library)	NR	NR	NR	NR	
Ag-108m	M	3	D	7.62E-06	FGR11	NR	NR	NR	NR	
Am-241	M	3	D	3.64E-03	FGR11	NR	NR	NR	NR	
Am-243+D	M	3	D	3.63E-03	FGR11	NR	NR	NR	NR	
C-14	M	3	D	2.09E-06	FGR11	NR	NR	NR	NR	
Cm-243	M	3	D	2.51E-03	FGR11	NR	NR	NR	NR	
Co-60	M	3	D	2.69E-05	FGR11	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR11	NR	NR	NR	NR	
Cs-137+D	M	3	D	5.00E-05	FGR11	NR	NR	NR	NR	
Eu-152	M	3	D	6.48E-06	FGR11	NR	NR	NR	NR	
Eu-154	M	3	D	9.55E-06	FGR11	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Eu-155	M	3	D	1.53E-06	FGR11	NR	NR	NR	NR	
Fe-55	M	3	D	6.07E-07	FGR11	NR	NR	NR	NR	
Gd-152	M	3	D	1.61E-04	FGR11	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR11	NR	NR	NR	NR	
Nb-94	M	3	D	7.14E-06	FGR11	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR11	NR	NR	NR	NR	
Np-237+D	M	3	D	4.44E-03	FGR11	NR	NR	NR	NR	
Pa-231	M	3	D	1.06E-02	FGR11	NR	NR	NR	NR	
Pb-210+D	M	3	D	5.37E-03	FGR11	NR	NR	NR	NR	
Pu-238	M	3	D	3.20E-03	FGR11	NR	NR	NR	NR	
Pu-239	M	3	D	3.54E-03	FGR11	NR	NR	NR	NR	
Pu-241+D	M	3	D	6.85E-05	FGR11	NR	NR	NR	NR	
Ra-226+D	M	3	D	1.33E-03	FGR11	NR	NR	NR	NR	
Sb-125	M	3	D	2.81E-06	FGR11	NR	NR	NR	NR	
Sr-90+D	M	3	D	1.53E-04	FGR11	NR	NR	NR	NR	
Tc-99	M	3	D	1.46E-06	FGR11	NR	NR	NR	NR	
Th-229+D	M	3	D	4.03E-03	FGR11	NR	NR	NR	NR	
Th-230	M	3	D	5.48E-04	FGR11	NR	NR	NR	NR	
U-233	M	3	D	2.89E-04	FGR11	NR	NR	NR	NR	
U-234	M	3	D	2.83E-04	FGR11	NR	NR	NR	NR	
U-235+D	M	3	D	2.67E-04	FGR11	NR	NR	NR	NR	
Plant Transfer Factors (pCi/g plant)/(pCi/g soil										
Ac-227+D	P	1	S	Loguniform	Chemical analogy to Am [3]	1.06E-05	3.31E-03	NR	NR	1.87E-04
Ag-108m	P	1	S	Uniform	Chemical analogy to Cu [3]	3.06E-06	2.63E-05	NR	NR	1.47E-05

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Am-241	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	3.31E-3	NR	NR	1.87E-04
Am-243+D	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	3.31E-3	NR	NR	1.87E-04
C-14	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	6.44E-04	6.96E-01	NR	NR	2.12E-02
Cm-243	P	1	S	Uniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.87E-04	3.31E-03	NR	NR	1.75E-03
Co-60	P	1	S	Uniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.04E-03	3.95E-03	NR	NR	2.50E-03
Cs-134	P	1	S	Uniform	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	2.39E-03	7.83E-02	NR	NR	4.03E-02
Cs-137+D	P	1	S	Uniform	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	2.39E-03	7.83E-02	NR	NR	4.03E-02
Eu-152	P	1	S	Loguniform	Chemical analogy to Am Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	3.31E-03	NR	NR	1.87E-04
Eu-154	P	1	S	Loguniform	Chemical analogy to Am Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	3.31E-03	NR	NR	1.87E-04
Eu-155	P	1	S	Loguniform	Chemical analogy to Am Min and Max values calculated [3] and [2, Eqn 3.9-2]	1.06E-05	3.31E-03	NR	NR	1.87E-04
Fe-55	P	1	S	Uniform	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	2.44E-01	1.32E+00	NR	NR	7.80E-01
Gd-152	P	1	S	Loguniform	Chemical analogy to Am [3]	1.06E-05	3.31E-03	NR	NR	1.87E-04
H-3	P	1	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	1.57	1.1	0.001	0.999	4.8

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Nb-94	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.87E-04	1.14E-02	NR	NR	1.46E-03
Ni-63	P	1	S	Uniform	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	2.76E-02	6.96E-01	NR	NR	3.62E-01
Np-237+D	P	1	S	Loguniform	Min and Max values calculated [3] and [2, Eqn 3.9-2]	1.06E-05	1.14E-02	NR	NR	3.47E-04
Pa-231	P	1	S	Loguniform	Chemical analogy to Nb [3]	1.87E-04	1.14E-02	NR	NR	1.46E-03
Pb-210+D	P	1	S	Lognormal-n	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	-15.48	1.57	NR	NR	1.88E-07
Pu-238	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	6.44E-04	NR	NR	8.24E-05
Pu-239	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	6.44E-04	NR	NR	8.24E-05
Pu-241+D	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	6.44E-04	NR	NR	8.24E-05
Ra-226+D	P	1	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.22	0.9	0.001	0.999	4.0E-02
Sb-125	P	1	S	Lognormal-n	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	-9.37	1.98	NR	NR	8.50E-05
Sr-90+D	P	1	S	Uniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	5.87E-01	6.96E-01	NR	NR	6.42E-01
Tc-99	P	1	S	Uniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.85E-01	1.73E+00	NR	NR	9.60E-01
Th-229+D	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	6.44E-04	NR	NR	8.24E-05

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Th-230	P	1	S	Loguniform	Min and Max values calculated using [3] and [2, Eqn 3.9-2]	1.06E-05	6.44E-04	NR	NR	8.24E-05
U-233	P	1	S	Lognormal-n	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	-5.17	4.23	NR	NR	5.71E-03
U-234	P	1	S	Lognormal-n	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	-5.17	4.23	NR	NR	5.71E-03
U-235+D	P	1	S	Lognormal-n	Mean and Std Dev calculated using [3] and [2, Eqn 3.9-2]	-5.17	4.23	NR	NR	5.71E-03
Meat Transfer Factors (pCi/kg per pCi/d)										
Ac-227+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-10.82	1.0	0.001	0.999	2.0E-05
Ag-108m	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	0.7	0.001	0.999	2.0E-03
Am-241	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.90	0.2	0.001	0.999	5.0E-05
Am-243+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.90	0.2	0.001	0.999	5.0E-05
C-14	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.47	1.0	0.001	0.999	3.1E-02
Cm-243	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-10.82	1.0	0.001	0.999	2.0E-05
Co-60	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.51	1.0	0.001	0.999	3.0E-02
Cs-134	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.00	0.4	0.001	0.999	5.0E-02
Cs-137+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.00	0.4	0.001	0.999	5.0E-02
Eu-152	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	1.0	0.001	0.999	2.0E-03
Eu-154	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	1.0	0.001	0.999	2.0E-03

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Eu-155	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	1.0	0.001	0.999	2.0E-03
Fe-55	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.51	0.4	0.001	0.999	3.0E-02
Gd-152	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	1.0	0.001	0.999	2.0E-03
H-3	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-4.42	1.0	0.001	0.999	1.2E-02
Nb-94	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.82	0.9	0.001	0.999	1.0E-06
Ni-63	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-5.30	0.9	0.001	0.999	5.0E-03
Np-237+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.91	0.7	0.001	0.999	1.0E-03
Pa-231	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-12.21	1.0	0.001	0.999	5.0E-06
Pb-210+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.13	0.7	0.001	0.999	8.0E-04
Pu-238	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.21	0.2	0.001	0.999	1.0E-04
Pu-239	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.21	0.2	0.001	0.999	1.0E-04
Pu-241+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.21	0.2	0.001	0.999	1.0E-04
Ra-226+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.91	0.7	0.001	0.999	1.0E-03
Sb-125	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.91	0.9	0.001	0.999	1.0E-03
Sr-90+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-4.61	0.4	0.001	0.999	1.0E-02
Tc-99	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.21	0.7	0.001	0.999	1.0E-04

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Th-229+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.21	1.0	0.001	0.999	1.0E-04
Th-230	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.21	1.0	0.001	0.999	1.0E-04
U-233	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.13	0.7	0.001	0.999	8.0E-04
U-234	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.13	0.7	0.001	0.999	8.0E-04
U-235+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.13	0.7	0.001	0.999	8.0E-04
Milk Transfer Factors (pCi/L)/(pCi/d										
Ac-227+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.12	0.9	0.001	0.999	2.0E-06
Ag-108m	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-5.12	0.7	0.001	0.999	6.0E-03
Am-241	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.12	0.7	0.001	0.999	2.0E-06
Am-243+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.12	0.7	0.001	0.999	2.0E-06
C-14	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-4.4	0.9	0.001	0.999	1.2E-02
Cm-243	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.12	0.9	0.001	0.999	2.0E-06
Co-60	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	0.7	0.001	0.999	2.0E-03
Cs-134	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-4.61	0.5	0.001	0.999	1.0E-02
Cs-137+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-4.61	0.5	0.001	0.999	1.0E-02
Eu-152	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.72	0.9	0.001	0.999	6.0E-05

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Eu-154	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.72	0.9	0.001	0.999	6.0E-05
Eu-155	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.72	0.9	0.001	0.999	6.0E-05
Fe-55	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-8.11	0.7	0.001	0.999	3.0E-04
Gd-152	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.72	0.9	0.001	0.999	6.0E-05
H-3	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-4.6	0.9	0.001	0.999	1.0E-02
Nb-94	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.12	0.7	0.001	0.999	2.0E-06
Ni-63	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-3.91	0.7	0.001	0.999	2.0E-02
Np-237+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-11.51	0.7	0.001	0.999	1.0E-05
Pa-231	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-12.21	0.9	0.001	0.999	5.0E-06
Pb-210+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-8.11	0.9	0.001	0.999	3.0E-04
Pu-238	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.82	0.5	0.001	0.999	1.0E-06
Pu-239	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.82	0.5	0.001	0.999	1.0E-06
Pu-241+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-13.82	0.5	0.001	0.999	1.0E-06
Ra-226+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.91	0.5	0.001	0.999	1.0E-03
Sb-125	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-9.72	0.9	0.001	0.999	6.0E-05
Sr-90+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.21	0.5	0.001	0.999	2.0E-03

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Tc-99	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-6.91	0.7	0.001	0.999	1.0E-03
Th-229+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-12.21	0.9	0.001	0.999	5.0E-06
Th-230	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-12.21	0.9	0.001	0.999	5.0E-06
U-233	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.82	0.6	0.001	0.999	4.0E-04
U-234	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.82	0.6	0.001	0.999	4.0E-04
U-235+D	P	2	S	Truncated lognormal-n	NUREG/CR-6697, Att. C [2]	-7.82	0.6	0.001	0.999	4.0E-04
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/L))										
Ac-227+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	2.7	1.1	NR	NR	1.5E+01
Ag-108m	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	1.6	1.1	NR	NR	5.0E+00
Am-241	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
Am-243+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
C-14	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	10.8	1.1	NR	NR	4.9E+04
Cm-243	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
Co-60	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	5.7	1.1	NR	NR	3.0E+02
Cs-134	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	7.6	0.7	NR	NR	2.0E+03
Cs-137+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	7.6	0.7	NR	NR	2.0E+03
Eu-152	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.9	1.1	NR	NR	4.9E+01
Eu-154	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.9	1.1	NR	NR	4.9E+01
Eu-155	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.9	1.1	NR	NR	4.9E+01
Fe-55	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	5.3	1.1	NR	NR	2.0E+02
Gd-152	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.2	1.1	NR	NR	2.5E+01
H-3	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	0	0.1	NR	NR	1.0E+00
Nb-94	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	5.7	1.1	NR	NR	3.0E+02

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Ni-63	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	4.6	1.1	NR	NR	9.9E+01
Np-237+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
Pa-231	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	2.3	1.1	NR	NR	1.0E+01
Pb-210+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	5.7	1.1	NR	NR	3.0E+02
Pu-238	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
Pu-239	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
Pu-241+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.4	1.1	NR	NR	3.0E+01
Ra-226+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3.9	1.1	NR	NR	4.9E+01
Sb-125	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	4.6	1.1	NR	NR	9.9E+01
Sr-90+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	4.1	1.1	NR	NR	6.0E+01
Tc-99	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	3	1.1	NR	NR	2.0E+01
Th-229+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	4.6	1.1	NR	NR	9.9E+01
Th-230	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	4.6	1.1	NR	NR	9.9E+01
U-233	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	2.3	1.1	NR	NR	1.0E+01
U-234	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	2.3	1.1	NR	NR	1.0E+01
U-235+D	P	2	S	Lognormal-n	NUREG/CR-6697, Att. C [2]	2.3	1.1	NR	NR	1.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/L))										
Ac-227+D	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
Ag-108m	P	3	D	7.70E+02	RESRAD Default	NR	NR	NR	NR	
Am-241	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
Am-243+D	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
C-14	P	3	D	9.10E+03	RESRAD Default	NR	NR	NR	NR	
Cm-243	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
Co-60	P	3	D	2.00E+02	RESRAD Default	NR	NR	NR	NR	
Cs-134	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Cs-137+D	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Eu-152	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
Eu-154	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
Eu-155	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
Fe-55	P	3	D	3.20E+03	RESRAD Default	NR	NR	NR	NR	
Gd-152	P	3	D	1.00E+03	RESRAD Default	NR	NR	NR	NR	
H-3	P	3	D	1.00E+00	RESRAD Default	NR	NR	NR	NR	
Nb-94	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Ni-63	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Np-237+D	P	3	D	4.00E+02	RESRAD Default	NR	NR	NR	NR	
Pa-231	P	3	D	1.10E+02	RESRAD Default	NR	NR	NR	NR	
Pb-210+D	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Pu-238	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Pu-239	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Pu-241+D	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Ra-226+D	P	3	D	2.50E+02	RESRAD Default	NR	NR	NR	NR	
Sr-90+D	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Sb-125	P	3	D	1.00E+01	RESRAD Default	NR	NR	NR	NR	
Tc-99	P	3	D	5.00E+00	RESRAD Default	NR	NR	NR	NR	
Th-229+D	P	3	D	5.00E+02	RESRAD Default	NR	NR	NR	NR	
Th-230	P	3	D	5.00E+02	RESRAD Default	NR	NR	NR	NR	
U-233	P	3	D	6.00E+01	RESRAD Default	NR	NR	NR	NR	
U-234	P	3	D	6.00E+01	RESRAD Default	NR	NR	NR	NR	
U-235+D	P	3	D	6.00E+01	RESRAD Default	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	

Table 6M-1
Input Parameters for Sensitivity Analysis, Cellar Hole Concrete Debris
Resident Farmer/Intruder Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Spacing				log	RESRAD Default	NR	NR	NR	NR	
Time integration parameters										
Maximum number of points for dose				17	RESRAD Default	NR	NR	NR	NR	

Notes:

[a] P = physical, B = behavioral, M = metabolic; (see NUREG/CR-6697, Attachment B, Table 4.)
[b] 1 = high-priority parameter, 2 = medium-priority parameter, 3 = low-priority parameter (see NUREG/CR-6697, Attachment B, Table 4.1)
[c] D = deterministic, S = stochastic
[d] Distributions Statistical Parameters:
Lognormal-n: 1= mean, 2 = standard deviation
Bounded lognormal-n: 1= mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Truncated lognormal-n: 1= mean, 2 = standard deviation, 3 = lower quantile, 4 = upper quantile
Bounded normal: 1 = mean, 2 = standard deviation, 3 = minimum, 4 = maximum
Beta: 1 = minimum, 2 = maximum, 3 = P-value, 4 = Q-value
Triangular: 1 = minimum, 2 = mode, 3 = maximum
Uniform: 1 = minimum, 2 = maximum

Additional Sensitivity Analysis Data:
Sampling technique = Latin Hypercube
Random Seed = 1000
Number of observations =2000
Number of repetitions = 1

Input Rank Correlation Coefficients:
Total porosity and Bulk density = - 0.99 (contaminated zone)
Evapotranspiration and Irrigation rate = 0.99
Distribution coefficient and Plant transfer factor = -0..99 (contaminated zone)

References:

1. Code of Federal Regulations, Title10, Section 20.1402, "Radiological Criteria for Unrestricted Use".

2. NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes", December 2000.
3. YA-REPT-01-003-03, "Basis for Selection of Concrete Kd Values," August 2004.
4. Yu, C. et al., "Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil"; US Department of Energy – Argonne National Laboratory, April 1993.
5. Correspondence between J. Lynch and P. Littlefield, "RE. Concrete Debris," August 4, 2004 (Attachment 1)
6. YA-REPT-00-008-03, "Evaluation of GeoTesting Express Soil Testing and Determination of Depth to Groundwater," December 2003
7. NUREG-1757. "Consolidated NMSS Decommissioning Guidance," Volume 2: Characterization, Survey and Determination of Radiological Criteria," September 2003.
8. YA-CALC-02-001-03, "RESRAD 6.21 Sensitivity Analysis for Resident Farmer Scenario – Soil," DATE
9. NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination From Decommissioning: Parameter Analysis, Draft Report for Comment," October 1999.
10. Eckerman, K.F., et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, Federal Guidance Report No. 11, U.S EPA, 1988.

Table 6M-2
Input Parameters for Sensitivity Analysis, H-3 Graded Concrete Debris
Resident Farmer/Intruder Scenario

Graded Concrete Debris (Basis for scenario is Reference 2)

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				
						1	2	3	4	Median
Contaminated Zone										
Thickness of contaminated zone (m)	P	2	S	Uniform	Minimum equal depth of soil mixing layer (0.15m); maximum equal depth to water table (3.8m) [4]	0.15	3.8	NR	NR	1.975
Saturated Zone Hydrological Data										
Density of saturated zone (g/cm^3)	P	1	S	Bounded Normal	NUREG 6697 dist for site soil type - sand [3]	1.5105	0.159	1.019	2.002	1.5105
Saturated zone total porosity	P	1	S	Bounded Normal	NUREG 6697 dist for site soil type - sand [3]	0.43	0.06	0.2446	0.6154	0.43
Saturated zone effective porosity	P	1	S	Bounded Normal	NUREG 6697 dist for site soil type - sand [3]	0.383	0.0610	0.195	0.572	0.383
Saturated zone field capacity	P	3	D	0.05	Site-specific value calculated using Equation 4.4 from [2, 3]	NR	NR	NR	NR	0.05
Saturated zone hydraulic conductivity (m/yr)	P	1	S	Beta	NUREG 6697 dist for site soil type - sand [3]	110	5870	1.398	1.842	2506
Saturated zone b parameter	P	2	S	Bounded Log Normal n	NUREG 6697 dist for site soil type - sand [3]	- 0.0253	0.216	0.501	1.90	0.975
Model: Nondispersion (ND)	P	3	D	ND	ND model for contaminated area > 1000 m2 [1, 2]					
Well pumping rate (m**/yr)	P	2	S	Uniform	Min, Max, median value based on site irrigation and area and calculated according to NUREG/CR-6697, Att. C section 3.10 method. [3]	957	1689	NR	NR	1323
Unsaturated Zone Hydrological Data										
Number of unsaturated zones	P	3	D	1	[3]					

Table 6M-2
Input Parameters for Sensitivity Analysis, H-3 Graded Concrete Debris
Resident Farmer/Intruder Scenario

Graded Concrete Debris (Basis for scenario is Reference 2)										
Unsat. zone 1, thickness (m)	P	1	S	Uniform	Assumes 0.15 to 3.8 m contaminated zone thickness and 3.8 m depth to water table [3]	0.01	3.65			1.82
Unsat. zone 1, soil density (g/cm^3)	P	2	S	Bounded Normal	NUREG 6697 dist for site soil type - sand [3]	1.5105	0.159	1.019	2.002	1.5105
Unsat. zone 1, total porosity	P	2	S	Bounded Normal	NUREG 6697 dist for site soil type - sand [3]	0.43	0.06	0.2446	0.6154	0.43
Unsat. zone 1, effective porosity	P	2	S	Bounded Normal	NUREG 6697 dist for site soil type - sand [3]	0.383	0.0610	0.195	0.572	0.383
Unsat. zone 1, field capacity	P	3	D	0.05	Site-specific value calculated using Equation 4.4 [2, 3]	NR	NR	NR	NR	0.05
Unsat. zone 1, hydraulic conductivity (m/yr)	P	2	S	Beta	NUREG 6697 dist for site soil type - sand [3]	110	5870	1.398	1.842	2506
Unsat. zone 1, soil-specific b parameter	P	2	S	Bounded Log Normal n	NUREG 6697 dist for site soil type - sand [3]	- 0.0253	0.216	0.501	1.90	0.975

References:

1. ANL/EAD-4, "Users Manual for RESRAD Version 6.0," Yu, C. et al., July 2001
2. YA-CALC-02-001-03, "RESRAD 6.21 Sensitivity Analysis for Resident Farmer Scenario – Soil," DATE
3. NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes", December 2000.
4. YA-REPT-00-008-03, "Evaluation of GeoTesting Express Soil Testing and Determination of Depth to Groundwater," December 2003

Appendix N
Sensitivity Analysis Summary

Sensitivity Analysis Summary, Percentile Values and Assignment of Conservative Values for Concrete Debris DCGL Determination

Nuclide R^2	Sensitive Parameter	PRCC	Distribution	Distribution Statistical Parameters 1	2	3	4	Mean	Percentile Values 25%	75%	Assigned Value
Ag-108m $R^2 = 1.0$	External gamma shielding factor	1.0	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
	Milk transfer factor for Ag	0.72	Truncated Lognormal -n	-5.12	0.7	0.001	0.999	7.64E-03		9.57E-03	9.57E-03
Am-241 $R^2 = 1.0$	Kd of Am in contaminated zone	-0.96	Loguniform	200	5000				4.47E02		
	Weathering removal constant of all vegetation	-0.87	Triangular	5.1	18	84			2.15E01		
	Wet foliar interception fraction of leafy vegetables	0.60	Triangular	0.06	0.67	0.95		5.6E-01		7.00E-01	7.00E-01
	Wet weight crop yield of fruit, grain and non-leafy vege.	-0.56	Truncated Lognormal-n	0.56	0.48	0.001	0.999		1.27 E00		
	Plant transfer factor for Am	0.26	Loguniform	1.06E-05	3.31E-03			5.74E-04		7.86E-04	7.86E-04
	Fish transfer factor for Am	0.26	Lognormal-n	3.4	1.1			5.49E+01		6.29E01	6.29E01
C-14 $R^2 = 0.84$	Thickness of evasion layer of C-14 in soil	0.84	Triangular	0.2	0.3	0.6		3.67E-01		4.27E-01	4.27E-01
	Fish transfer factor for C	0.67	Lognormal-n	10.8	1.1			8.98E04		1.03E05	1.03E05
Cm-243 $R^2 = 0.99$	Weathering removal constant of all vegetation	-0.81	Triangular	5.1	18	84			2.15E01		
	Kd of Cm-243 in Contaminated Zone	-0.76	Uniform	200	1000				4.00E02		
	Wet foliar interception fraction of leafy vegetables	0.53	Triangular	0.06	0.67	0.95		5.6E-01		7.00E-01	7.00E-01
	Wet weight crop yield of fruit, grain and no-leafy vegetables	-0.48	Truncated Lognormal-n	0.56	0.48	0.001	0.999		1.27 E00		
	External gamma shielding factor	0.40	Bounded Lognormal-n	-1.3	0.59	0.044		3.24E-01		3.98E-01	3.98E-01
	Plant transfer factor for Cm	0.32	Uniform	1.87E-04	3.31E-03			1.75E-03		2.53E-03	2.53E-03
Co-60 $R^2 = 1.0$	External gamma shielding factor	1.0	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
	Meat transfer factor for Co	0.66	Truncated Lognormal-n	-3.51	1.0	0.001	0.999	4.93E-02		5.85E-02	5.85E-02
Cs-134 $R^2 = 0.91$	External gamma shielding factor	0.88	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
	Milk transfer factor for Cs	0.61	Truncated Lognormal-n	-4.61	0.5	0.001	0.999	1.13E-02		1.39E-02	1.39E-02
	Meat transfer factor for Cs	0.42	Truncated Lognormal-n	-3.00	0.4	0.001	0.999	5.39E-02		6.51E-02	6.51E-02
	Weathering removal constant of all vegetation	-0.29	Triangular	5.1	18	84			2.15E01		

Sensitivity Analysis Summary, Percentile Values and Assignment of Conservative Values for Concrete Debris DCGL Determination

Nuclide R^2	Sensitive Parameter	PRCC	Distribution	Distribution Statistical Parameters				Mean	Percentile Values		Assigned Value
				1	2	3	4		25%	75%	
Cs-137	External gamma shielding factor	0.81	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
R^2 = 0.94	Milk transfer factor for Cs	0.72	Truncated Lognormal-n	-4.61	0.5	0.001	0.999	1.13E-02		1.39E-02	1.39E-02
	Meat transfer factor for Cs	0.53	Truncated Lognormal-n	-3.00	0.4	0.001	0.999	5.39E-02		6.51E-02	6.51E-02
	Weathering removal constant of all vegetation	-0.37	Triangular	5.1	18	84			2.15E01		
	Fish transfer factor for Cs	0.30	Lognormal-n	7.6	0.7			2.55E03		3.20E03	3.20E03
	Plant transfer factor for Cs	0.28	Uniform	2.39E-03	7.83E-02			4.03E-02		5.93E-02	4.03E-02
Eu-152	External gamma shielding factor	1	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
R^2 = 1.0											
Eu-154	External gamma shielding factor	1	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
R^2 = 1.0											
Eu-155	External gamma shielding factor	1	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
R^2 = 1.0											
Fe-55	Meat transfer factor for Fe	0.89	Truncated Lognormal-n	-3.51	0.4	0.001	0.999	3.23E-02		3.91E-02	3.91E-02
R^2 = 0.98	Plant transfer factor for Fe	0.66	Uniform	2.44E-01	1.32E00			7.82E-01		1.05E00	1.05E00
	Weathering removal constant of all vegetation	-0.54	Triangular	5.1	18	84			2.15E01		
	Fish transfer factor for Fe	0.31	Lognormal-n	5.3	1.1			3.67E02		4.20E02	4.20E02
	Milk transfer factor for Fe	0.30	Truncated Lognormal-n	-8.11	0.7	0.001	0.999	3.84E-04		4.81E-04	4.81E-04
H-3 cellar hole	Density of contaminated zone	0.62	Uniform	1.41	1.67			1.54E00		1.60E00	1.60E00
R^2 = 0.98	Irrigation	-0.58	Uniform	0.252	0.618			4.35E-01	3.43E-01		3.43E-01
H-3 graded	Depth of roots	-0.73	Uniform	0.3	3.8			2.05E00	1.17E00		1.17E00
R^2 = 0.97	Thickness of contaminated zone	0.66	Uniform	0.15	3.8			1.98E00		2.89E00	2.89E00
Nb-94	External gamma shielding factor	1	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
R^2 = 1.0											
Ni-63	Milk transfer factor for Ni	0.93	Truncated Lognormal-n	-3.91	0.7	0.001	0.999	2.56E-02		3.21E-02	3.21E-02
R^2 = 0.95	Plant transfer factor for Ni	0.41	Uniform	2.76E-02	6.96E-01			3.62E-01		5.29E-01	5.29E-01
	Weathering removal constant of all vegetation	-0.31	Triangular	5.1	18	84			2.15E01		

Sensitivity Analysis Summary, Percentile Values and Assignment of Conservative Values for Concrete Debris DCGL Determination

Nuclide R^2	Sensitive Parameter	PRCC	Distribution	Distribution Statistical Parameters				Mean	Percentile Values		Assigned Value
				1	2	3	4		25%	75%	
Pu-238 R^2 = 0.99	Kd of Pu in contaminated zone	-0.83	Loguniform	500	5000				8.88E02		
	Weathering removal constant of all vegetation	-0.74	Triangular	5.1	18	84			2.15E01		
	Wet foliar interception fraction of leafy vegetables	0.41	Triangular	0.06	0.67	0.95		5.6E-01		7.00E-01	7.00E-01
	Wet weight crop yield of fruit, grain and non-leafy vege.	-0.36	Truncated Lognormal-n	0.56	0.48	0.001	0.999		1.27E00		
Pu-239 R^2 = 1.0	Kd of Pu in contaminated zone	-0.91	Loguniform	500	5000				8.88E02		
	Weathering removal constant of all vegetation	-0.87	Triangular	5.1	18	84			2.15E01		
	Wet foliar interception fraction of leafy vegetables	0.59	Triangular	0.06	0.67	0.95		5.6E-01		7.00E-01	7.00E-01
	Wet weight crop yield of fruit, grain and non-leafy vege	-0.55	Truncated Lognormal-n	0.56	0.48	0.001	0.999		1.27E00		
	Fish transfer factor for Pu	0.27	Lognormal-n	3.4	1.1			5.49E+01		6.29E01	6.29E01
Pu-241 R^2 = 1.0	Kd of Am241 in contaminated zone	-0.95	Loguniform	200	5000				4.47E02		
	Weathering removal constant of all vegetation	-0.85	Triangular	5.1	18	84			2.15E01		
	Wet foliar interception fraction of leafy vegetables	0.56	Triangular	0.06	0.67	0.95		5.6E-01		7.00E-01	7.00E-01
	Wet weight crop yield of fruit, grain and non-leafy vege	-0.51	Truncated Lognormal-n	0.56	0.48	0.001	0.999		1.27E00		
Sb-125 R^2 = 1.0	External gamma shielding factor	1	Bounded Lognormal-n	-1.3	0.59	0.044	1	3.24E-01		3.98E-01	3.98E-01
Sr-90 R^2 = 0.91	Milk transfer factor	0.91	Truncated Lognormal-n	-6.21	0.5	0.001	0.999	2.28E-03		2.81E-03	2.81E-03
	Weathering removal constant of all vegetation	-0.76	Triangular	5.1	18	84			2.15E01		
	Meat transfer factor for Sr	0.74	Truncated Lognormal-n	-4.61	0.4	0.001	0.999	1.08E-02		1.3E-02	1.3E-02
Tc-99 R^2 = 0.99	Milk transfer factor for Tc	0.84	Truncated Lognormal-n	-6.91	0.7	0.001	0.999	1.28E-03		1.60E-03	1.60E-03
	Plant transfer factor for Tc	0.79	Uniform	1.85E-01	1.73E00			9.60E-01		1.34E00	1.34E00
	Weathering removal constant of all vegetation	-0.48	Triangular	5.1	18	84			2.15E01		

- Source of percentile values is RESRAD ".MCO" files.

Loguniform mean calculated using NUREG/CR-6697, Attachment C, Appendix A
Mean = b -a / (lnb – lna)
a = min
b = max

Triangular mean calculated using NUREG/CR-6697, Attachment C, Appendix A
Mean = (a + b + c) / 3
a = min
b = most likely
c = max

Lognormal mean calculated using the following:
$\mu = \exp([2m+s^2] / 2)$
Where the mean = m and std dev = s, both of the underlying normal distribution

Appendix 6O
DCGL for Concrete Debris
And
Equilibrium Groundwater Concentrations

Table 6O-1 – DCGL for Concrete Debris and % Dose from Exposure Pathways													
Nuclide	DCGL for Concrete Debris (pCi/gm)	Time to Maximum Dose (yr)	Dose Fraction from Water-Independent Pathways (%)					Dose Fraction from Water-Dependent Pathways (%)					
			Ground	Inhalation	Plant	Meat	Milk	Soil	Water	Fish	Plant	Meat	Milk
H-3 cellar hole	100	0	0.0	0.05	1.74	0.03	0.18	0.0	85.13	0.00	12.13	0.10	0.64
H-3 graded	300	0	0.0	0.57	42.55	7.02	47.95	0.0	0.65	0.02	0.92	0.03	0.28
C-14	7.6	0	0.0	0.01	51.50	22.15	21.01	0.0	0.16	4.64	0.34	0.10	0.09
Fe-55	150	0	0.0	0.0	46.39	24.42	2.48	0.0	11.89	1.85	4.09	8.15	0.71
Co-60	4.5	0	97.02	0.0	0.15	0.99	0.14	0.01	0.73	0.06	0.13	0.62	0.14
Ni-63	110	0	0.0	0.0	17.74	1.21	63.82	0.0	3.21	0.09	0.97	0.028	12.67
Sr-90	0.8	0	0.02	0.0	40.85	7.23	12.83	0.01	20.51	0.37	6.42	4.66	7.11
Nb-94	7.4	0	99.57	0.0	0.04	0.0	0.0	0.0	0.30	0.03	0.05	0.0	0.0
Tc-99	64	0.23	0.01	0.0	67.83	0.09	12.06	0.01	12.71	0.07	4.65	0.02	2.55
Ag-108m	7.4	0	99.70	0.0	0.0	0.01	0.25	0.0	0.02	0.0	0.0	0.0	0.01
Sb-125	33	0	99.82	0.0	0.0	0.01	0.0	0.01	0.12	0.0	0.02	0.0	0.0
Cs-134	5.0	0	54.85	0.0	6.53	8.20	11.99	0.02	3.90	2.22	1.06	4.45	6.71
Cs-137	7.1	0	33.63	0.0	11.16	12.59	19.40	0.02	4.66	4.16	1.25	5.24	7.88
Eu-152	10	0	99.82	0.0	0.01	0.02	0.0	0.0	0.12	0.0	0.02	0.0	0.0
Eu-154	9.6	0	99.76	0.0	0.01	0.02	0.0	0.0	0.16	0.0	0.03	0.0	0.0
Eu-155	400	0	98.46	0.0	0.06	0.15	0.02	0.03	1.04	0.03	0.18	0.03	0.01
Pu-238	10	0.12	0.0	0.25	1.60	0.46	0.02	1.97	71.14	0.68	23.74	0.12	0.01
Pu-239	9.3	0.28	0.0	0.25	1.59	0.45	0.02	1.96	70.68	1.32	23.59	0.12	0.01
Pu-241	150	65	0.05	0.39	3.29	0.07	0.03	3.08	68.67	0.03	22.85	0.11	0.01
Am-241	4.3	0.12	0.23	0.12	6.97	0.11	0.02	0.94	67.14	1.99	22.41	0.06	0.02
Cm-243	4.9	0.10	4.20	0.09	17.13	0.04	0.02	0.72	57.42	1.18	19.18	0.02	0.01

Table 6O-2 – Comparison of Well Water Concentrations and Equilibrium Ground Water Concentrations at One Year

Nuclide	Well Water Concentration (pCi/L)	Equilibrium Ground Water Concentration (pCi/L)
H-3 cellar hole	1143	218.78
C-14	1.34	1.17
Fe-55	61.38	60.65
Co-60	3.11	3.11
Ni-63	27.88	27.77
Sr-90	92.09	90.80
Nb-94	3.16	3.16
Tc-99	73.51	72.71
Ag-108m	0.15	0.15
Sb-125	0.50	0.50
Cs-134	5.21	5.21
Cs-137	7.13	7.12
Eu-152	0.95	0.95
Eu-154	0.92	0.92
Eu-155	0.87	0.87
Pu-238	1.12	1.12
Pu-239	1.13	1.13
Pu-241	0.60	0.60
Am-241	2.23	2.23
Cm-243	2.44	2.44

Appendix 6 P

Input Parameter Values for Area Factors, Soil

1. General Information

The input parameters for the soil area factor calculations are, in general, the same as those in LTP Appendix 6D. Areas of difference in input parameter values are highlighted in the sections to follow.

2. Conceptual Model, Scenario, and Dose Pathways

The resident farmer scenario, as described in Volume 1 of NUREG/CR-5512 (Ref. 1), assumes a reasonably conservative scenario for establishing DCGL values for residual radioactivity in soil. The same scenario is assumed for the area factor (AF) calculations.

The conceptual model used in the code is based on the site characteristics expected at the time of release of the site. The model is comprised of a contaminated zone underlain by an unsaturated zone underlain by a saturated zone. The contaminated zone is assumed to be at the ground surface with no cover material and the ground water is initially uncontaminated.

The potential exposure pathways that apply to the resident farmer are listed below and are based upon those in NUREG/CR-5512, Volume 1:

- Direct exposure to external radiation from residual radioactivity;
- Internal dose from inhalation of airborne radionuclides; and
- Internal dose from ingestion of
  - Plant foods grown in media containing residual radioactivity and irrigated with water containing residual radioactivity,
  - Meat and milk from livestock fed with fodder grown in soil containing residual radioactivity and water containing residual radioactivity,
  - Drinking water (containing residual radioactivity) from a well,
  - Fish from a pond containing residual radioactivity, and
  - Soil containing residual radioactivity.

3. Contaminated Fractions — Food Pathways

As the size of the contaminated area (A) varies, the fraction of the total food consumed by the receptor grown in the contaminated area will also vary. The fraction of the food supply grown in the contaminated is referred to as a "contaminated fraction." Accordingly, with the decrease in the size of the contaminated area, a decrease in the values for the contaminated fraction of plant food ingested (FPLANT), the contaminated fraction of meat ingested (FMEAT), and contaminated fraction of milk ingested (FMILK) will also result.

The variation in the contaminated fraction of plant food ingested, with the variation in the size of the contaminated area, is described by Equation D.5 of the RESRAD User Manual (Ref. 2):

$$\text{FPLANT} = A/2000, \text{ when } A \leq 1000 \text{ m}^2$$

$$\text{FPLANT} = 0.5, \text{ when } A \geq 1000 \text{ m}^2$$

However, the assumption used in calculating soil DCGLs is that 100% of the plant food consumed is grown in the contaminated area (equivalent to a contaminated fraction = 1.0), when the size of the contaminated area is 13,022 m^2. Thus, Equation D.5 of the RESRAD User Manual has been adjusted, as follows, to match that assumption, and this adjusted relationship is used in the calculation of area factors:

FPLANT = A/1000, when A < 1000 m^2
FPLANT = 1.0, when A ≥ 1000 m^2

The variation in the contaminated fraction of meat and milk ingested, with the variation in the size of the contaminated area, is also described by Equation D.5 of the RESRAD User Manual (Ref. 2):

FA = A/20000, when A ≤ 20000 m^2
FA = 1.0, when A ≥ 20000 m^2

Where FA = FMEAT or FMILK

Again the assumption used in calculating soil DCGLs is that 100% of the meat food and milk consumed are grown in the contaminated area (equivalent to a contaminated fraction = 1.0 for meat and milk), when the size of the contaminated area is 13,022 m^2. Equation D.5 of the RESRAD User Manual has been adjusted, as follows, to match that assumption, and this adjusted relationship is used in the calculation of area factors:

FA = A/13,022	A < 13022 m^2
FA = 1	A = 13022 m^2

Where FA = FMEAT or FMILK

Table 1 shows the values for FPLANT, FMEAT, and FMILK as a function of the size of the contaminated zone.

4. Contaminated Fraction – Water Pathways

Unlike the contaminated fractions of food described above, the contaminated fractions for drinking water (FDW), livestock water (FLW), irrigation water (FIRW), and aquatic food (FR9) are assumed not to decrease as the size of the contaminated zone decreases. Setting the values for these input parameters to 1.0 maintains the assumption that all water used by the resident farmer comes from a well on site, regardless of the size of the contaminated area.

5. Size of the Contaminated Zone

Another input parameter that is influenced by changes in the size of the contaminated zone is the length parallel to aquifer flow (LCZPAQ). As the area of the contaminated zone decreases, the value of LCZPAQ will also decrease. As the contaminated zone is assumed to be circular, the value for LCZPAQ is equal to the diameter of the circle:

$$LCZPAQ(m) = 2\sqrt{\frac{A(m^2)}{\pi}}$$

Table 1 shows the values for LCZPAQ as a function of the size of the contaminated zone.

Table 1
Contaminated Fractions Versus Size of Contaminated Zone

RESRAD Parameter	Input Value						
Contaminated Zone Area (m^2)	**13022**	**11500**	**10000**	**7500**	**5000**	**2500**	**1000**
LCZPAQ (m)	129	121	113	98	80	56	36
FPLANT	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00
FMEAT	1.0E+00	8.8E-01	7.7E-01	5.8E-01	3.8E-01	1.9E-01	7.7E-02
FMILK	1.0E+00	8.8E-01	7.7E-01	5.8E-01	3.8E-01	1.9E-01	7.7E-02
Contaminated Zone Area (m^2)	**750**	**500**	**250**	**100**	**75**	**50**	**25**
LCZPAQ (m)	31	25	18	11	9.8	8.0	5.6
FPLANT	7.5E-01	5.0E-01	2.5E-01	1.0E-01	7.5E-02	5.0E-02	2.5E-02
FMEAT	5.8E-02	3.8E-02	1.9E-02	7.7E-03	5.8E-03	3.8E-03	1.9E-03
FMILK	5.8E-02	3.8E-02	1.9E-02	7.7E-03	5.8E-03	3.8E-03	1.9E-03
Contaminated Zone Area (m^2)	**10**	**8**	**6**	**4**	**2**	**1**	--
LCZPAQ (m)	3.6	3.2	2.8	2.3	1.6	1.1	--
FPLANT	1.0E-02	8.0E-03	6.0E-03	4.0E-03	2.0E-03	1.0E-03	--
FMEAT	7.7E-04	6.1E-04	4.6E-04	3.1E-04	1.5E-04	7.7E-05	--
FMILK	7.7E-04	6.1E-04	4.6E-04	3.1E-04	1.5E-04	7.7E-05	--

6. Year of Maximum Dose

The year in which the maximum dose occurs may vary depending on the nuclide. The concentration delivering the maximum dose is selected for the basis of the AF without regard to year of occurrence.

7. Initial Concentration

An initial soil concentration of 1 pCi/g is assumed for each nuclide.

References:

1. NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning," Volume 1: "Technical Basis for Translating Contamination Levels to Annual TEDE," October 1992.

2. Yu, C. et al., "Users Manual for RESRAD Version 6," ANL/EAD-4, July 2001.

Appendix 6 Q

Area Factors for Soil

Area Factors for Soil

Nuclide	Area of Source (m^2)									
	13022	11500	10000	7500	5000	2500	1000	750	500	250
H-3	1.0E+00	1.1E+00	1.1E+00	1.3E+00	1.5E+00	1.8E+00	2.0E+00	2.7E+00	4.0E+00	8.0E+00
C-14	1.0E+00	1.1E+00	1.3E+00	1.6E+00	2.3E+00	3.7E+00	6.4E+00	9.7E+00	1.7E+01	4.5E+01
Fe-55	1.0E+00	1.1E+00	1.3E+00	1.6E+00	2.2E+00	3.4E+00	5.2E+00	7.0E+00	1.1E+01	2.1E+01
Co-60	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.2E+00	1.2E+00	1.3E+00
Ni-63	1.0E+00	1.1E+00	1.2E+00	1.5E+00	2.0E+00	2.8E+00	3.8E+00	5.1E+00	7.7E+00	1.5E+01
Sr-90	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.2E+00	1.3E+00	1.4E+00	1.8E+00	2.7E+00	5.4E+00
Nb-94	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
Tc-99	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00	1.5E+00	2.3E+00	4.5E+00
Ag-108m	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
Sb-125	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
Cs-134	1.0E+00	1.0E+00	1.1E+00	1.2E+00	1.3E+00	1.4E+00	1.5E+00	1.6E+00	1.7E+00	1.8E+00
Cs-137	1.0E+00	1.1E+00	1.1E+00	1.2E+00	1.4E+00	1.6E+00	1.7E+00	1.9E+00	2.1E+00	2.4E+00
Eu-152	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
Eu-154	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
Eu-155	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.1E+00	1.1E+00	1.1E+00	1.1E+00
Pu-238	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.4E+00	2.0E+00	4.0E+00
Pu239	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.4E+00	2.0E+00	4.0E+00
Pu241	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.3E+00	2.0E+00	3.8E+00
Am-241	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.3E+00	2.0E+00	3.8E+00
Cm-243	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.0E+00	1.2E+00	1.6E+00	2.3E+00

Nuclide	Area of Source (m^2)									
	100	75	50	25	10	8	6	4	2	1
H-3	2.0E+01	2.6E+01	3.9E+01	7.5E+01	1.8E+02	2.2E+02	2.9E+02	4.2E+02	8.0E+02	1.5E+03
C-14	1.5E+02	2.2E+02	3.7E+02	8.6E+02	2.4E+03	3.1E+03	4.1E+03	6.0E+03	1.2E+04	2.4E+04
Fe-55	5.2E+01	7.0E+01	1.0E+02	2.1E+02	5.2E+02	6.5E+02	8.5E+02	1.3E+03	2.5E+03	4.7E+03
Co-60	1.4E+00	1.4E+00	1.5E+00	1.8E+00	2.4E+00	2.7E+00	3.2E+00	4.1E+00	6.5E+00	1.1E+01
Ni-63	3.8E+01	5.1E+01	7.7E+01	1.5E+02	3.8E+02	4.8E+02	6.4E+02	9.5E+02	1.9E+03	3.8E+03
Sr-90	1.4E+01	1.8E+01	2.7E+01	5.4E+01	1.3E+02	1.6E+02	2.2E+02	3.2E+02	6.4E+02	1.3E+03
Nb-94	1.2E+00	1.2E+00	1.3E+00	1.5E+00	2.0E+00	2.3E+00	2.8E+00	3.5E+00	5.5E+00	9.3E+00
Tc-99	1.1E+01	1.5E+01	2.3E+01	4.5E+01	1.1E+02	1.4E+02	1.9E+02	2.8E+02	5.6E+02	1.1E+03
Ag-108m	1.2E+00	1.2E+00	1.3E+00	1.5E+00	2.0E+00	2.3E+00	2.7E+00	3.5E+00	5.5E+00	9.2E+00
Sb-125	1.2E+00	1.2E+00	1.3E+00	1.5E+00	2.0E+00	2.3E+00	2.7E+00	3.5E+00	5.4E+00	9.1E+00
Cs-134	2.0E+00	2.1E+00	2.3E+00	2.7E+00	3.6E+00	4.0E+00	4.8E+00	6.1E+00	9.7E+00	1.6E+01
Cs-137	2.8E+00	2.9E+00	3.1E+00	3.7E+00	4.9E+00	5.6E+00	6.6E+00	8.5E+00	1.3E+01	2.2E+01
Eu-152	1.2E+00	1.2E+00	1.3E+00	1.5E+00	2.1E+00	2.3E+00	2.8E+00	3.5E+00	5.6E+00	9.4E+00
Eu-154	1.2E+00	1.3E+00	1.3E+00	1.5E+00	2.1E+00	2.4E+00	2.8E+00	3.6E+00	5.6E+00	9.6E+00
Eu-155	1.2E+00	1.2E+00	1.3E+00	1.5E+00	1.9E+00	2.2E+00	2.6E+00	3.2E+00	5.0E+00	8.0E+00
Pu-238	9.7E+00	1.3E+01	1.9E+01	3.4E+01	7.2E+01	8.4E+01	1.0E+02	1.3E+02	1.8E+02	2.4E+02
Pu239	9.7E+00	1.3E+01	1.9E+01	3.4E+01	7.2E+01	8.4E+01	1.0E+02	1.3E+02	1.8E+02	2.4E+02
Pu241	8.7E+00	1.1E+01	1.5E+01	2.5E+01	4.5E+01	5.2E+01	6.3E+01	8.0E+01	1.2E+02	1.6E+02
Am-241	8.7E+00	1.1E+01	1.5E+01	2.5E+01	4.5E+01	5.2E+01	6.2E+01	7.9E+01	1.2E+02	1.6E+02
Cm-243	3.3E+00	3.6E+00	4.0E+00	4.9E+00	6.8E+00	7.7E+00	9.1E+00	1.2E+01	1.8E+01	3.0E+01

Appendix 6R

Input Parameter Values for Area Factors, Building Occupancy

1. Changes to Input Parameter Set for Building Occupancy DCGLs.

In calculating area factors (AF) for building surfaces, RESRAD-BUILD (v 3.21) was used with the building occupancy scenario to determine the annual dose from 1pCi/m2 for various size sources. A modification of the input assumptions, used for calculating building occupancy DCGLs, was made to consider that only the specified area of the floor as contaminated. The size of this contaminated area is varied from the value of the entire floor surface area (19.7 m^2) to a value of 1 m^2. In calculating the AFs, the contamination of the entire floor is considered as the base case and a specific derived concentration guideline is defined. This specific DCGL is designated $DCGL_{w1}$ to differentiate it from the $DCGL_w$ determined for the entire room. The remaining parameters are those described in LTP Appendix 6G.

Appendix 6S

Area Factors for Building Surface Areas

Area Factors for Building Surfaces

Nuclide	Area of Source (m^2)								
	19.7	15	12	10	8	6	4	2	1
H-3	1.0	1.3	1.6	2.0	2.5	3.3	4.9	9.9	19.7
C-14	1.0	1.3	1.6	2.0	2.4	3.3	4.9	9.7	19.4
Fe-55	1.0	1.3	1.6	2.0	2.5	3.3	4.9	9.9	19.7
Co-60	1.0	1.1	1.3	1.4	1.6	1.9	2.4	4.1	7.3
Ni-63	1.0	1.3	1.6	2.0	2.5	3.3	4.9	9.9	19.7
Sr-90	1.0	1.3	1.6	1.9	2.4	3.2	4.8	9.4	18.6
Nb-94	1.0	1.1	1.3	1.4	1.6	1.9	2.4	4.0	7.2
Tc-99	1.0	1.3	1.6	1.9	2.4	3.2	4.7	9.2	18.2
Ag-108m	1.0	1.1	1.3	1.4	1.6	1.9	2.4	4.0	7.2
Sb-125	1.0	1.1	1.3	1.4	1.6	1.9	2.4	4.1	7.2
Cs-134	1.0	1.1	1.3	1.4	1.6	1.9	2.5	4.2	7.4
Cs-137	1.0	1.1	1.3	1.4	1.6	1.9	2.5	4.2	7.6
Eu-152	1.0	1.1	1.3	1.4	1.6	1.9	2.4	4.0	7.2
Eu-154	1.0	1.1	1.3	1.4	1.6	1.9	2.4	4.0	7.2
Eu-155	1.0	1.1	1.3	1.4	1.6	1.9	2.5	4.1	7.4
Pu-238	1.0	1.3	1.6	2.0	2.5	3.3	4.9	9.9	19.7
Pu-239	1.0	1.3	1.6	2.0	2.5	3.3	4.9	9.8	19.8
Pu-241	1.0	1.3	1.6	2.0	2.5	3.3	4.9	9.8	19.5
Am-241	1.0	1.3	1.6	2.0	2.4	3.3	4.9	9.7	19.5
Cm-243	1.0	1.3	1.6	1.9	2.4	3.2	4.7	9.3	18.5

At the time of license termination, the site will be a backfilled and graded land area, with the potential for selected above grade structures to remain. In general, structures are being demolished to site elevation 1022'-8" with the demolition debris passing final status survey or meeting the "no detectable" criteria able to be used as backfill onsite. Any remaining partial basements will be perforated, to allow groundwater to flow through.

In general buried piping and utilities have been or will be removed. Any buried piping or utilities to remain will be evaluated and surveyed in place, as appropriate, in accordance with plant procedures to ensure that no detectable radioactivity exists.

8.1.3 Remaining Dismantlement and Decommissioning Activities

YAEC originally submitted a Decommissioning Plan (Reference 8-7), which was approved in February of 1995. In accordance with Regulatory Guide 1.185 (Reference 8-8), licensees with approved decommissioning plans were permitted to "replace their decommissioning plans with a Post-Shutdown Decommissioning Activities Report (PSDAR) update that uses the format and content specified in this document." YAEC later elected to relocate pertinent information to a PSDAR (Reference 8-9) conforming to the guidance of Regulatory Guide 1.185.

YAEC continues to implement the DECON alternative as the most appropriate alternative for decommissioning the YNPS site. Evaluation of the environmental effects of the DECON alternative is contained in NUREG-0586 and its supplement.

8.1.3.1 General Description of Decommissioning Activities

Since 1993 YAEC has removed and disposed of the steam generators, pressurizer, reactor vessel and reactor vessel internals. Portions of the reactor vessel internals are considered to be greater-than-Class-C (GTCC) waste and are stored in the ISFSI.

As indicated in the PSDAR, the decommissioning activities are being completed in three phases:

- The first phase of decommissioning consisted of mechanically and electrically isolating the Spent Fuel Pit, removing of any systems and components that did not support fuel storage in the SFP or subsequent decommissioning, and moving spent fuel and GTCC to the ISFSI. The first phase of decommissioning was completed when the spent fuel and all GTCC waste was removed from the SFP in June of 2003.

- The second phase of decommissioning involves the dismantlement and de-contamination of remaining systems, structures, and components (SSCs), including the SFP and its supporting SSCs. It also includes the removal of most of the structures to grade. This phase of decommissioning is ongoing.

- The final phase of decommissioning is the termination of the possession only license.

A more detailed discussion of the activities to be performed in each of the phases is provided in Section 3 of this LTP

and licensed facilities. If technology, resources, and approved processes become available, they will be evaluated to render the mixed waste non-hazardous.

8.1.3.3.8 Storage/Removal of Spent Fuel and GTCC Waste

YAEC will store spent fuel and GTCC waste in the ISFSI, until the DOE takes title to such wastes. Movement of fuel to the ISFSI began in June of 2002 and was completed in June of 2003. GTCC wastes were moved to the ISFSI in June of 2003.

YAEC cannot make a precise determination of when spent fuel and GTCC wastes will be removed from the YNPS site. Currently, YAEC expects that turnover to the DOE of spent fuel and GTCC wastes will be completed in 2022.

8.1.3.3.9 LTP, Final Status Survey, and Site Release Criteria

The ultimate goal of decommissioning the YNPS site is to release it for unrestricted use. This requires assurance that future uses of the site, after license termination, will not expose members of the general public to unacceptable levels of radiation.

Section 1 provides a history of previous LTP and final status survey (also referred to as the final radiological survey) activities. Consistent with a commitment made in the PSDAR, this LTP uses the guidance of NUREG-1700 to address the 10CFR20 criteria for license termination. Final status surveys will then be conducted to verify that structures and open land areas meet the release criteria. An independent NRC contractor will then conduct a verification survey, thereby allowing unrestricted release of the site. After final status survey and NRC verification, some of the remaining surveyed structures and open land areas may be removed from the license. YAEC will then maintain control over the site until license termination.

8.1.3.3.10 Site Restoration

Many site restoration activities may be initiated during the dismantlement period. During decommissioning those remaining plant structures are to be demolished. All building foundations will be back filled with structural fill or concrete debris (with no detectable radioactivity or which has passed final status survey). Site areas will be graded and landscaped as necessary.

8.1.3.4 Schedule of Decommissioning Activities

The current schedule for decommissioning activities is provided in Section 3 of this LTP. Planning sequences and dates are based upon current knowledge and could change in the future. Yankee will continue to inform the NRC of all major changes to the planned decommissioning activities in accordance with 10CFR50.82(a)(7).

Since 1994, a number of systems that contributed to water usage have been removed from operation. Section 3 of this LTP describes those water-containing systems that have been removed from service or drained and identifies the systems remaining in operation. Only a few systems remain, and as described in Supplement 1 to NUREG-0586, the operational demands for cooling and make-up water have been eliminated with the removal of spent fuel and GTCC waste from the spent fuel pit.

Use of water for decontamination of systems such as the Reactor Coolant System and the Spent Fuel Pit are addressed in the FGEIS. Other water usage, such as for dust abatement, are similar to those that occurred during construction of the plant. In addition, potable water for decommissioning contractor staff is being provided via bottled water, and sanitary services are provided by portable toilet facilities, thus minimizing the impacts on the on-site water supply.

In summary, the conditions for YNPS decommissioning are consistent with the assumptions of Supplement 1 to the FGEIS, and thus there are no significant environmental impacts associated with water use during the decommissioning of the YNPS. YAEC has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.3 Water Quality

The environmental impacts associated with surface water quality have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of surface water quality is documented in Section 4.3.3 of Supplement 1 to NUREG-0586.

All discharges are controlled under the National Pollutant Discharge Elimination System (NPDES) permit (Reference 8-13). This permit is issued jointly by the U.S. Environmental Protection Agency (EPA) and the Massachusetts Department of Environmental Protection (MDEP). The Offsite Dose Calculation Manual (Reference 8-14) also addresses limitations on doses to members of the public from liquid effluent and requires that they be maintained below the limits in:

- 10CFR50, Appendix I;
- 10CFR20, Appendix B, Table 2, Column 1; and
- 40CFR190.

Radiological impacts are being assessed and monitored by use of on- and offsite groundwater monitoring wells for aquifers that discharge to Sherman Reservoir, including monitoring Sherman Spring. Currently the levels of radionuclides in these well samples, with the exception of tritium, are below the EPA's drinking water MCLs. A detailed discussion about the groundwater assessments (completed and planned) and available data are provided in Section 2 of this LTP.

As previously discussed, site buildings are being removed to ground level at 1022'-8", and basements are being cleaned to meet the appropriate DCGLs. These basements are also being perforated to allow equilibrium with the water table, and soils are being used to backfill the holes. Concrete debris from demolition of the buildings may be used as backfill onsite if it

passes a final status survey or meet the "no detectable" criteria. A "beneficial use determination" (BUD) to use this concrete as backfill is being filled with the State of Massachusetts Department of Environmental Protection. As a part of the BUD approval, the DEP must make the conclusion that the reuse will not cause significant risk or impact or create a nuisance condition.

Thus, the conditions for YNPS decommissioning are consistent with the assumptions of Supplement 1 to the FGEIS, and thus there are no significant environmental impacts associated with surface water quality during the decommissioning of YNPS. YAEC has not identified any new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.4 Air Quality

The environmental impacts of decommissioning associated with air quality have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of air quality is documented in Section 4.3.4 of Supplement 1 to the FGEIS.

Supplement 1 to the FGEIS identifies the following decommissioning activities as having the potential for non-radiological impacts on air quality:

- Worker transportation to and from the site,
- Dismantling of systems and removal of equipment,
- Movement and open storage of materials onsite,
- Demolition of buildings and structures, and
- Shipment of material and debris to offsite locations.

Worker transportation: Consistent with the assumptions in the FGEIS, the work force at YNPS has decreased from the time the plant ceased operation. The work force will further decrease as decommissioning nears completion. There will and have been occasional increases during specific decontamination and decommissioning activities. The work force during decommissioning is smaller than that associated with plant construction and refueling at YNPS. Accordingly, the adverse changes in air quality, associated with changes in worker transportation, will not be detectable and are not destabilizing.

Dismantling systems and removal of equipment: Generation of particulate matter associated with the physical activities of dismantlement and by the release of gases from systems during removal are potential sources that could impact air quality. Methods and provisions are available to minimize fugitive dust (e.g., wet suppression and chemical stabilization agents) and to minimize airborne contamination in buildings (e.g., isolation of areas and HEPA filtration). Local filtration systems can also be used when activities are located in areas that are not ventilated to the plant stack, and are likely to generate airborne radioactivity. Thus, it is highly unlikely that particulate matter generated during decommissioning and released to the environment will be detectable offsite. Any refrigerants will be disposed of in accordance with the applicable state and federal regulations.