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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 Revision 1 to the Yankee Nuclear Power Station's License Termination Plan (LTP)

This letter submits Revision 1 to the Yankee Atomic Electric Company (YAEC) License Termination Plan (LTP) for the Yankee Nuclear Power Station (YNPS). This revision incorporates our responses to the Request for Additional Information (RAIs)[1] as well as the modifications to our materials management program as presented at the June 17, 2004 meeting[2]. Some editorial and minor clarifications have also been incorporated. For convenience the enclosed Revision 1 is a complete document, rather than replacement pages. All changes are 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 us.

Sincerely,

YANKEE ATOMIC ELECTRIC COMPANY

J. A. Kay

J. A. Kay
Principal Licensing Engineer

Enclosure: YNPS License Termination Plan, Revision 1

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

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

Yankee Nuclear Plant Station
License Termination Plan

Yankee Atomic Electric Company

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List of Acronyms

ALARA	As Low As Reasonably Achievable
AMDA	Alternate Method of Disposal Authorization
AOR	Abnormal Operating Report
ASWS	Auxiliary Service Water System
CFR	Code of Federal Regulations
cpm	Counts per minute
CR	Condition Report
DCGL	Derived Concentration Guideline Level
$DCGL_w$	DCGL for average concentration over a wide area, used with statistical tests
$DCGL_{EMC}$	DCLGS for small areas of elevated activity
DEP	[Massachusetts] Department of Environmental Protection
DOD	Department of Defense
DOE	Department of Energy
DOT	Department of Transportation
DPH	[Massachusetts] Department of Public Health
dpm	Disintegrations per minute
DQO	Data quality objective
EMC	Elevated Measurement Comparison
EPA	Environmental Protection Agency
FERC	Federal Energy Regulatory Commission
FGEIS	Final Generic Environmental Impact Statement
FSS	Final Status Survey
FSAR	Final Safety Analysis Report
GPS	Global positioning system
GTCC	Greater than Class C [Waste]
HEPA	High Efficiency Particulate Air
HSA	Historical Site Assessment
ISFSI	Independent Spent Fuel Storage Installation
LBGR	Lower Bound Grey Region
LER	License Event Report
LLW	Low Level Waste
LTP	Licence Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PAB	Plant Auxiliary Building
PIR	Plant Investigation Report
PSDAR	Post-Shutdown Decommissioning Activities Report
QA	Quality Assurance
QAP	Quality Assurance Program
QAPP	Quality Assurance Program Plan
QC	Quality Control
RCA	Radiologically Controlled Area
RESRAD	RESidual RADioactivity [Computer Code]
REMP	Radiological Environmental Monitoring Program

List of Acronyms

RETS	Radiological Environmental Technical Specifications
RIR	Radiological Incident Report
SSCs	Structures, Systems, and Components
SFP	Spent Fuel Pit
TEDE	Total Effective Dose Equivalent
TRU	Trans-Uranics
WRS	Wilcoxon Rank Sum [test]
YAEC	Yankee Atomic Electric Company
YNPS	Yankee Nuclear Power Station

1 GENERAL INFORMATION

1.1 Executive Summary

The objective for decommissioning the Yankee Nuclear Power Station (YNPS) site is to reduce residual radioactivity to levels that permit release of the site for unrestricted use and for termination of the 10CFR50 license, in accordance with the Nuclear Regulatory Commission's (NRC's) site release criteria set forth in 10CFR20, Subpart E. The purpose of this YNPS License Termination Plan (LTP) is to satisfy the requirements of 10CFR50.82, "Termination of License" (Reference 1-1) using the guidance provided in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 1-2). NRC staff review guidance, in the form of NUREG-1700 (Reference 1-3) and NUREG-1757 (Reference 1-4), has also been considered.

This LTP describes the decommissioning activities that will be performed, the process for performing the Final Status Surveys, and the method for demonstrating that the site meets the criteria for release for unrestricted use. The LTP contains specific information on:

- historical site assessment;
- site characterization;
- remaining decommissioning activities;
- site remediation plans;
- final status survey design and implementation;
- dose modeling scenarios;
- update to the site-specific decommissioning cost estimate; and
- supplement to the environmental report.

Each section of the LTP is summarized in Section 1.4.

1.2 Description of the YNPS Site and Surrounding Areas

1.2.1 YNPS Site

The Yankee Nuclear Power Station (YNPS) is located at 49 Yankee Road, Rowe, in Franklin County, Massachusetts. Yankee Atomic Electric Company (YAEC) is the license holder for YNPS. The plant site contains 2200 acres, approximately 10 acres of which were developed for plant use. The site is at the bottom of a deep valley along the Deerfield River (elevation 1022') at the southeast corner of Sherman Reservoir (also referred to as Sherman Pond). The area surrounding the site is mostly wooded with very steep slopes on both sides of the Deerfield River. The hills on either side of the site rise about 1000 feet above the river and extend from 12 miles north to 8 miles southeast of the site. Sherman Reservoir served as the source of cooling water for the plant.

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).

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.

<u>Major Bodies of Water:</u> In addition to Sherman Reservoir and the Deerfield River (including tributaries and brooks feeding it), there are other major bodies of water 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.

<u>Industry:</u> 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 that are 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.

<u>Public Lands and Conservation Areas:</u> 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.

<u>Schools:</u> 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 Route 100 near the center of Readsboro, Vermont.

<u>Farms</u>: 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.

<u>Water Supplies:</u> 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

water within 5 miles of the plant site is Phelps Brook, which services some of the residents of Monroe, Massachusetts. Beyond 5 miles, downstream there are two small water supply wells servicing local private developments: the Deerfield River Club and Heath Stage Apartments in Charlemont, Massachusetts. Still further downstream, the closest public water supply wells, Stillwater Springs, are in the town of Deerfield, 20 to 25 miles south of the YNPS. Stillwater Springs has a safe yield of about 120,000 gallons per day. This well field is immediately adjacent to the Deerfield River. Another supply well, the Deerfield Well Field, off Route 116, has been closed due to contamination from nearby agricultural uses. The Quabbin Reservoir, serving the greater Boston area, is 35 to 40 miles southeast of the YNPS.

Population: The population within 10 miles of the site is estimated to be 39,300 and includes 17 municipalities in two states. Table 1-2 shows the total population in each town with borders within 10 miles of the plant. In general, the area is rural, with North Adams being the most populous municipality.

1.3 Historical Information

YNPS (Docket No. 50-029) achieved initial criticality in 1960 and began commercial operations in 1961. The nuclear steam supply system was a four-loop pressurized water reactor designed by Westinghouse Electric Corporation. The original thermal power design limit of 485 MWt was upgraded to 600 MWt in 1963. The turbine generator, also designed by Westinghouse, was rated to produce 185 MWe.

On February 26, 1992, the Yankee Atomic Electric Company (YAEC) Board of Directors decided to cease power operations permanently at YNPS. This decision was based upon the following two factors:

1. Economic analyses indicated that shutdown of the plant before expiration of the NRC operating license in July 2000 could produce a substantial savings to the electricity producers.

2. Significant regulatory uncertainty existed concerning the timing and cost of completion of the NRC's review of the integrity of the YNPS Reactor Pressure Vessel.

On August 5, 1992, the NRC amended the YNPS Facility Operating License to a possession only status.

The YNPS Decommissioning Plan (Reference 1-6) was submitted March 29, 1994, and received final approval on October 28, 1996 (References 1-7 and 1-8). In May 1997, Yankee submitted to the NRC for approval a License Termination Plan (LTP) for YNPS, pursuant to 10CFR50.82(a)(9). The initial YNPS LTP employed a survey methodology based upon the "Manual for Conducting Radiological Surveys in Support of License Termination, (Reference 1-9)," also referred to as the Draft NUREG/CR-5849 methodology. Subsequently the NRC, jointly with the DOD, DOE, and EPA, approved an alternate survey methodology documented in MARSSIM ("Multi-Agency Radiation Survey and Site Investigation Manual" or

NUREG-1575, Reference 1-10). In May 1999, Yankee advised the NRC that it intended to shift from the survey methodology in NUREG/CR-5849 to the MARSSIM methodology and, therefore, withdrew its previously submitted LTP application. The current LTP is written to reflect the MARSSIM methodology, as well as appropriate regulatory guidance made available since the previous LTP submittal.

In 2000, Yankee created a Post-Shutdown Decommissioning Activities Report (PSDAR) within the Final Safety Analysis Report (FSAR). NRC Draft Regulatory Guide DG-1071 recommends that licensees with approved Decommissioning Plans (D Plans) "extract pertinent detail from the decommissioning plan and submit a PSDAR update in the format and content specified by [DG-1071]." Based on the NRC draft guidance, Yankee segregated, updated and condensed certain information concerning post-shutdown decommissioning activities in a manner that conforms to the standard format and content of a PSDAR.

1.4 Plan Summary

1.4.1 General Information

This LTP has been prepared by YAEC in accordance with the requirements of 10CFR50.82(a)(9). The LTP is being maintained as a supplement to the YNPS FSAR to support the application for a license amendment to meet 10CFR50.82(a)(9) and 10CFR50.90. Each of the sections required by 10CFR50.82(a)(9) are outlined in the subsections below.

1.4.2 HSA and Site Classification

The objectives of the site classification are:

1. To divide the site into survey areas for classification purposes;

2. To identify the potential and known sources of radioactive contamination in systems, on structures, in surface or subsurface soils, and in groundwater;

3. To determine the initial classification of each survey area; and

4. To develop the information to support Final Status Survey design including instrument performance standards and quality requirements.

The site classification is based upon the Historical Site Assessment (HSA). The HSA consisted of a review and compilation of the following types of information: historical records, plant and radiological incident files, operational survey records, and annual environmental reports to the NRC. Personnel interviews were conducted with present and former plant employees and contractors to obtain additional information regarding operational events that caused contamination in areas or systems not designed to contain radioactive or hazardous materials.

Information from previous surveys, including those in support of the previous Final Status Survey campaign, was reviewed for radiological conditions throughout the site. The radiological data collected during this process provide 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 1999, based upon the availability of new survey guidance in MARSSIM. The focus then shifted from decommissioning activities to spent fuel storage activities. 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 were completed in June 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 radioactivity to levels meeting the NRC radiological release criteria. This means that levels of radioactivity will not exceed 25 mrem/yr total effective dose equivalent (TEDE) and will be 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 levels 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 doses from small areas of elevated activity are below the release criterion for each survey unit.

The purpose of the Final Status Survey Plan is to describe the methods that will 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 that 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 that will be used to demonstrate that the YNPS site complies with the radiological criteria of 10CFR20.1402 for unrestricted use. 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. Section 7 also compares these estimated costs to the amount of funds presently set aside for decommissioning and describes the methods that will ensure 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).

Section 8 is focused on the evaluation of those aspects of decommissioning whose impacts could not be generically addressed (i.e., those determined to have site-specific impacts) and on whether remaining license termination activities and end use of the site are bounded by prior assessments.

1.5 Partial Site Release Process

YAEC may choose to remove specific areas from the license in a phased manner before license termination. The approach for phased release and removal from the license, after approval of the License Termination Plan, is as follows:

1. Following completion of decommissioning activities, YAEC will compile a report with the following information for NRC review:

    - a description and location of the survey unit or area being surveyed;

    - certification that dismantlement/decommissioning activities, as described in the LTP, have been completed for the subject building or area;

    - an evaluation of the potential for possible recontamination of the area and a description of controls in place to prevent such recontamination;

    - Final Status Survey results for the survey unit or area, as demonstration of compliance with the LTP release criteria (not applicable to areas designated as "non-impacted");

    - Expected date of removal of the area from the 10CFR50 license.

2. YAEC will review and assess the impacts on the following programs and documents in preparation for removal of a survey unit or area from the license:

    - Final Safety Analysis Report and Technical Specifications;
    - Radiological Environmental Monitoring Program;
    - Offsite Dose Calculation Manual;
    - Defueled Emergency Plan;
    - Security Plan;
    - License Termination Plan;
    - Ground Water Monitoring Program;
    - 10CFR100 Siting Criteria; and
    - Decommissioning Environmental Report.

    The reviews will include an assessment to ensure that the land area(s), and any associated building(s), to be released will have no adverse impact on the site's ability to meet the Part 20, Subpart E, criteria for unrestricted release. The reviews will also include the impacts on the discharge of effluents and the limits of 10CFR 20, as they pertain to the public.

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, to summarize 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

1.7 References

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

1-2 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Power Reactors," dated January 1999.

1-3 NUREG-1700, Revision 1, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," dated April 2003.

1-4 NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance," dated September 2003.

1-5 Yankee Rowe Station 2002 Annual Radiological Environmental Operating Report, dated April 28, 2003.

1-6 Yankee Nuclear Power Station Decommissioning Plan, Revision 0.0.

1-7 Letter, M.B. Fairtile (USNRC) to J.A. Kay (YAEC), "Order Approving the Decommissioning of the Yankee Nuclear Power Station, February 14, 1995.

1-8 Letter, M.B Fairtile (USNRC) to J.A. Kay (YAEC), "Completion of Hearing Process Regarding Approval of Decommissioning Plan for the Yankee Nuclear Power Station, October 28, 1996.

1-9 NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," dated June 1992.

1-10 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual," Revision 1, dated August 2000.

1-11 Supplement 1 to NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated November 2002.

1-12 YNPS Decommissioning Environmental Report, dated December 1993.

1-13 "Massachusetts: 2000, Summary Population and Housing Characteristics," U.S. Department of Commerce, issued September 2002.

1-14 "Vermont: 2000, Summary Population and Housing Characteristics," U.S. Department of Commerce, issued October 2002.

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Table 1-1

Current Nearest Resident, Garden, and Milk Animal Locations within 5 Miles of YNPS (Reference 1-5)

Sector	Nearest Resident (mi)	Nearest Garden (mi)	Nearest Milk Animal (mi)
N	3.2	3.7	*
NNE	2.7	3.0	*
NE	2.1	2.1	*
ENE	2.3	3.6	*
E	1.8	2.3	*
ESE	2.1	2.1	*
SE	1.3	2.1	*
SSE	1.2	1.2	*
S	1.3	1.8	*
SSW	*	*	2.0**
SW	0.8	4.5	*
WSW	0.8	1.2	*
W	1.3	1.8	*
WNW	1.3	1.3	*
NW	1.5	2.0	*
NNW	1.8	2.3	*

* No location was identified within 5 miles of the plant.
**Limited number of goats. Not able to supply enough milk for sampling.

Table 1-2

Permanent Population Estimates for Municipalities within 10 Miles of the Yankee Nuclear Power Station

	1980 Census (Ref 1-12)	1990 Census (Ref 1-12)	2000 Census (Ref 1-13 and 1-14)
Massachusetts			
Adams	10,381	9,445	8,809
Clarksburg	1,871	1,745	1,686
Florida	730	732	676
North Adams	18,063	16,797	14,681
Savoy	644	634	705
Buckland	1,864	1,928	1,996
Charlemont	1,149	1,249	1,358
Colrain	1,552	1,757	1,813
Hawley	280	317	336
Heath	482	716	805
Monroe	179	115	93
Rowe	336	387	351
Vermont			
Halifax	488	782	782
Whitingham	1,043	1,298	1,298
Wilmington	1,808	1,968	2,225
Readsboro	638	762	809
Stamford	773	773	813

Figure 1-1

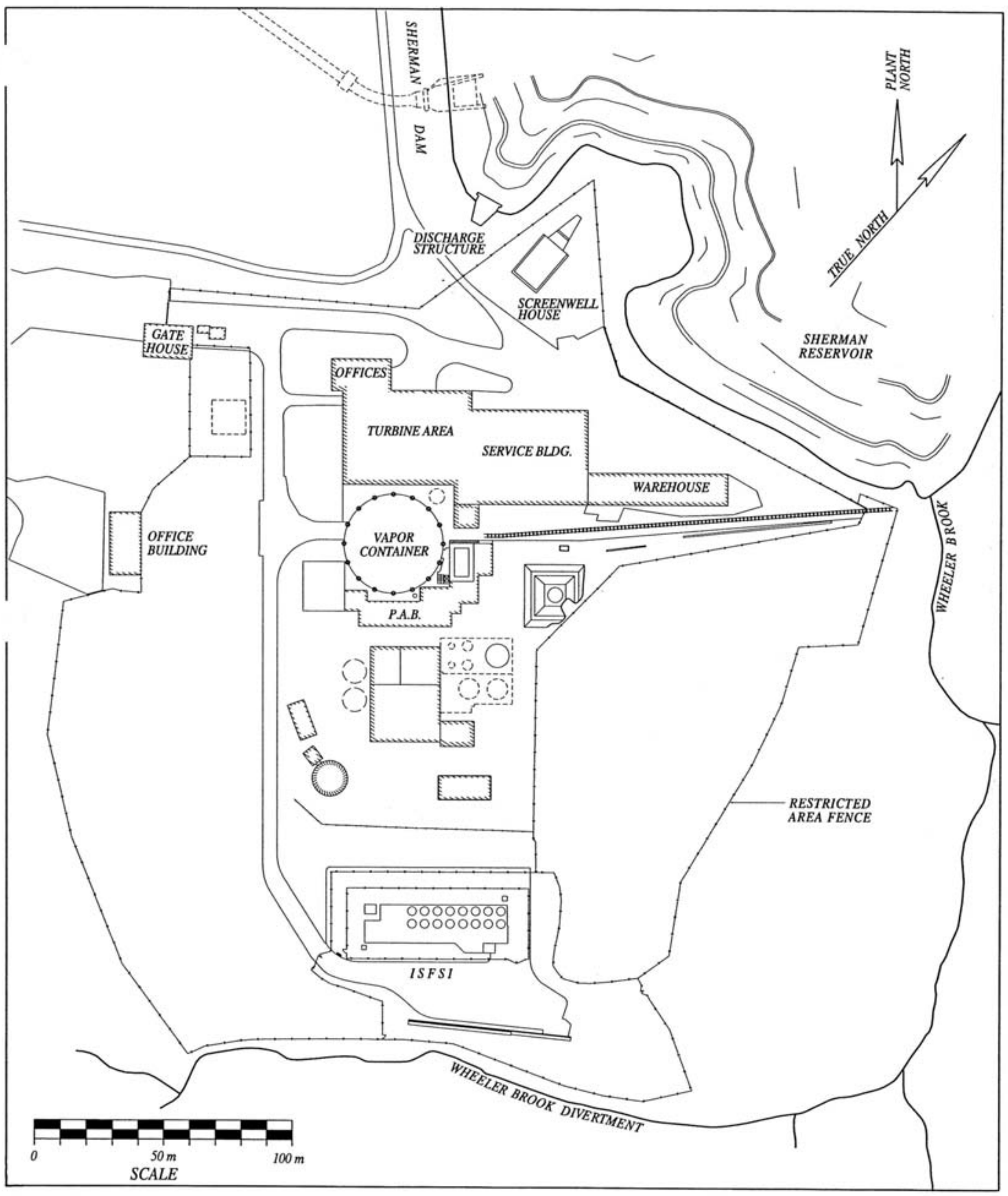
YNPS Site Boundary

(from June 2003 FSAR)

Site Boundary and Plant Exclusion Area

Figure 1-2

Site Design (As of June 2003)



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: Area 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.

Process for Integrating HSA with Characterization and FSS

Review YNPS operational history and identify significant events contributing to contamination of the site.

Establish Survey Area boundaries based on common history.

Assemble available historical reference documentation into Survey Area packages.

Assign Survey Area Classification.

Publish Historical Site Assessment Document.

Obtain Site-Specific DCGL's.

Update Survey Area packages as decommissioning activities progress. Include results of continuing characterization activities and document progress of remediation/ mitigation of sub-surface survey areas.

End-state of decommissioning attained. All areas below appropriate DCGL's.

Evaluate Data Quality in survey area packages using DQO/DQA process and prepare characterization survey plans.

Obtain additional characterization data if necessary and document turn-over surveys.

Begin Final Status Survey process.

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

   - Scoping and characterization surveys performed as part of Decommissioning Plan development
   - Remediation support surveys conducted during decommissioning activities
   - Surveys conducted under the guidance of NUREG/CR-5849 (Reference 2-4)
- The historical evaluations performed for the previously submitted LTP.
- The YAEC Decommissioning Plan
   - Decommissioning Work Plans
   - Secondary Side Work Plans
   - Engineering Change Notifications
   - Field Change Notifications
   - Temporary Change Requests
- The documented radiological end point of decommissioning activities
- Documentation of remediation area stabilization and restoration activities.

2.1.4 Property Inspections

The YNPS site is at an advanced stage of decommissioning with only those plant systems necessary to support the ISFSI and portions of the site remaining in service (e.g., potable water, sanitary sewers, construction electrical power, fire protection and storm sewers). Plant operations, maintenance and security personnel continue to occupy portions of the site in support of the YNPS site operations and maintenance. Due to the advanced state of decommissioning, these activities have a minimal risk of spreading radioactive contamination. The demolition operations contractor occupies a portion of the site with temporary office spaces from which to conduct the decommissioning/demolition activities scheduled for completion during the current phase of decommissioning. These temporary office spaces will be removed from the site at the completion of this phase of decommissioning. The portion of the site historically identified as the Radiation Control Area (RCA) is posted and restricted for personnel access and radioactive material control. RCA access control is maintained through the Radiation Protection (RP) control point.

Decontamination processes have been performed on certain site structures and systems as part of site decommissioning activities under the site Decommissioning Plan. These processes include application of chemical paint strippers, dry ice (carbon dioxide) blasting, steel shot blasting and mechanical removal techniques (including rota-peen tools, needle guns, reciprocating chipping hammers and jackhammers). In addition, both the east and west storm drain system catch basins have routinely been cleaned of accumulated sediment. Sediment socks are now being installed at each catch basin to curtail the build up of sediment in the storm drain system.

Surveys were performed in those areas where decommissioning activities had been completed in accordance with the protocols established under the previously submitted and withdrawn License Termination Plan (Reference 2-5). Controls were instituted and maintained to preserve the radiological condition of most of these areas, and routine surveys are performed in all of these areas to verify that the radiological condition of these areas was not adversely impacted by ongoing plant operation, maintenance, or fuel transfer activities.

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 criterion 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 identified any potential radionuclide translocation pathways that impacted that area or 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.

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 License Termination Plan (LTP) for YNPS, pursuant to 10CFR50.82(a)(9). The initial YNPS

LTP employed a survey methodology based upon NUREG/CR-5849. Subsequently the NRC, jointly with the DOD, DOE, and EPA, approved an alternate survey methodology documented in NUREG-1575 (Reference 2-2). In May 1999, Yankee advised the NRC that it intended to shift from the survey methodology in NUREG/CR-5849 to the NUREG-1575 methodology, and withdrew its previously submitted LTP application.

In 2000, Yankee created a Post-Shutdown Decommissioning Activities Report (PSDAR) within the Final Safety Analysis Report (FSAR). NRC Draft Regulatory Guide DG-1071 recommends that licensees with approved Decommissioning Plans (D Plans) "extract pertinent detail from the decommissioning plan and submit a PSDAR update in the format and content specified by [DG-1071]." Based on the NRC draft guidance, Yankee segregated, updated and condensed certain information concerning post-shutdown decommissioning activities in a manner that conforms to the standard format and content of a PSDAR. The current LTP is written to reflect the NUREG-1575 (MARSSIM) methodology, as well as regulatory guidance made available since the previous LTP submittal.

Decommissioning activities completed as of May 1997 had removed the majority of systems and components not required to support the storage of spent fuel in the spent fuel pool. Detailed planning for the transfer of spent fuel from the Spent Fuel Pit began in February 2000. In June 2003 the transfer of all fuel and Greater Than Class "C" waste from the Spent Fuel Pit to the ISFSI was completed.

2.2.2 Regulatory Involvement

The NRC monitors YNPS site activities using inspectors from Region I offices to perform onsite inspections. Periodic calls are also held with NRC headquarters and Region I staff to monitor plant status and decommissioning progress. The NRC is notified of any incidents on site per the existing protocol established with NRC Region I and NRC reporting regulations.

The decommissioning of the YNPS site is also being performed under various Federal, State and local requirements in addition to the NRC regulations. For example, YNPS is subject to 29 CFR 1910 and 1926 (Reference 2-7) for worker health and safety protection under OSHA regulations. Asbestos and lead-based paint handling and removal are subject to OSHA regulations cited above, and EPA regulations 40 CFR Part 61, Subpart M (Reference 2-8). State and EPA requirements will be met for PCB paint removal activities. YNPS will also be required to meet the state standards for surface water and groundwater.

The Commonwealth of Massachusetts Department of Public Health also has state radiological remediation standards. Compliance with the state standards is not addressed in this document. This issue will be addressed in separate correspondence with the Commonwealth.

Permits and approvals from, or notifications to, several State (Commonwealth) and local agencies are required for safety and environmental protection purposes. Some of these are for 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, contributed to the residual contamination that needs to be address during 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 of 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.

- In the mid 1970s YNPS converted from stainless steel to zirconium clad fuel pins. Some of the zirconium fuel pins failed in the reactor due to vibrational stress from water jetting.

The pin failure resulted in a release of fuel pellets directly into the reactor coolant system. This event changed the isotopic mix within the Reactor Coolant System. In particular, detectable quantities of fission products such as Cs-137 and Cs-134 were dispersed throughout the primary side plant systems and the fuel handling facility for the first time in the plant operating history.

- During a refueling outage in 1981, while relocating the reactor head to its outside storage location, the reactor head made contact with the wall above the equipment hatch in the Vapor Container. The impact dislodged particulate radioactivity adhered to the under side of the reactor head. This resulted in contamination of the RCA yard area under and around the equipment hatch.

- Construction of the original PCA storage facility included a PVC drainpipe that connected the PCA storage building to the Waste Disposal Building. The PVC pipe joints failed allowing liquid to flow from the drainpipe into the surrounding soil.

- The use of an underwater plasma torch to section the reactor internals resulted in the release of highly radioactive cutting debris into the shield tank cavity shield water. This changed the radionuclide mix of the residual contamination in the shield tank cavity and, to a certain extent, in the Spent Fuel Pit.

2.2.4 History of Unplanned Events

As part of the HSA, a comprehensive review of all recorded events documented as having occurred outside the normal operational condition was performed to capture those events which contributed to the contamination of the site. These events were typically documented in the format suitable for reporting to regulatory authorities such as Abnormal Occurrence Reports (AOR's), submitted during the early site history, and Plant Incident Reports (PIR's) or Licensee Event Reports (LER's), submitted through the remainder of plant life. Where available, the information in these reports was supplemented by supporting documentation concerning the events in the form of plant memos and radiological survey data.

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 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, fewer 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, the associated MARSSIM classifications, and 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 performed 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 Impacted Area Assessments

The summary assessments provided in Appendices 2B and 2C of this section include a description, key elements of the history, contaminated media and an evaluation of the principle radionuclides expected to be present in the area. The summary also includes a current decommissioning status and a description of the work remaining to be done to attain the anticipated end-state. A survey area classification statement is provided at the end of each assessment. None of the impacted areas were classified based on the results of scoping or preliminary characterization data. The classifications assigned, based on historical activities performed in these survey areas alone, are substantiated by the large quantity of scoping data available in the form of soil sample analyses and survey data. Summaries of the sampling data as shown on Tables 2-4 and 2-5 are compiled from information detailed in the YNPS HSA. More detailed descriptions, histories and the radiological status of each of these survey areas are also contained within the YNPS HSA.

2.4.1 Buildings, Structures and Open Land Areas Inside the RCA

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

AUX	Primary Auxiliary Building
BRT	Vapor Container Support Structure (sub-surface)
NOL	Open Land Areas Inside the RCA
NSY	Yard Structures Inside the RCA
WST	Waste Disposal Building

Summary individual survey area assessments are described in Appendix 2B. In general, all survey areas within the confines of the historical RCA have been assigned a Class 1 status. The exceptions are NSY-10 and NOL-07 which are the ISFSI Pad and the open land area immediately surrounding this structure. The area was excavated to prepare a suitable surface for the new concrete pad structure. The soils removed from this excavation were evaluated by composite sampling and found to contain only naturally occurring radionuclides. The pad and surrounding land have been assigned a Class 3 status pending further evaluations following the final disposition of the spent fuel containers.

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 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 because 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).

2.5.2 Decommissioning Activities

There were no decommissioning or remediation activities performed in the non-impacted area. Most of the area is forested. The power line right-of-way is cleared of trees.

2.5.3 Basis of Area Classification

The survey unit is classified as "non-impacted" because there is no reasonable possibility of residual contamination based upon the following (References 2-11, 2-12 and 2-13):

- Samples collected as part of the Radiological Environmental Monitoring Program (REMP) throughout the plant's operational and post-operational history show no evidence of any significant radiological impact due to plant operations;

- Aerial photographs from 1966, 1970, 1974, 1980, 1981, 1982, 1989, and 1990 show no evidence of soil disturbance;

- A conservative evaluation of the impact of particulate effluents to soils outside of the industrial area using a Gaussian dispersion/deposition model substantiates the conclusion that this source of plant-derived radioactive material would be expected to contribute (at a maximum) a very small fraction of the DCGL. Beyond the impacted area boundary, concentrations of this plant-derived radioactive material would be non-detectable and indistinguishable from background;

- A statistical comparison of soil sample analytical data from the non-impacted area and an environmentally equivalent reference area (unaffected by plant releases) was performed.

2.5.4 Occurrence of Anthropogenic Radionuclides in the Environmental Background

According to the National Council on Radiation Protection and Measurements (References 2-14, 2-15 and 2-16), radionuclides present in environmental background are both naturally occurring and man-made. Carbon-14 is introduced cosmogenically and by the atmospheric detonation of nuclear weapons. Tritium is also introduced cosmogenically and through atmospheric detonation of nuclear weapons. Cesium-137 and Strontium-90 are fission products that occur in the environment as a result of atmospheric nuclear weapon detonations.

The range of concentrations of Cs-137 in environmental background due to fall-out from atmospheric atomic device testing is easily detectable in soil. Both Cs-137 and Sr-90 are fission products with similar half-lives. Accordingly, it is expected that Sr-90 due to fall-out from atmospheric testing would also occur in the environment where weapons derived Cs-137 is present.

2.5.5 Evaluation of the Impact of Elevated Releases of Particulate Radioactive Material

Covering the operating history of YNPS, YRC-1178 (Reference 2-11) provides a conservative evaluation of the deposition of particulate activity in gaseous effluents on soils in the impacted area downwind of the Primary Vent Stack (OOL-08). The study examined Semi-Annual Effluent Reports and Monthly Operating Reports that contain the total activity, by radionuclide, released from the plant in particulate form of gaseous effluents. The particulate fraction released from the Primary Vent Stack is determined from analyses of the waste gas discharge. The gaseous fraction of the effluent was disregarded when considering the impact to soils since there is no expectation that this fraction would be deposited. The individual radionuclide activity annual data were decay-corrected to the time of YRC-1178 (1998). A conservative atmospheric deposition factor was developed and applied to the decay-corrected particulate fraction of released activity to determine the maximum residual deposition on an area extending 100-200 meters beyond the industrial area boundary. The long-term average deposition factor was derived from plant specific meteorological and structural data and was determined to be $8.79E\text{-}08m^{-2}$. Soil radioactivity concentrations based on a penetration depth of 15 cm and a density of 1.6 gm/cc were calculated to be:

- Sr-90: 2.56E-4 pCi/g
- Cs-134: 4.91E-7 pCi/g
- Cs-137: 1.01E-4 pCi/g
- Co-60: 1.31E-4 pCi/g

These values are below the expected site-specific DCGLs and minimum detectable activities (MDAs). These projections demonstrate that the concentration of gaseous effluent-derived radioactive material in area OOL-08 (an impacted area) is expected to be much less than the soil DCGLs. Since the non-impacted area is further from the source, plant-derived radioactive material concentrations would be even lower than those typical of survey area OOL-08.

2.5.6 Statistical Evaluations

2.5.6.1 Description of Reference Areas

Cesium-137 derived from atmospheric nuclear weapon detonations occurs in all land areas, regardless of their proximity to YNPS. In order to assess properly the impact of plant operations alone on the non-impacted area of the site, the contribution from this source of Cs-137 must be quantified. To that end, reference areas that were not reasonably expected to contain plant-derived Cs-137 were identified. Reference 2-17 describes the selection criteria, sampling protocol, and summary results for these reference background areas. The areas selected were in the vicinity surrounding Pelham Lake. This area was selected for the following reasons:

- It is the direction of least prevalent winds, and therefore has the least likelihood of having been impacted by YNPS air effluents.

- It is in a separate valley and there is no known surface or groundwater communication between the two valleys. Therefore it most likely has not been impacted by liquid effluents.
- It has soil and flora typical of the non-impacted survey area surrounding the YNPS site.

2.5.6.2 Approach and Methodology for Evaluation of the Non-Impacted Area

Thirty (30) surface soil samples were collected from the non-impacted area in August 1998. The locations of each sample point and the general location of the plant site relative to the survey area are presented in Figure 2-5. Sixty surface soil samples were also obtained (in 1996) from a selected reference area beyond the boundaries of the YNPS-owned property as described in Section 2.5.6.1. The means and maximum values of the reference background area and the non-impacted areas compare favorably with the global concentrations of Cs-137 found from atmospheric deposition in topsoil.

Two types of statistical tests were performed to evaluate whether the soils from the non-impacted area contain excess Cs-137 relative to the soil samples from the reference area. These analyses are presented in Reference 2-11. The Student t-test was used to compare the mean values of the two data sets. The second test was a single-tailed Fisher's "F-Test" of the variances of the Cs-137 concentrations in the reference area and the non-impacted area. This comparison is also known as the Analysis of Variance or the Variance Ratio. The test compares the variances of both data sets.

Additional statistical analyses were performed on the shapes of the sample distribution to provide additional evidence that these two distributions may have the same source. These were tests for skewness and normality. These tests indicated that the parameters for the data sets are alike.

2.5.7 Summary

The classification of the area as non-impacted is based upon historical photographs, results of Radiological Environmental Monitoring Program surveys, particulate gaseous effluent deposition modeling and a statistical analysis of Cs-137 soil concentrations relative to a set of background reference areas.

2.6 Continuing Investigation of Subsurface Contamination

Subsurface radioactivity is residual radioactivity that is underneath structures such as building floors/foundations or that is covered with soil or some other material. Some areas known to be impacted are still under investigation. The reasons for this vary. Survey area information, as presented in the YNPS HSA, is the primary resource for identifying areas that may require subsurface investigation.

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. Finding activity in 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 the activity in soil from peripheral cores is 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, 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, is shown 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 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 the standing water that was exposed during removal of the concrete floor 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 allow sampling of that area.

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 further 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 typically applied to the FSS plan design will be applied to this survey.

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.

The survey design includes the selection of instruments and techniques needed to provide scans, static measurements, and samples of the proper quality and quantity to allow decisions to be made regarding the suitability of the current MARSSIM area classification. Technical basis documents will be developed as needed to justify the use of the measurement methods and to assess instrument detection limits.

Approved site procedures for field and laboratory instrument calibration and operation, survey techniques and reporting, data entry and management, and training and qualification of personnel will ensure that the plan is implemented consistently and according to applicable standards.

2.8.3 Characterization Survey Plans

The purpose of a Characterization Survey Plan is to describe the methods to be used in the planning, design, execution, and evaluation of characterization surveys. The “as found” condition of a given survey area is documented in the survey area classification packages. These packages contain sufficiently detailed information on the operational history and current decommissioning status to allow generation of a Characterization Survey Plan or to use the existing data provided it is qualified to be adequate as characterization data. If the completed classification package indicates that additional characterization is required to investigate potential presence of plant-derived radionuclides on the exterior of sub-grade surfaces or beneath the concrete floor of the end state structure, the results of such investigations will be included in the survey area classification information.

2.9 References

2-1. YAEC Historical Site Assessment.

2-2 NUREG-1575: Multi-Agency Radiation Survey and Site Investigation Manual, Revision 1, dated August 2000.

2-3 YAEC Deed Study Project Rowe and Monroe, Massachusetts, dated December 18, 1998.

2-4 DRAFT NUREG/CR-5849 (ORAU 92/C57): "Manual for Conducting Radiological Surveys in Support of License Termination," by J.D. Berger, dated June 1992.

2-5 YAEC License Termination Plan, dated December 1997.

2-6 YNPS Decommissioning Plan, dated March 29, 1994

2-7 Title 29 Code of Federal Regulations, “Labor.”

2-8 Title 40 Code of Federal Regulations, “Protection of Environment.”

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 2-2

Area of Open Land Survey Areas

SURVEY AREA	DESCRIPTION	MARSSIM CLASS	AREA (m^2)
OOL-01	SHERMAN POND SEDIMENTS	3	73971
OOL-02	YANKEE NON-RAD YARD AREAS	3	7134
OOL-03	SHERMAN RESERVOIR DAM AND SOUTH SHORELINE	3	16177
OOL-04	US GENERATION / SHERMAN STATION OVERLYING GROUNDWATER PLUME	3	17870
OOL-05	US GENERATION / DEERFIELD RIVER FRONTAGE	3	28574
OOL-06	YANKEE WESTERN ACCESS	3	37281
OOL-07	SOILS DEPOSIT AREA	2	2108
OOL-08	YANKEE SITE EXCLUSION ZONE	3	133368
OOL-09	SOUTHEAST CONSTRUCTION FILL AREA	3	2387
OOL-10	ISFSI/ACCESS, EXCLUSION ZONE, BUFFER ZONE	2	8408
OOL-11	EAST RCA BUFFER ZONE	2	1220
OOL-12	WAREHOUSE RAIL SPUR	1	876
OOL-13	US GENERATION/RAIL SPUR TERMINUS	1	1148
OOL-14	US GENERATION/WHEELER BROOK FRONTAGE	3	2354
OOL-15	US GENERATION/SHERMAN RESERVOIR EAST SHORELINE	3	4662
OOL-16	FURLON HOUSE PARKING LOT	3	2481
OOL-17	ASPHALT, BRICK AND CONCRETE STORAGE YARD	3	3247
NOL-01	EASTERN LOWER RCA YARD	1	1364
NOL-02	NORTHEASTERN UPPER RCA YARD	1	1990
NOL-03	SOUTHEASTERN UPPER RCA YARD	1	1575
NOL-04	SOUTHWESTERN UPPER RCA YARD	1	1753
NOL-05	NORTHWESTERN UPPER RCA YARD	1	1586
NOL-06	WESTERN LOWER RCA YARD	1	1329
NOL-07	ISFSI RCA YARD	3	1717

Table 2-3

AOR / PIR List of Unplanned Liquid Releases

Impacted Survey Area	AOR/ PIR #	Description
NOL-2/NOL-5	61-15	Radioactive Spill – 9/20/61
NOL-1/NOL-2 and NSY-2	63-12	Shield Tank Cavity Fill Water Spill – 9/18/63
OOL-5/OOL-6	63-17	De-watering Pump Packing Leakage – 10/8/63
AUX-1	64-08	Seal Water Tank Spill – 9/3/64
NOL-1/NSY-2 and OOL-5/OOL-6	64-13	IX Pit High Level – Leakage Coming Up through Pavement[1]– 10/3/64
SFP-1/NOL-1/ OOL-1	66-07	Spent Fuel Pit Water Spill – 9/27/66
OOL-5/OOL-6	66-08	Abnormal Activity in Storm Drain – 9/27/66
NOL-1/OOL-1	66-09	Hose Failure – 11/1/66
NSY-7	68-01	Waste Hold-up Tank Moat Spill – 1/16/68
NOL-1 thru 6	75-07	Yard Area Contamination – 7/16/75
NOL-2	77-16	Service Building Radioactive Sump Transfer Line Puncture – 12/21/77
NOL-2/NSY-2	80-09	Resin Spill – 8/6/80
NOL-1/NOL-6 OOL-12/OOL-13 and OOL-1	81-09	Contamination of Yard Area During Rx Head Removal – 5/15/81
WST-1/WST-2 and WST-3	84-16	Drain Pipe Failure[2] – 9/10/84
NOL-1	94-03	Leakage from Frozen Fuel Chute Dewatering Line 2/17&18/94
NOL-1	94-09	NST Tell-Tales/Fuel Chute Dewatering Line 2/23/94

[1] Routine leakage points, paths for subsurface contamination.

TABLE 2-4

Current Radiological Conditions of Buildings in the Industrial Area by Survey Area

Survey area	Description	Nominal exposure rate (μr/hr)	Nominal loose surface contamination (dpm/100cm^2)
SVC-01	NORTH PART OF SERVICE BLDG (CLEAN SIDE)	8	<1000
SVC-02	RAD PORTIONS OF SERVICE BLDG AND ANNEX	8	<1000
SVC-03	CLEAN SIDE OF SERVICE BLDG ANNEX	6	<1000
TBN-01	TURBINE BUILDING AND OFFICES	10	,<1000
SFP-01	**SPENT FUEL PIT AND TRANSFER CHUTE**	**500-10,000**	**300-8700**
SFP-02	**NEW FUEL VAULT**	**100-5000**	**<1000**
BRT-01	CONCRETE PEDESTALS, PAD AND ANNULUS	15	<1000
NSY-01	**NORTH AND SOUTH DECON PADS AND FTE**	**20-700**	**>1000**
NSY-02	IX-PIT, VALVE GALLERY/ PAB STAIRWAY	300	<1000
NSY-03	SI DIESEL/ACCUMULATOR TANK/BATTERY PADS	11	<1000
NSY-04	SAFE SHUTDOWN SYSTEM BUILDING	10	<1000
NSY-05	FIRE WATER TANK AND PUMP HOUSE	13	<1000
NSY-06	PCA#2 (NEW)	10	<1000
NSY-07	WHT / ADT / WASTE GAS PADS	40	<1000
NSY-08	NEW SI TANK	20	<1000
NSY-09	**ELEVATOR SHAFT**	**500**	**<1000**
NSY-10	**ISFSI**	**2000-5000**	**<1000**
NSY-11	**CHEM-WASTE TRANSFER PUMP PIT**	**2000 - 15000**	**>1000**
NSY-12	TANK #1 BASE AND PIPECHASE	15	<1000
NSY-13	**DEMIN WATER STORAGE TANK #39 BASE**	**2500**	**<1000**
WST-01	**PCA #1 (OLD)**	**30-200**	**<1000**
WST-02	**PCA WAREHOUSE**	**60-150**	**>1000**
WST-03	WASTE DISPOSAL BLDG	15	<1000
WST-04	COMPCTOR BLDG	20	<1000
AUX-01	PAB/ EAST END	10	<1000
AUX-02	PAB / WEST END	10	<1000

Survey area	Description	Nominal exposure rate (ur/hr)	Nominal loose surface contamination (dpm/100cm^2)
OMB-01	PUMPHOUSE AND SCREENWELL	11	<1000
OMB-02	SECURITY GATEHOUSE AND DIESEL GENERATOR	6	< 1000
OMB-03	ADMINISTRATION BUILDING	No data	<1000
OMB-04	WAREHOUSE AND LOADING DOCK	6	<1000
OMB-05	FURLON HOUSE	No Data	<1000
OMB-06	SEAL PIT	No Data	<1000

Note: The entries in **BOLD** in the table are either currently in use or the reported exposure rates are influenced by adjacent buildings or tanks that are currently in use.

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

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Table 2-7

Well Depths and Sampling Results

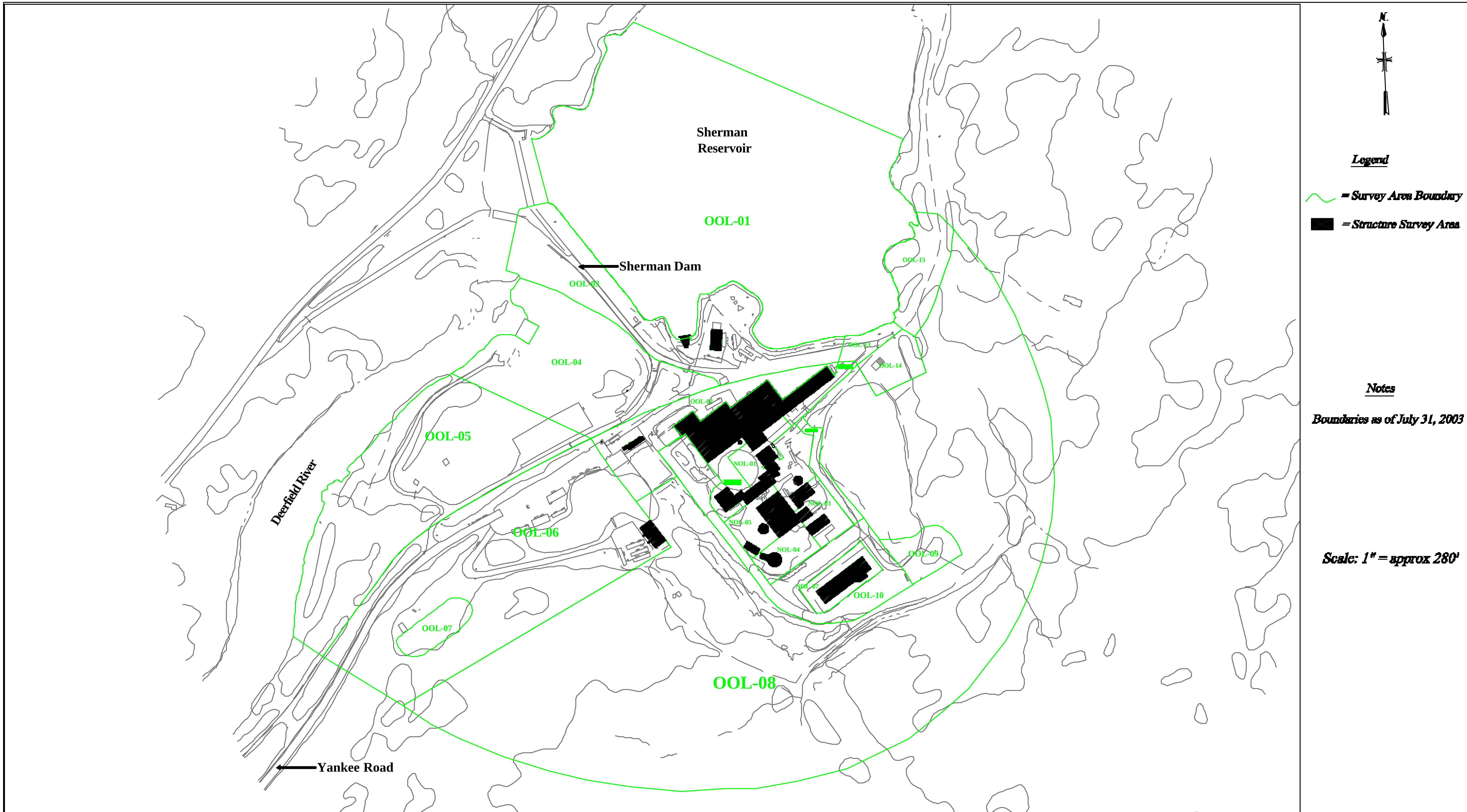
Well no.	Well Type*	Depth of Well (feet)	3rd Quarter 2003 Results (pCi/l)			4th Quarter 2003 Results (pCi/l)			1st Quarter 2004 Results (pCi/l)			2nd Quarter 2004 Results (pCi/l)		
			H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta
B-1	Intermediate Bedrock	79	1.36E03	2.80E00	9.16E00	9.00E02	-	6.53E00	9.60E02	*	6.5E00	*	*	9.0E00
CB-1	Shallow Intermediate	25	1.76E03	-	1.35E01	2.14E03	-	1.26E01	1.83E03	*	1.35E01	1.16E03	*	1.14E01
CB-2	Shallow Intermediate	24.5	4.11E02	-	1.62E01	1.16E03	-	1.18E01	5.33E02	*	9.96E00	8.90E02	*	1.00E01
CB-3	Shallow	13	-	4.50E00	2.48E01	-	-	-	*	2.6E00	1.69E01	-	-	-
CB-4	Shallow	19	-	-	1.41E01	-	-	8.20E00	*	*	6.24E00	4.00E02	*	7.7E00
CB-5	Intermediate	59	-	1.54E00	2.44E00	-	-	-	*	*	4.3E00	-	-	-
CB-6	Shallow	25	-	-	1.90E01	4.30E02	-	1.14E01	2.79E02	*	1.06E01	*	*	8.3E00
CB-7	Shallow	17	-	-	2.60E01	-	-	-	*	*	1.28E01	-	-	-
CB-8	Shallow Intermediate	19	-	3.90E00	1.32E01	-	-	-	*	*	1.77E01	-	-	-
CB-9	Shallow Intermediate	24	2.33E03	-	6.70E00	2.62E03	-	7.60E00	2.40E03	*	7.0E00	1.74E03	*	8.3E00
CB-10	Shallow	11	9.00E02	-	1.91E01	1.21E03	-	1.25E01	8.60E02	1.17E01	5.17E01	8.00E02	3.4E00	2.00E01
CB-11A	Shallow	20	-	-	1.31E01	2.12E03	8.70E00	3.30E01	-	-	-	1.68E03	*	1.39E01
CB-12	Shallow	7	-	6.80E00	2.81E01	5.40E02	-	1.05E01	*	2.8E00	1.56E01	*	2.0E00	1.58E01
CW-1	Shallow Intermediate	21	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
CW-2	Shallow	20	-	9.20E00	4.25E01	-	-	-	*	*	1.43E01	-	-	-
CW-3	Intermediate and Bedrock	23	-	-	1.83E01	1.62E02	-	5.91E01	*	*	2.76E01	*	3.5E00	1.39E01
CW-4	Shallow Intermediate	17	-	-	1.77E01	-	-	-	2.92E02	1.12E00	1.14E01	-	-	-
CW-5	Shallow and Bedrock	16.5	-	-	1.28E01	-	3.50E00	6.60E00	-	-	-	*	*	5.38E00
CW-6	Shallow	22	-	-	1.10E01	1.58E02	-	4.01E00	*	3.4E00	9.1E00	3.2E02	2.1E00	6.66E00
CW-7	Shallow Intermediate	31	-	2.50E00	1.13E01	-	-	-	2.02E03	*	8.71E00	6.10E02	*	9.7E00
CW-8	Shallow Intermediate	26	-	-	1.11E01	-	-	-	4.00E02	*	7.00E00	-	-	-
CW-9	Shallow Intermediate	17	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
CW-10	Bedrock	30	-	4.20E00	1.16E01	-	-	-	*	5.9E00	1.45E01	-	-	-
CW-11	Shallow	9	3.67E03	-	8.60E00	1.85E03	-	1.02E01	-	-	-	5.20E03	2.5E00	8.4E00
DW-1	Bedrock	280	-	-	3.89E00	-	-	-	*	1.42E00	1.5E01	*	1.28E00	6.20E00
MW-1	Shallow Intermediate	21	-	3.30E00	3.39E01	5.80E02	-	2.21E01	*	*	3.4E00	*	*	4.16E00
MW-2	Shallow	17	1.25E03	-	8.30E00	1.78E03	-	1.11E01	1.25E03	*	3.39E01	1.87E03	*	1.85E01
MW-3	Shallow	20	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	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	4.49E03	*	1.05E01	3.42E03	*	8.4E00
MW-6	No log available	17	-	5.64E00	1.05E01	2.14E02	3.42E00	8.90E00	4.77E00	4.8E00	1.3E01	*	4.8E00	1.30E01

* Denotes value below the critical level for 1st and 2nd quarter 2004 data.
† Well has been closed and grouted over, and thus are no longer available for sampling.

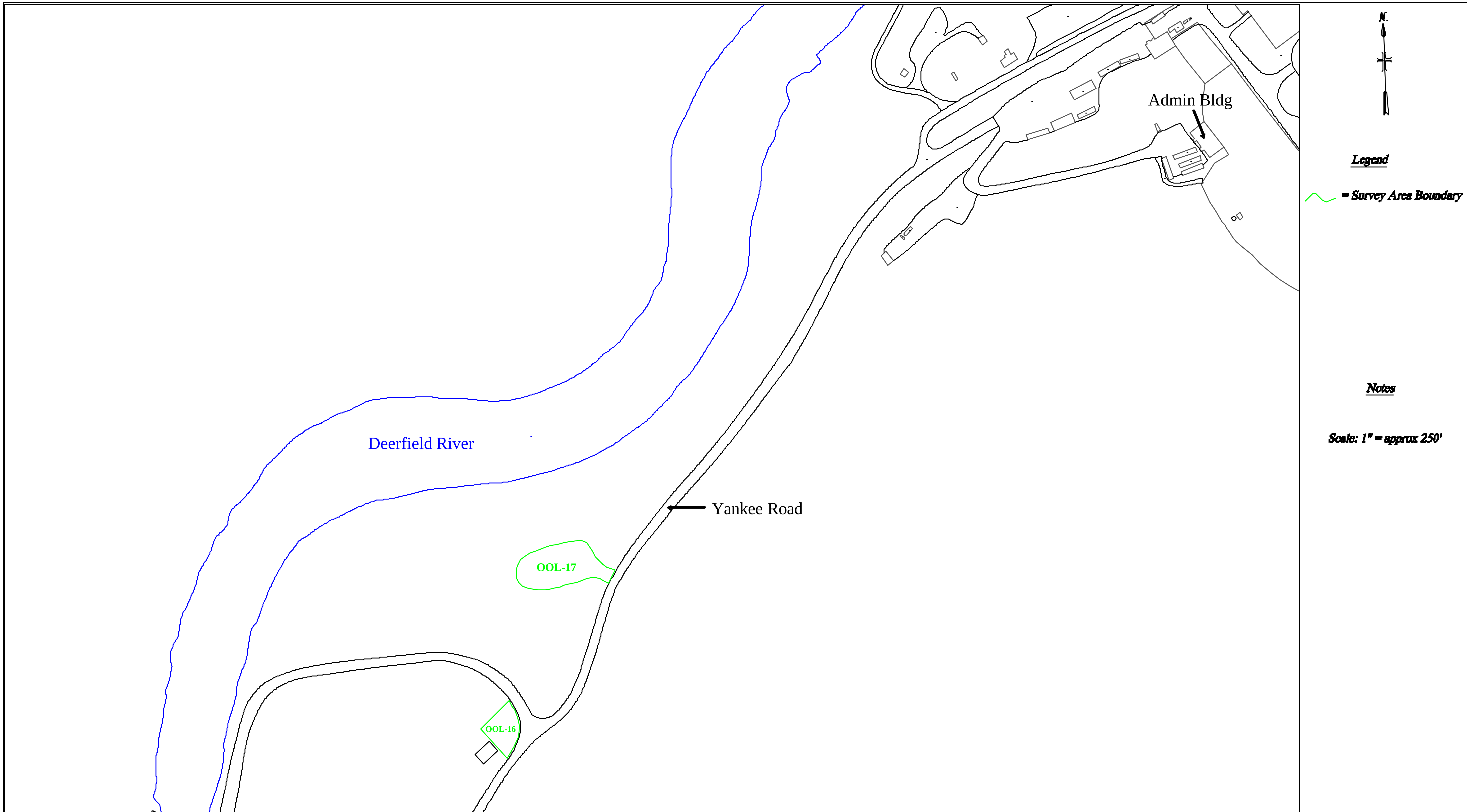
Table 2-7

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)			1st Quarter 2004 Results (pCi/l)			2nd Quarter 2004 Results (pCi/l)		
			H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta	H-3	Gross Alpha	Gross Beta
NSR-1	Shallow and Bedrock	23	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†	N/A†
OSR-1	Shallow	13	-	-	7.50E00	-	-	-	*	*	4.0E00	*	*	7.2E00
CFW-1	No log available	8	-	-	2.97E00	2.66E02	1.97E00	-	*	1.34E00	2.94E00	*	1.22E00	3.34E00
CFW-2	No log available	20	-	-	7.37E00	-	-	3.10E00	*	*	3.03E01	*	1.61E00	4.9E00
CFW-3	No log available	34	-	-	6.44E00	-	1.93E00	9.68E00	*	*	1.01E01	*	3.9E00	8.4E00
CFW-4	No log available	53	-	2.70E00	6.70E00	-	2.50E00	8.80E00	*	2.14E00	5.3E00	*	2.01E00	4.8E00
CFW-5	No log available	5	-	-	4.80E00	-	2.20E00	5.20E00	*	1.56E00	1.91E01	*	1.61E00	9.2E00
CFW-6	No log available	6	-	-	4.70E00	-	-	2.30E00	*	1.37E00	7.2E00	*	*	3.4E00
CFW-7	No log available	Not known	-	-	7.60E00	-	1.70E00	2.60E00	*	*	5.2E00	*	1.11E00	3.7E00
MW-100A	Shallow	20	-	3.70E00	1.02E01	-	-	-	-	-	-	*	1.38E00	4.55E00
MW-100B	Bedrock	43	2.50E02	3.30E00	1.31E01	-	-	-	-	-	-	*	1.81E00	2.3E00
MW-101B	Bedrock	152	-	-	3.90E00	2.52E2	3.15E00	1.27E01	1.07E02	*	1.91E01	*	4.3E00	2.26E01
MW-101C	Intermediate	99	-	9.20E00	2.58E01	-	-	9.50E00	2.64E02	*	1.73E01	*	*	1.08E01
MW-102A	Shallow	38	4.58E03	-	4.80E00	4.91E03	-	2.71E00	4.76E03	*	4.53E00	4.60E03	*	4.8E00
MW-102B	Bedrock	130	3.90E02	-	5.20E00	-	1.60E00	5.15E00	*	*	7.1E00	*	*	2.87E00
MW-102C	Intermediate	99	5.75E03	-	5.20E00	6.59E03	2.13E00	3.42E00	-	-	-	6.63E03	1.07E00	4.12E00
MW-103A	Shallow	25	3.50E02	4.20E00	1.28E01	-	-	9.35E00	2.11E02	*	1.26E01	*	3.8E00	6.6E00
MW-103B	Bedrock	295	-	4.10E00	8.90E00	-	1.79E00	1.10E01	*	2.0E00	8.2E00	4.10E02	2.29E00	5.62E00
MW-103C	Intermediate	125	2.70E02	2.07E00	1.07E01	-	5.10E00	9.30E00	*	*	7.68E00	*	1.70E00	6.4E00
MW-104B	Bedrock	194	-	-	-	-	-	1.13E01	*	1.58E00	1.30E01	*	*	7.5E00
MW-104C	Intermediate	97	-	-	-	-	-	7.20E00	*	*	1.08E01	3.20E02	*	2.9E00
MW-105B	Bedrock	74	4.85E03	-	1.13E01	5.22E03	5.50E00	1.28E01	4.89E03	3.26E00	1.58E01	4.53E03	3.17E00	1.4E01
MW-105C	Intermediate	37	1.86E03	-	9.32E00	3.72E03	2.50E00	8.20E00	3.96E03	*	5.3E00	2.06E03	*	7.5E00
MW-107B	Bedrock	110	<2.00E03‡	-	-	-	2.70E00	1.05E01	*	8.6E00	2.35E01	*	*	1.06E01
MW-107C	Intermediate	32	4.8E04	-	-	4.58E04	-	5.00E00	8.88E03	*	1.55E01	3.90E04	*	7.8E00
MW-107D	Intermediate	80	9.15E03	-	-	9.71E03	-	1.12E01	5.94E03	8.9E00	1.63E01	1.09E04	*	6.3E00



Yankee Atomic Power Company *Preliminary Land Surface Impacted Area Boundaries*	YANKEE	*Date: November 2003* *Revision: 5*	*Figure: 2-1a*



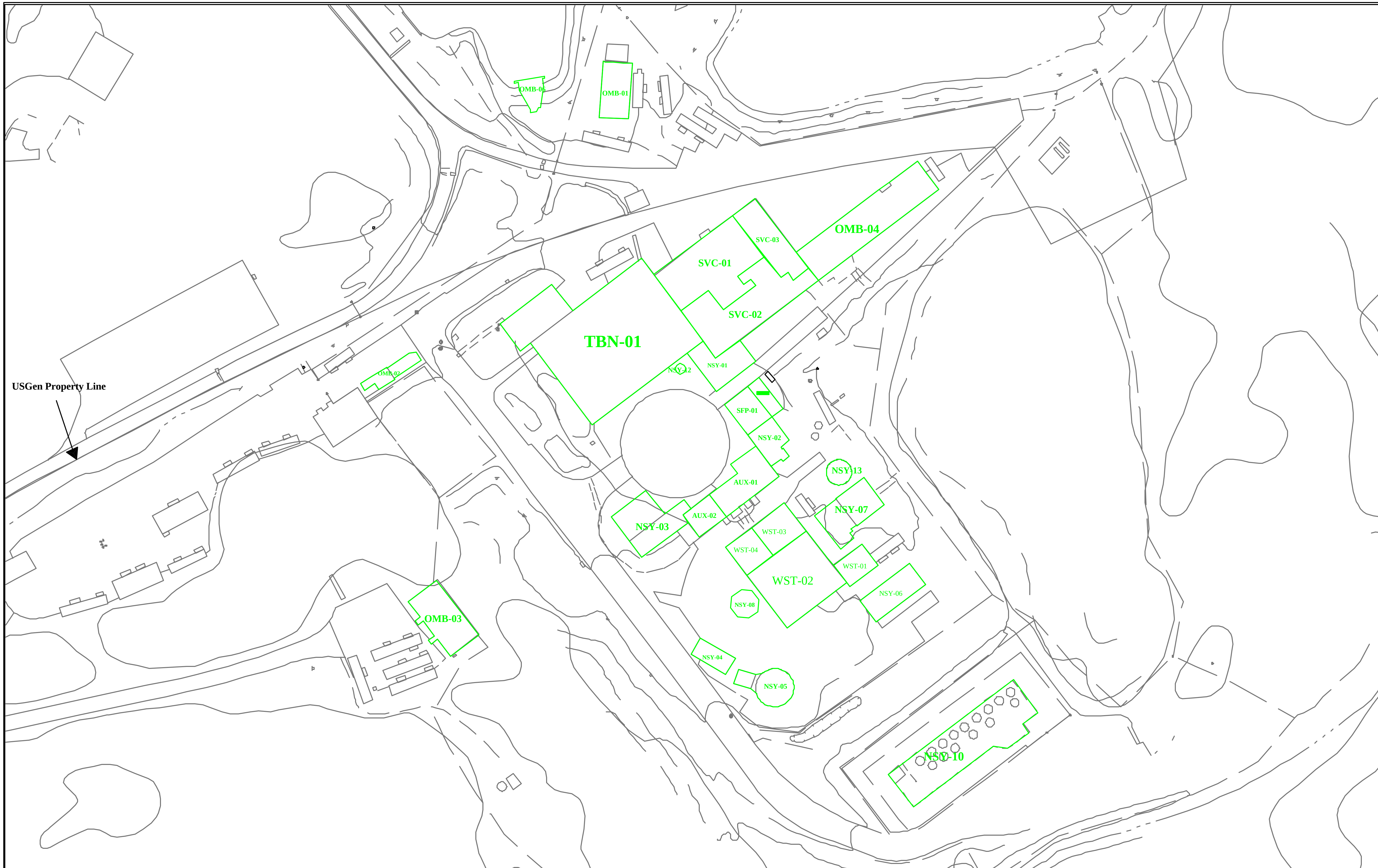
Yankee Atomic Power Company
Preliminary Land Surface Impacted Area Boundaries

YANKEE

Date: November 2003

Revision: 1

Figure: 2-1b



N.

Legend

= Survey Area Boundary

Notes

Boundaries as of July 31, 2003

Scale: 1" = approx 125'

NOTE:
Furlon House not shown

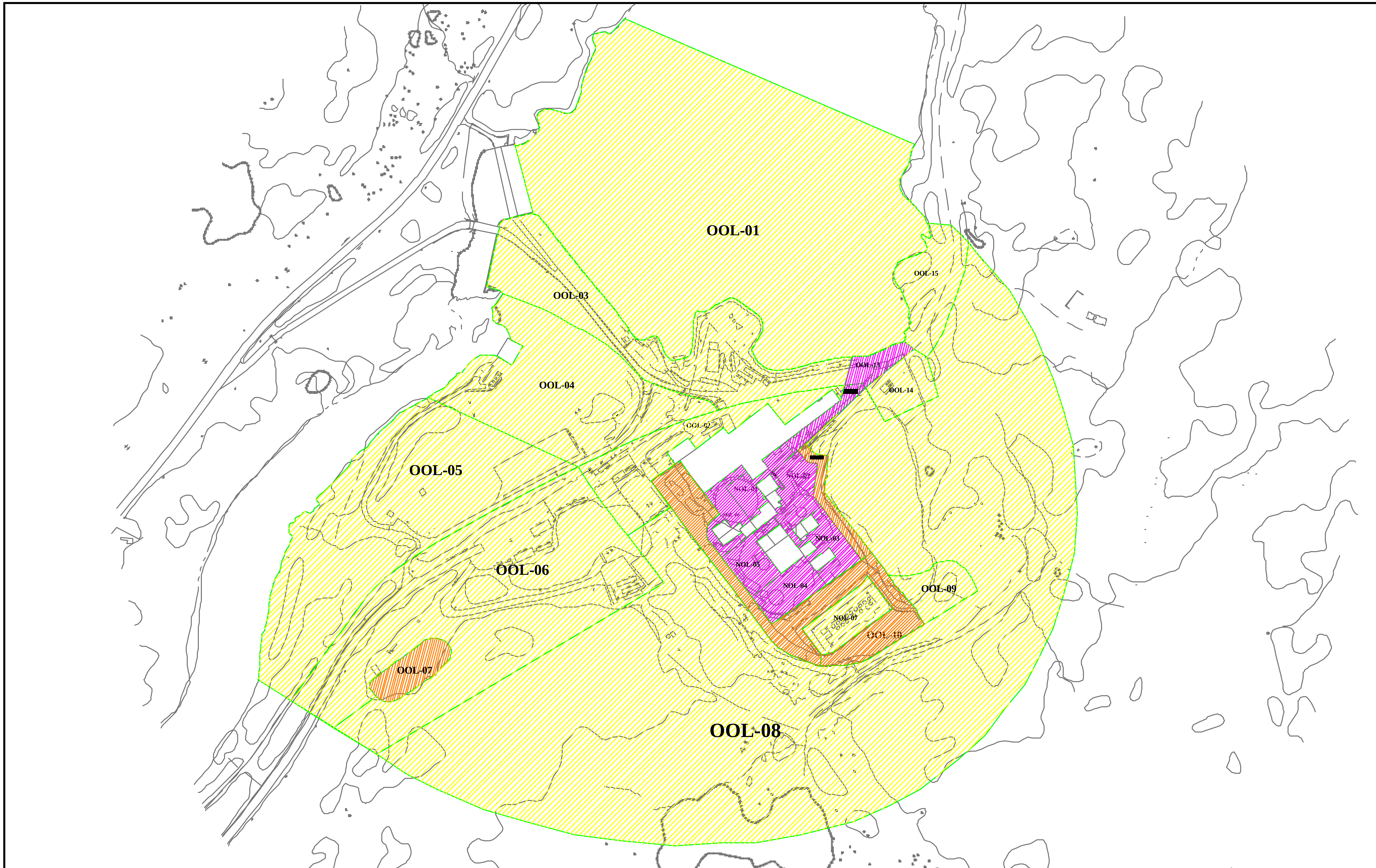
Yankee Atomic Power Company
Preliminary Structure Survey Area Boundaries

YANKEE

Date: November 2003

Revision: 5

Figure: 2-2



N

Legend

= Class 3

= Class 2

= Class 1

Notes

Classifications as of July 31, 2003

Scale: 1" = approx 280'

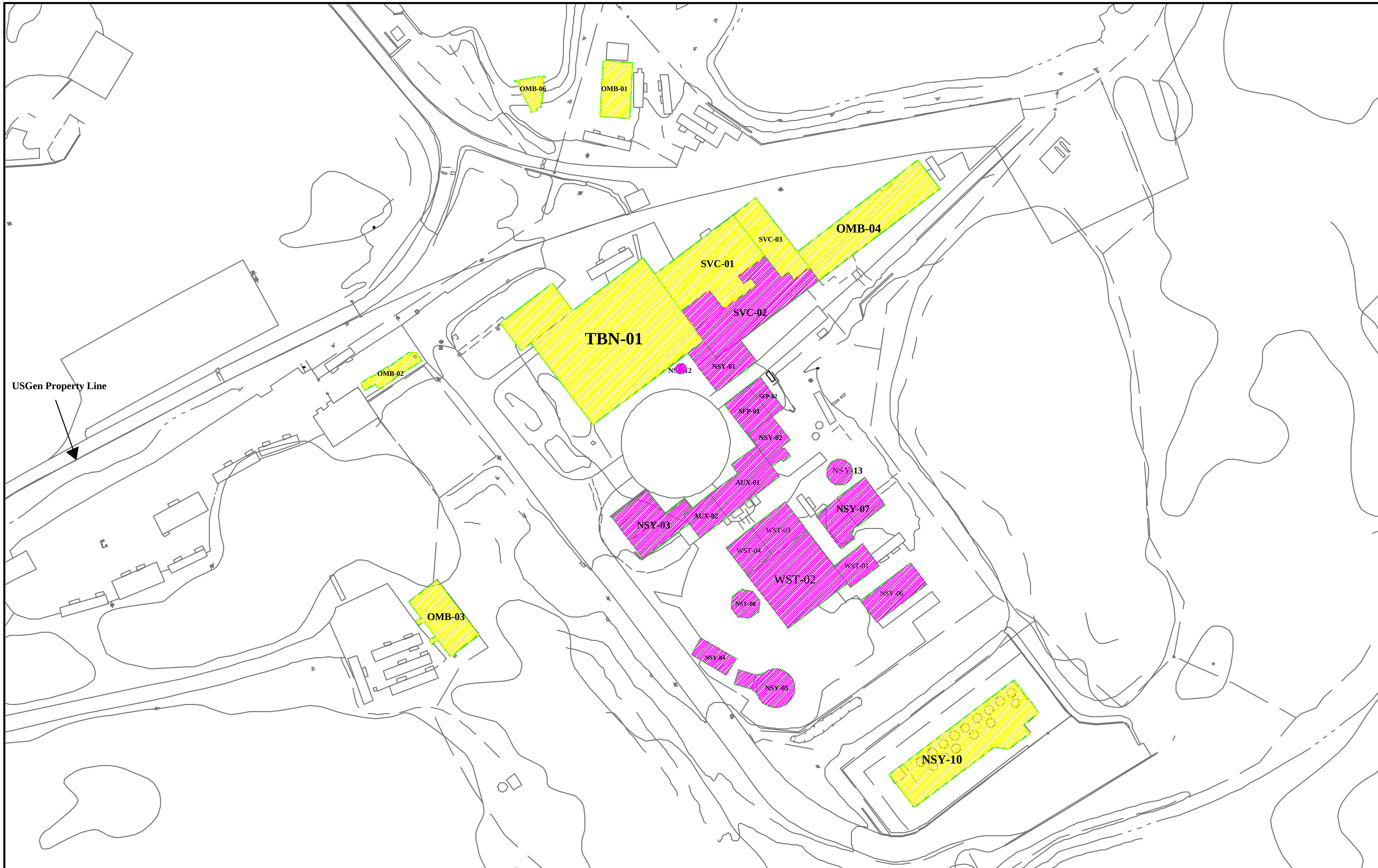
Yankee Atomic Power Company
Preliminary Land Surface Impacted Area Classifications

YANKEE

Date: October 2003

Revision: 4

Figure: 2-3



N

Legend

= Class 3

= Class 2

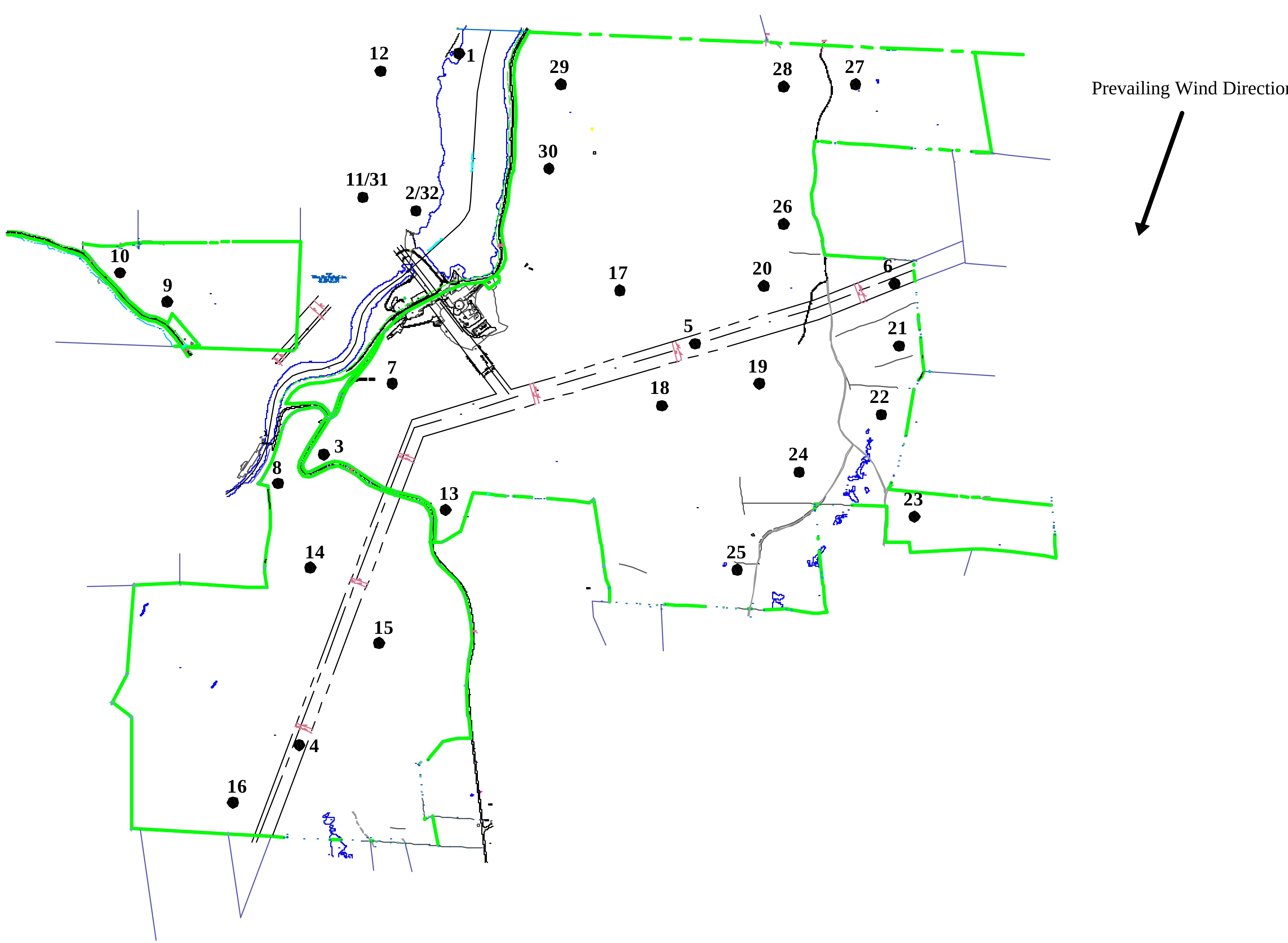
= Class 1

Notes

Classifications as of July 31, 2003

Scale: 1" = approx 125'

Yankee Atomic Power Company Preliminary Structure Classifications	YANKEE	Date: November 2003 Revision: 5	Figure: 2-4



N

Legend

= Site Boundary

= Water

= Soil Sample Location

Notes

Scale: 1" = approx 2000'

Yankee Atomic Power Company

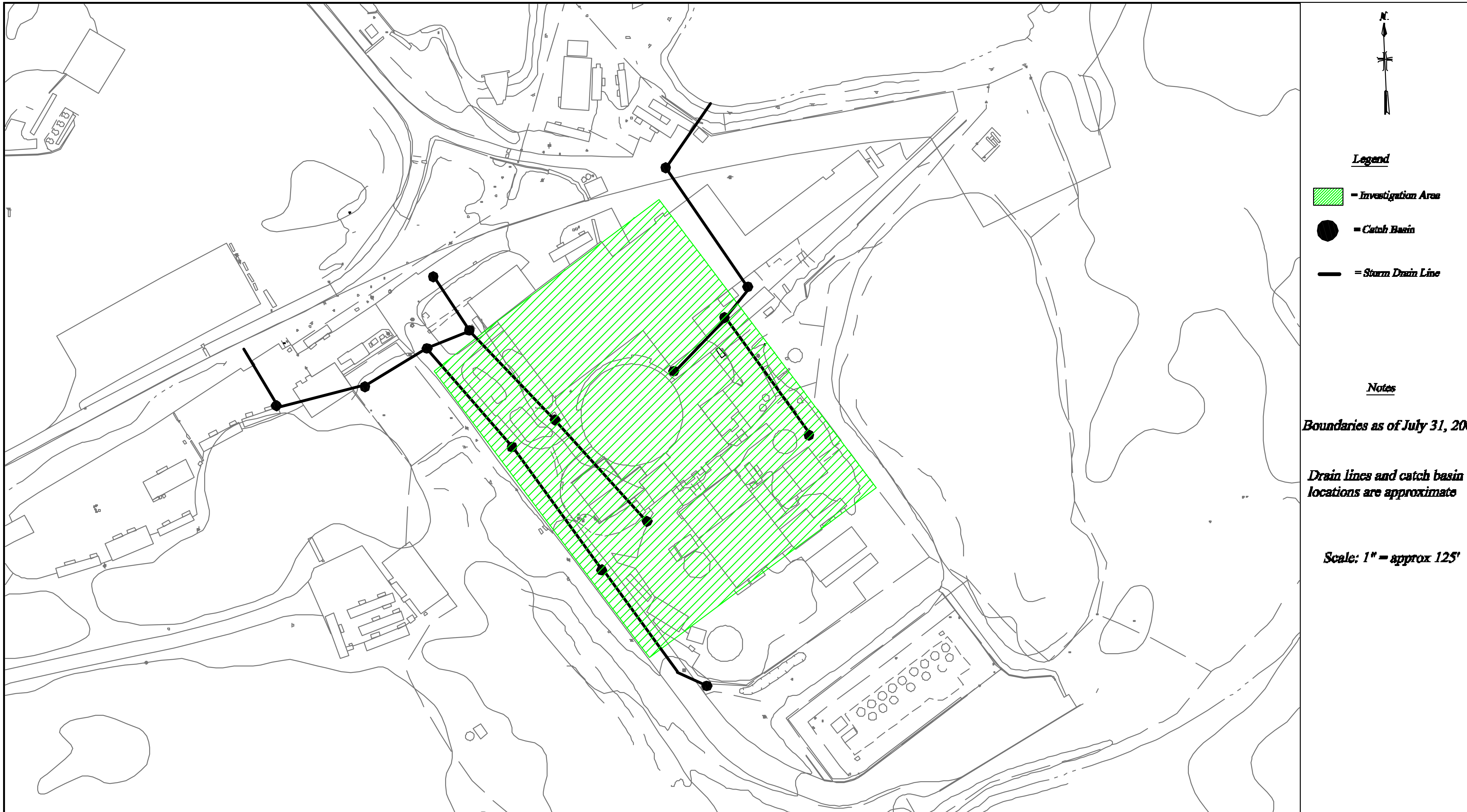
Locations of Samples to Determine Background Cs-137 in Soil

YANKEE

Date: November 2003

Revision: 1

Figure: 2-5



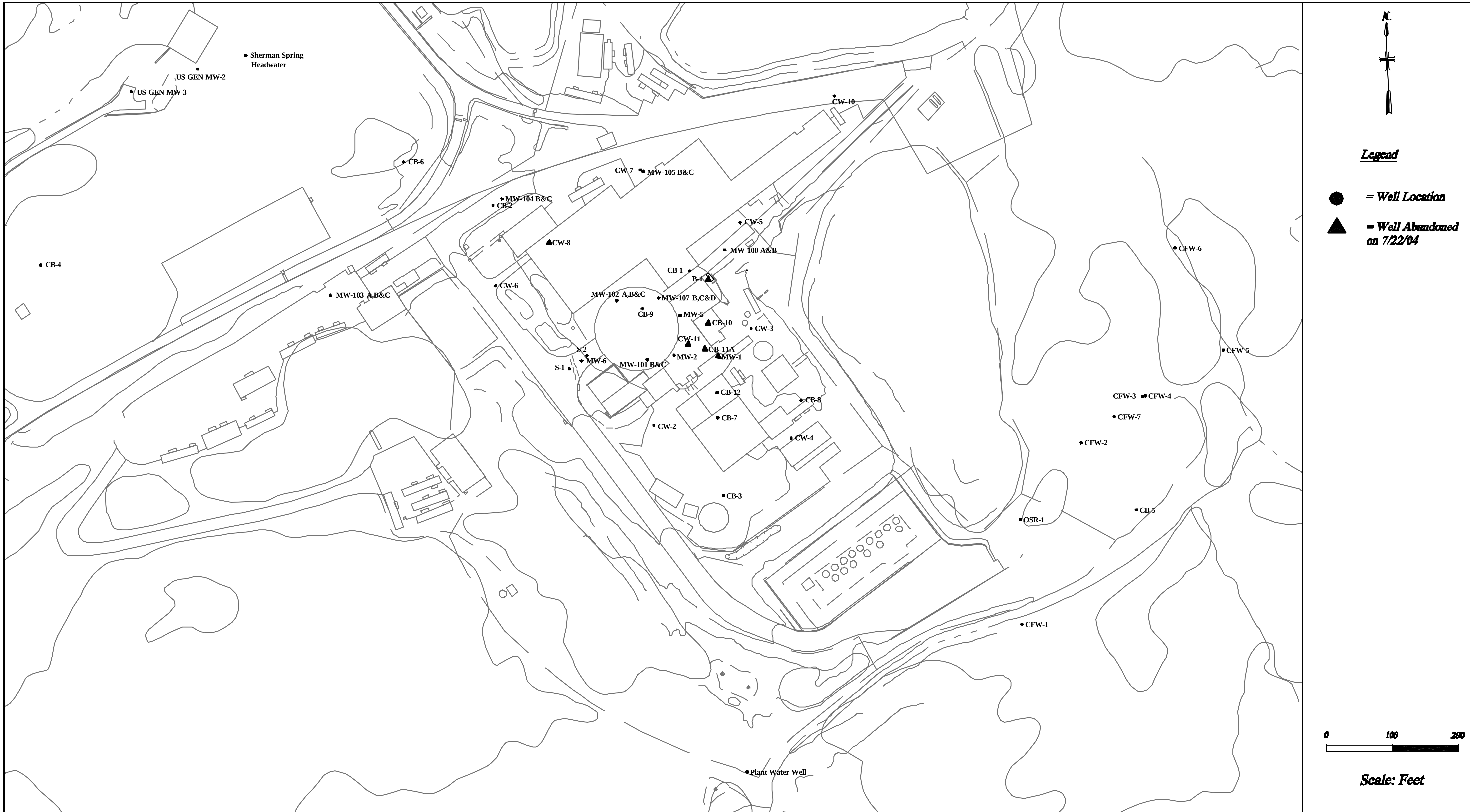
Yankee Atomic Power Company
Preliminary Subsurface Investigation Areas

YANKEE

Date: November 2003

Revision: 2

Figure: 2-6



Yankee Nuclear Power Station

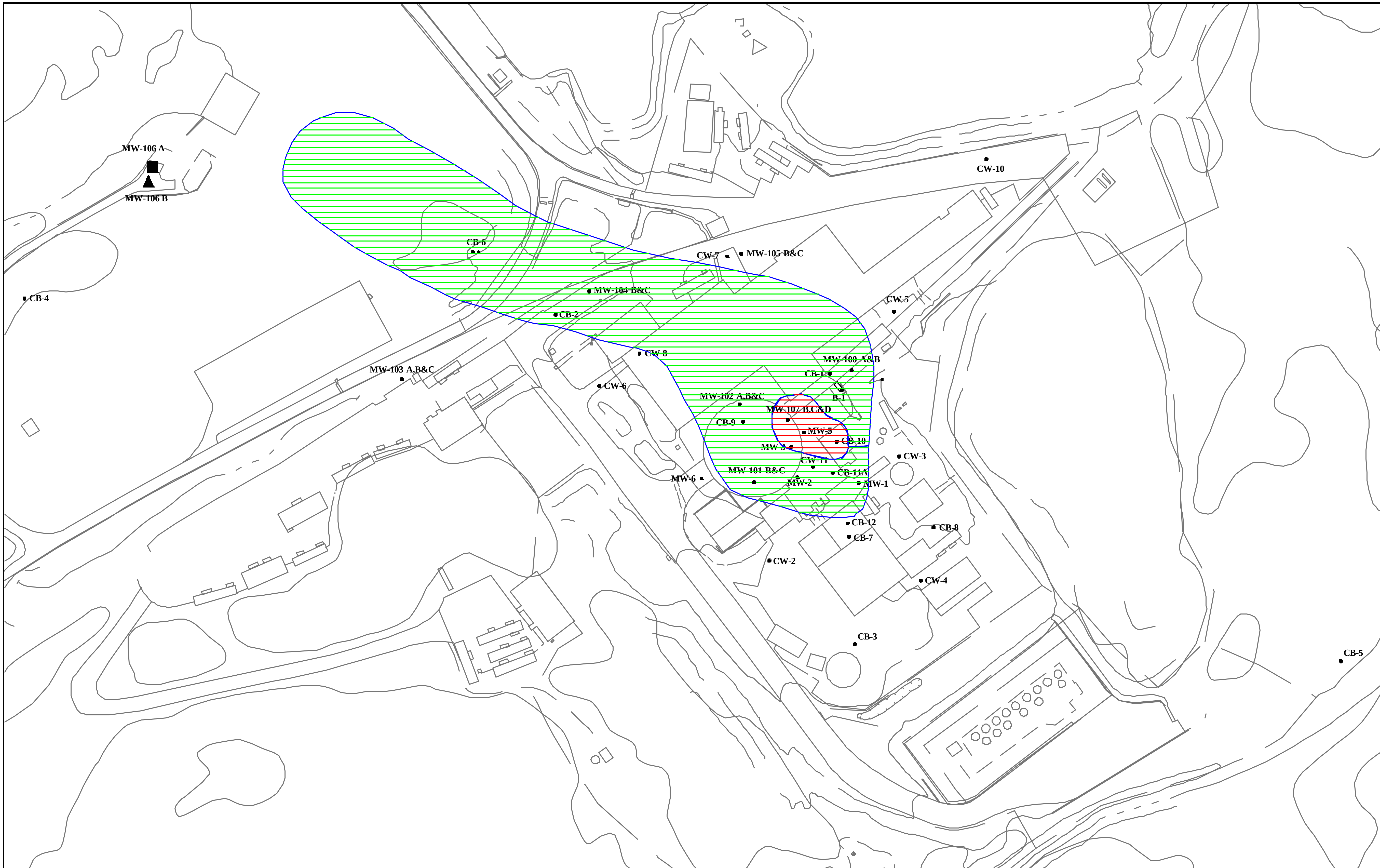
Monitoring Well Location Map

YANKEE

Date: August 2004

Revision: 1

Figure: 2-7



N

Legend

= Existing well location

= Proposed bedrock well location

= Proposed overburden well location

= 300 to 3000 pCi/l

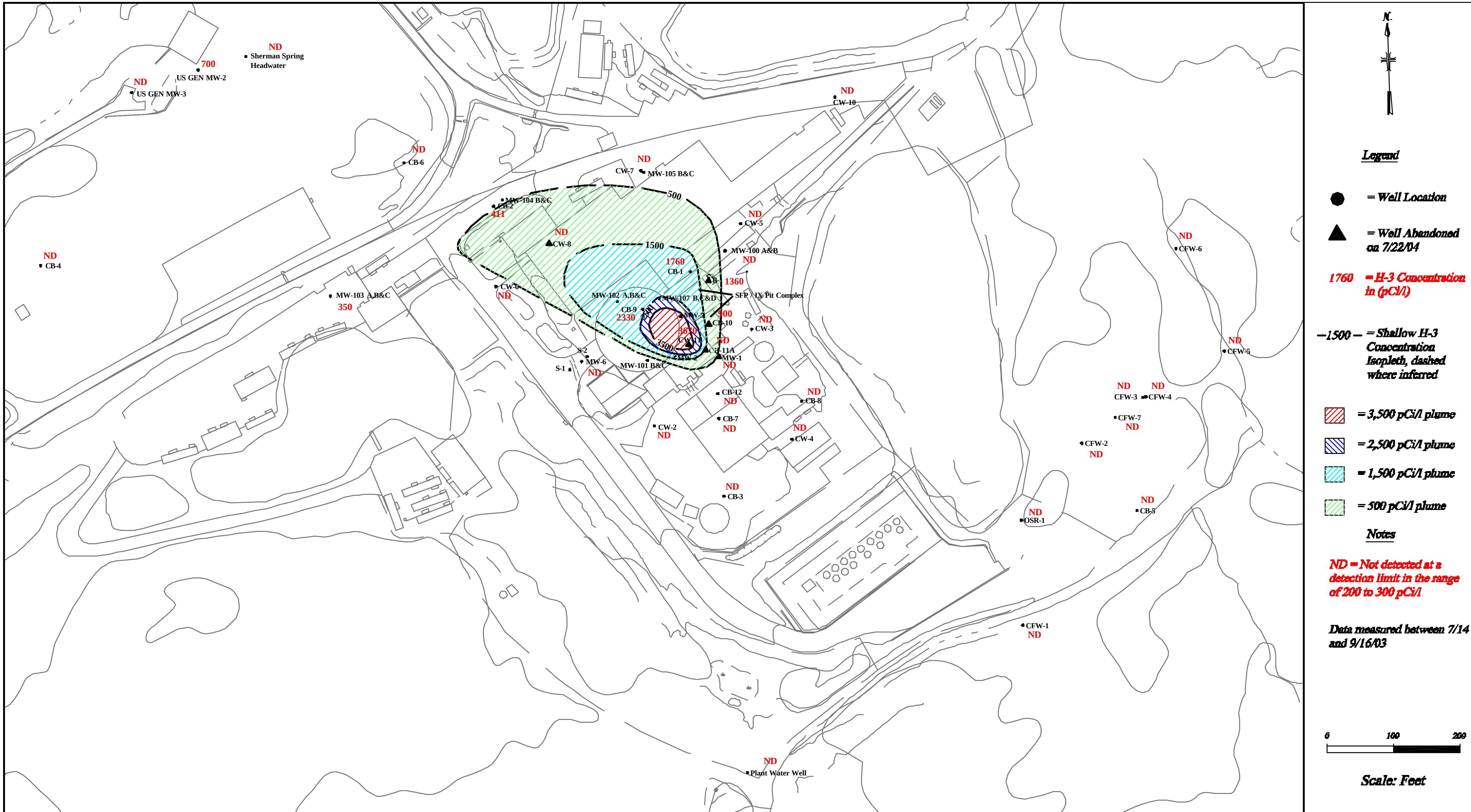
= 3000 to 6000 pCi/l

Notes

Data as of Fall 2001

Scale: 1" = approx 125'

Yankee Atomic Power Company *Concentration of H-3 in Groundwater*	YANKEE	*Date: November 2003* *Revision: 1*	*Figure: 2-8*



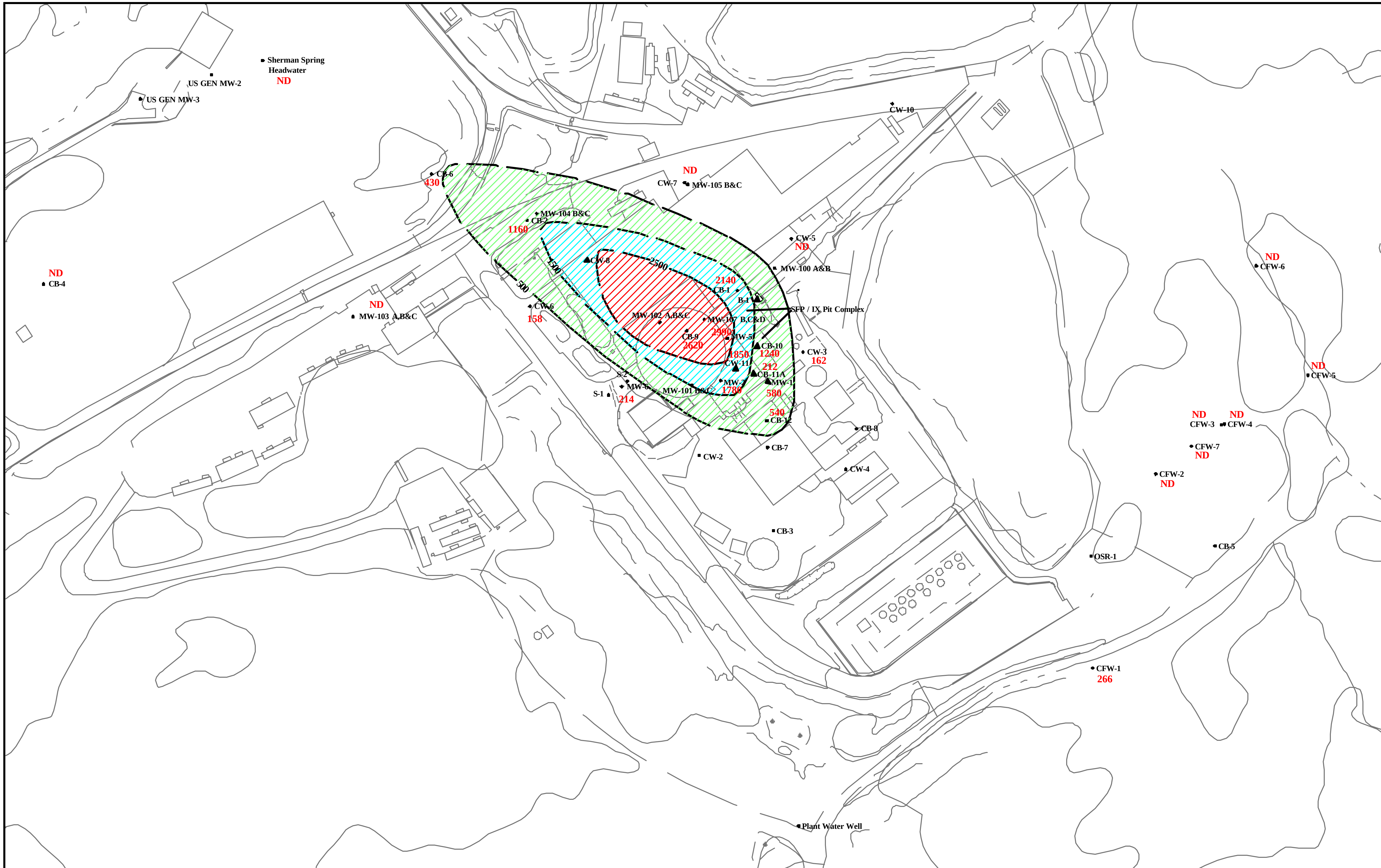
Yankee Nuclear Power Station
Shallow Tritium Plume Map
for July 2003

YANKEE

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

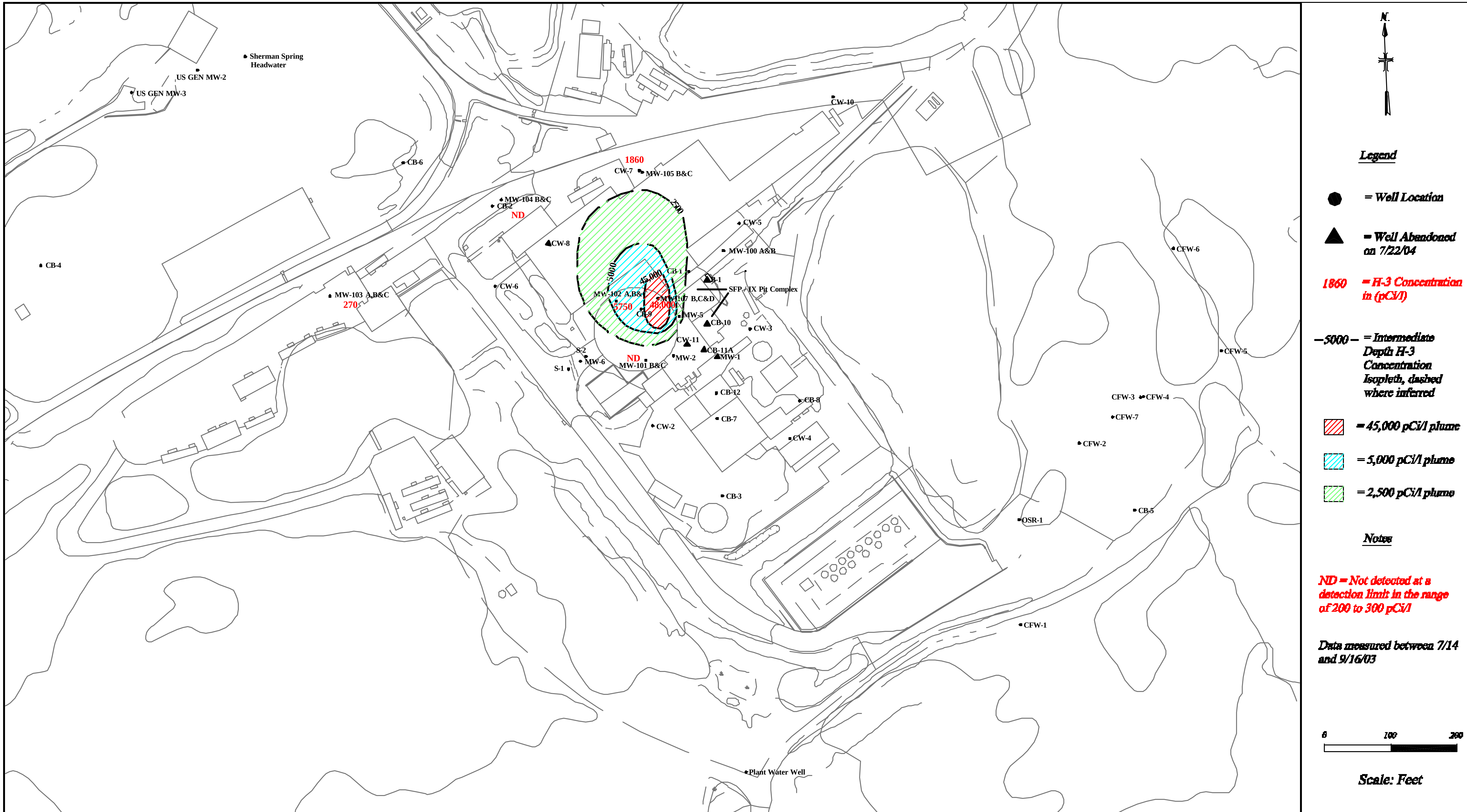
Yankee Nuclear Power Station
Shallow Tritium Plume Map
for November 2003

YANKEE

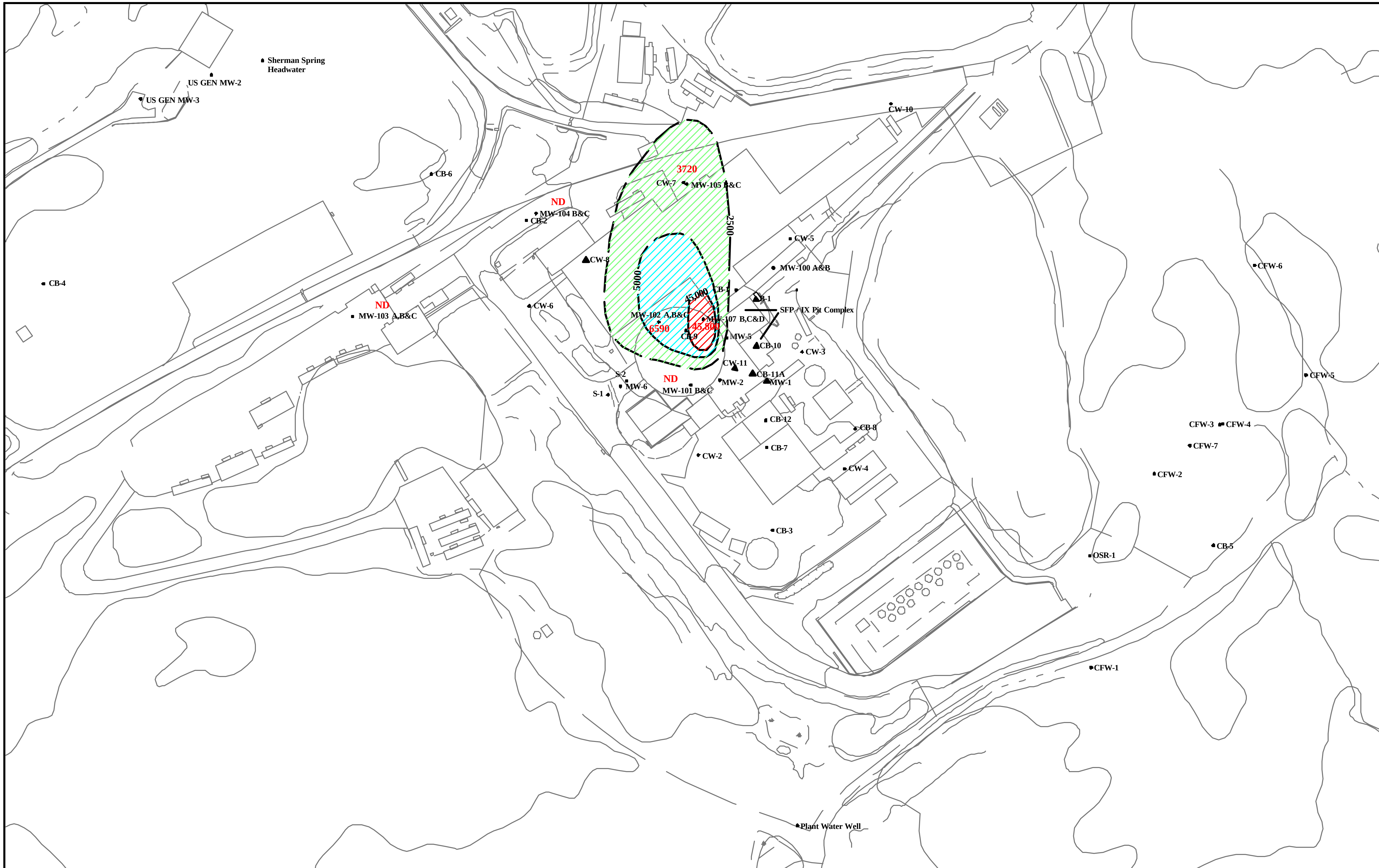
Date: August 2004

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*



N.

Legend

- = Well Location
- = Well Abandoned on 7/22/04
- 3720 = H-3 Concentration in (pCi/l)
- —5000— = Intermediate Depth H-3 Concentration Isopleth, dashed where inferred
- = 45,000 pCi/l plume
- = 5,000 pCi/l plume
- = 2,500 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
Intermediate Depth Tritium Plume Map
for November 2003

YANKEE

Date: August 2004

Revision: 1

Figure: 2-9d

Yankee Nuclear Power Station

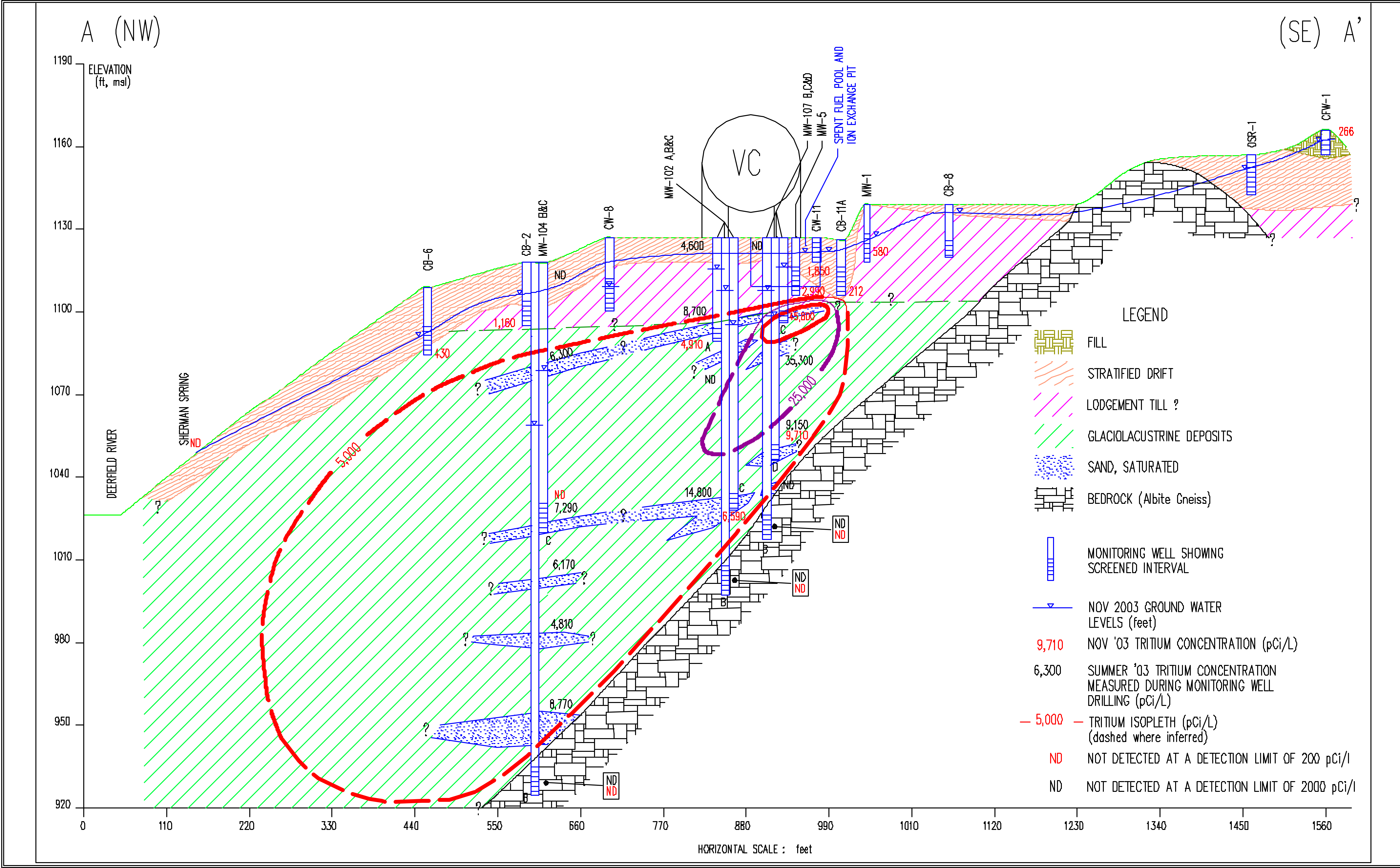
Location of Cross Section Lines

YANKEE

Date: August 2004

Revision: 1

Figure: 2-10a



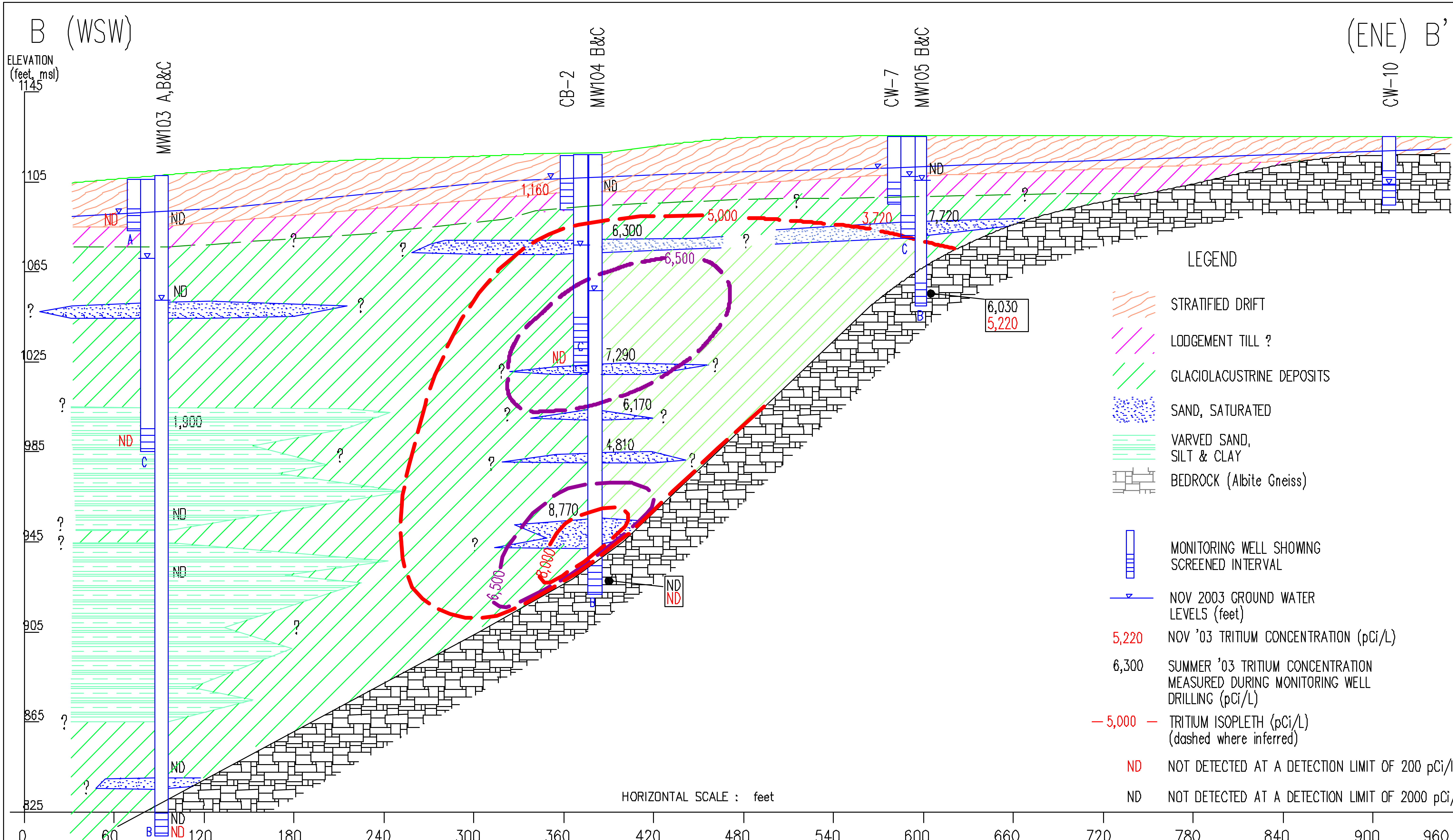
Yankee Nuclear Power Station

Hydrogeologic Cross Section A - A'

Date: August 2004

Revision: 1

Figure: 2-10b



Yankee Nuclear Power Station

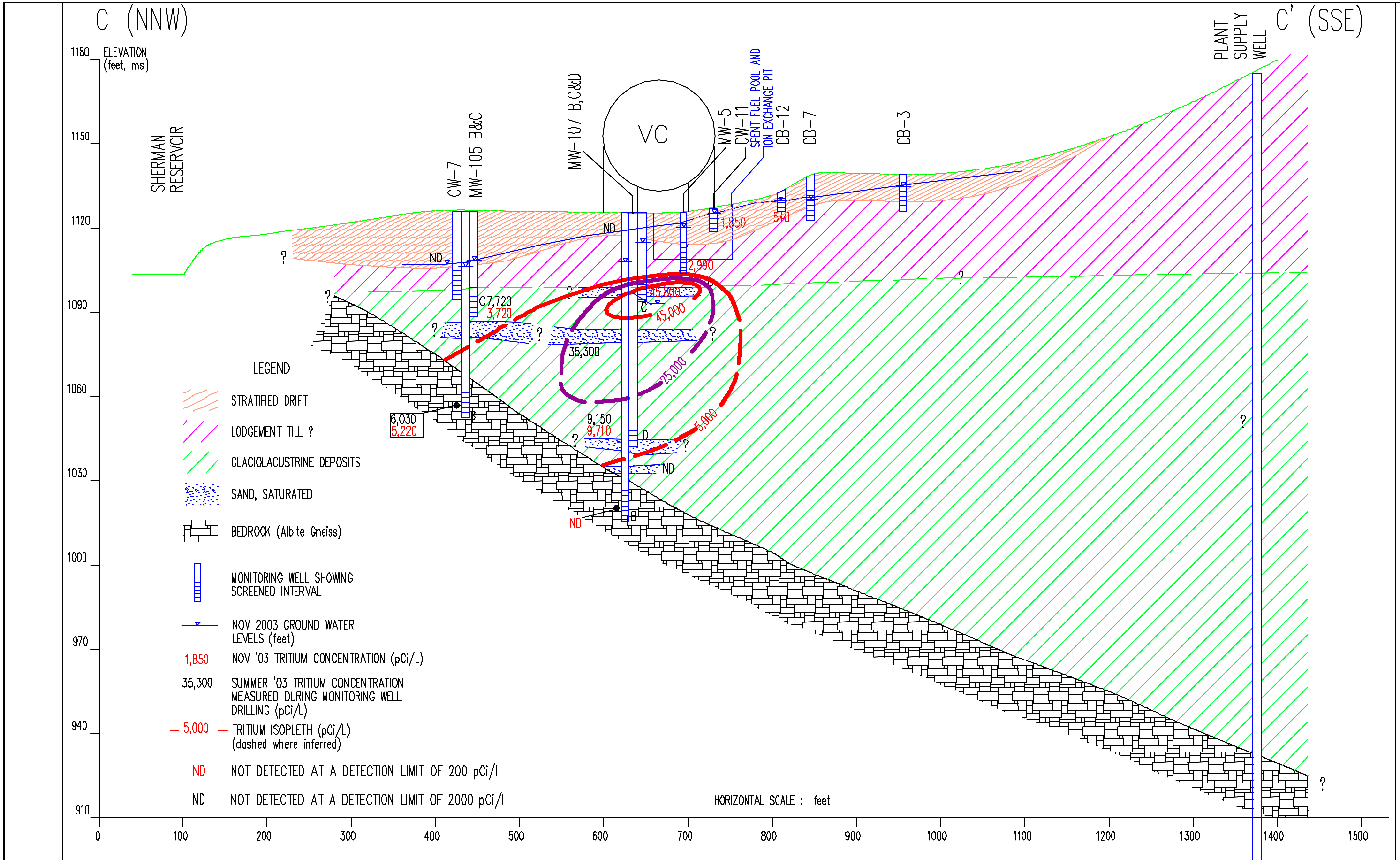
Hydrogeologic Cross Section B - B'

YANKEE

Date: August 2004

Revision: 1

Figure: 2-10c



Yankee Nuclear Power Station

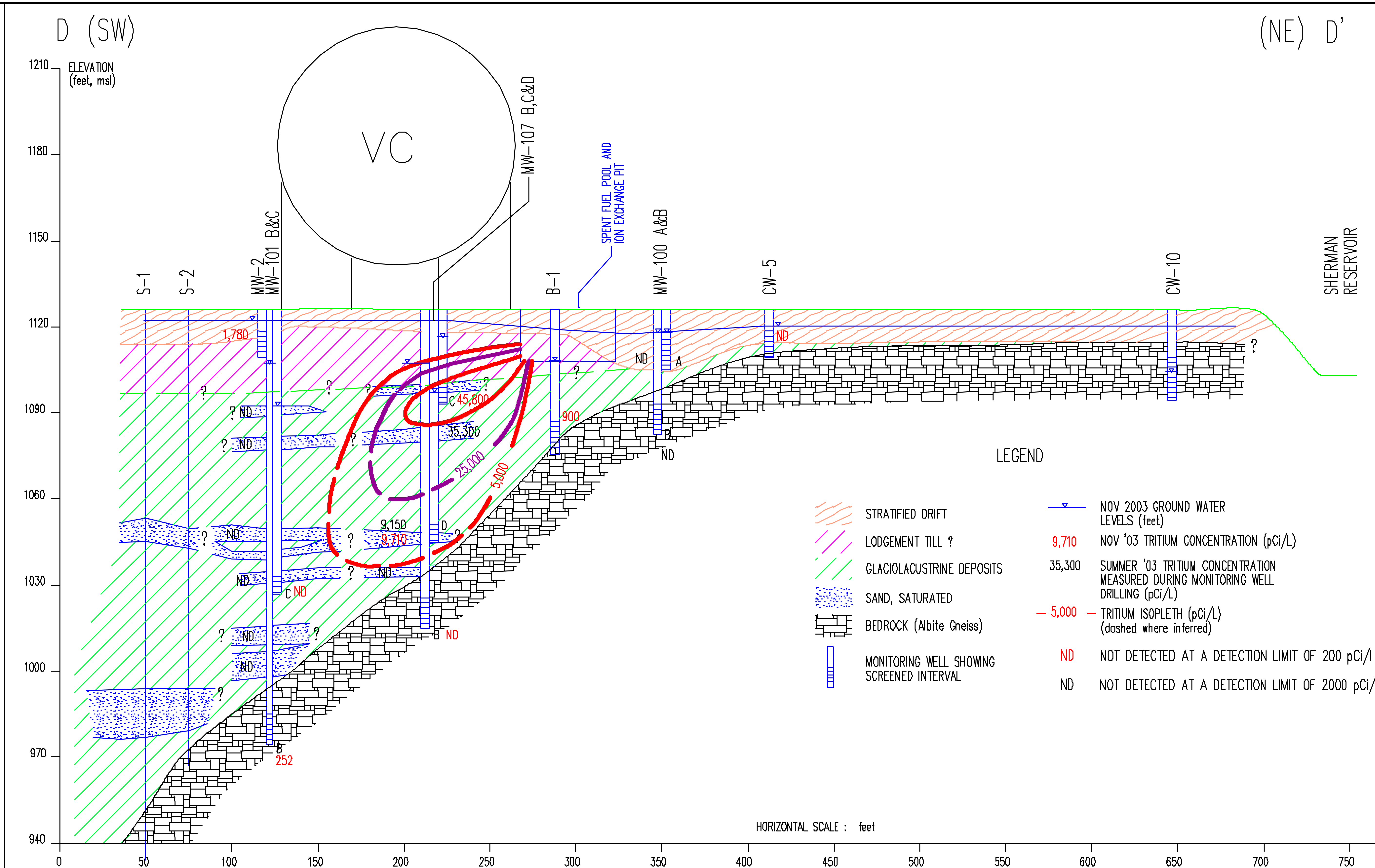
Hydrogeologic Cross Section C - C'

YANKEE

Date: August 2004

Revision: 1

Figure: 2-10d



Yankee Nuclear Power Station

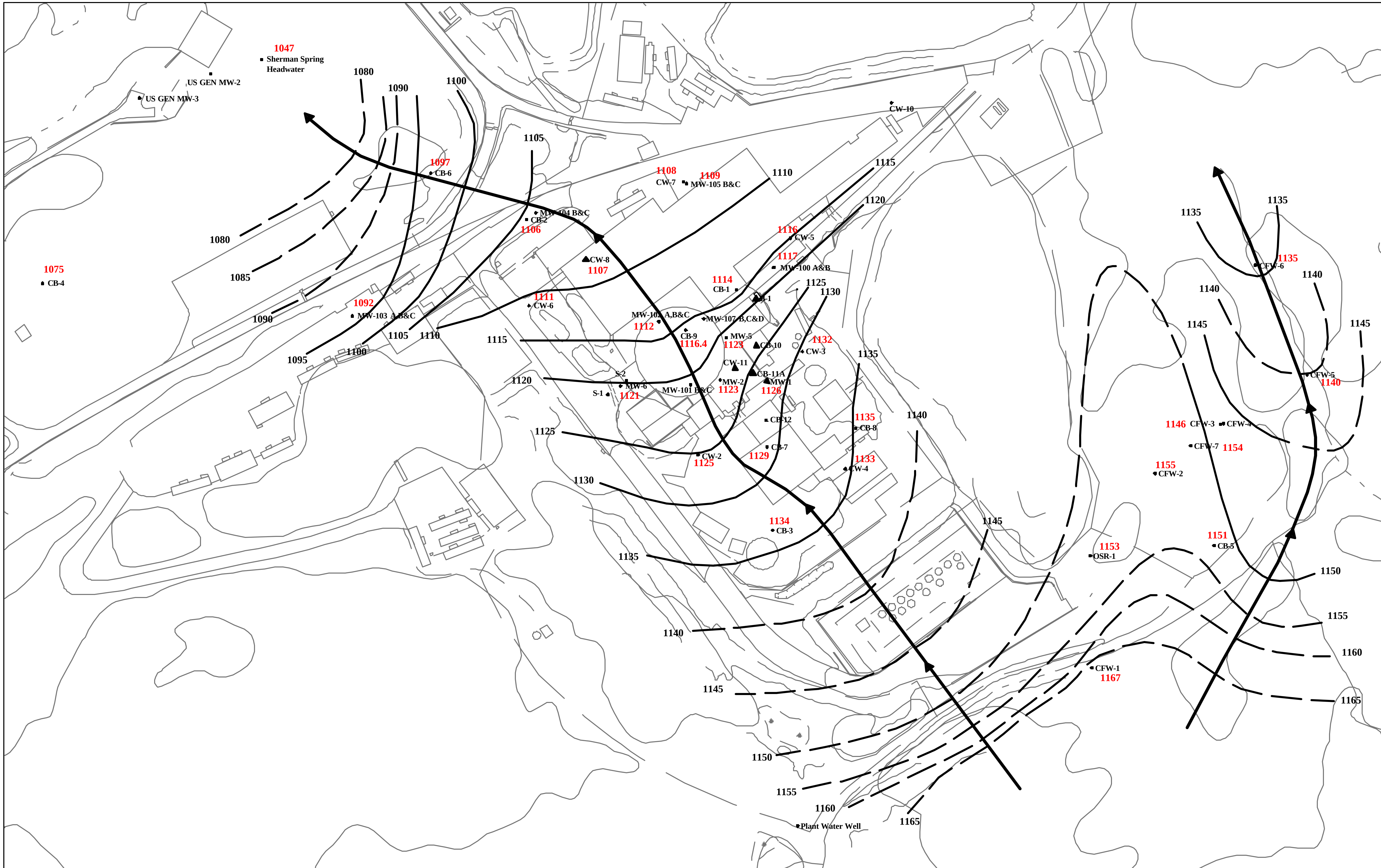
Hydrogeologic Cross Section D - D'

YANKEE

Date: August 2004

Revision: 1

Figure: 2-10e



Legend

= Well Location

= Well Abandoned on 7/22/04

1107 = Shallow groundwater elevations in feet

–1110– = Shallow groundwater elevation Isopleth, dashed where inferred

= Inferred shallow groundwater flow direction

Notes

Datum is Mean Sea Level (MSL)

Data measured between 7/14 and 9/16/03

0 100 200

Scale: Feet

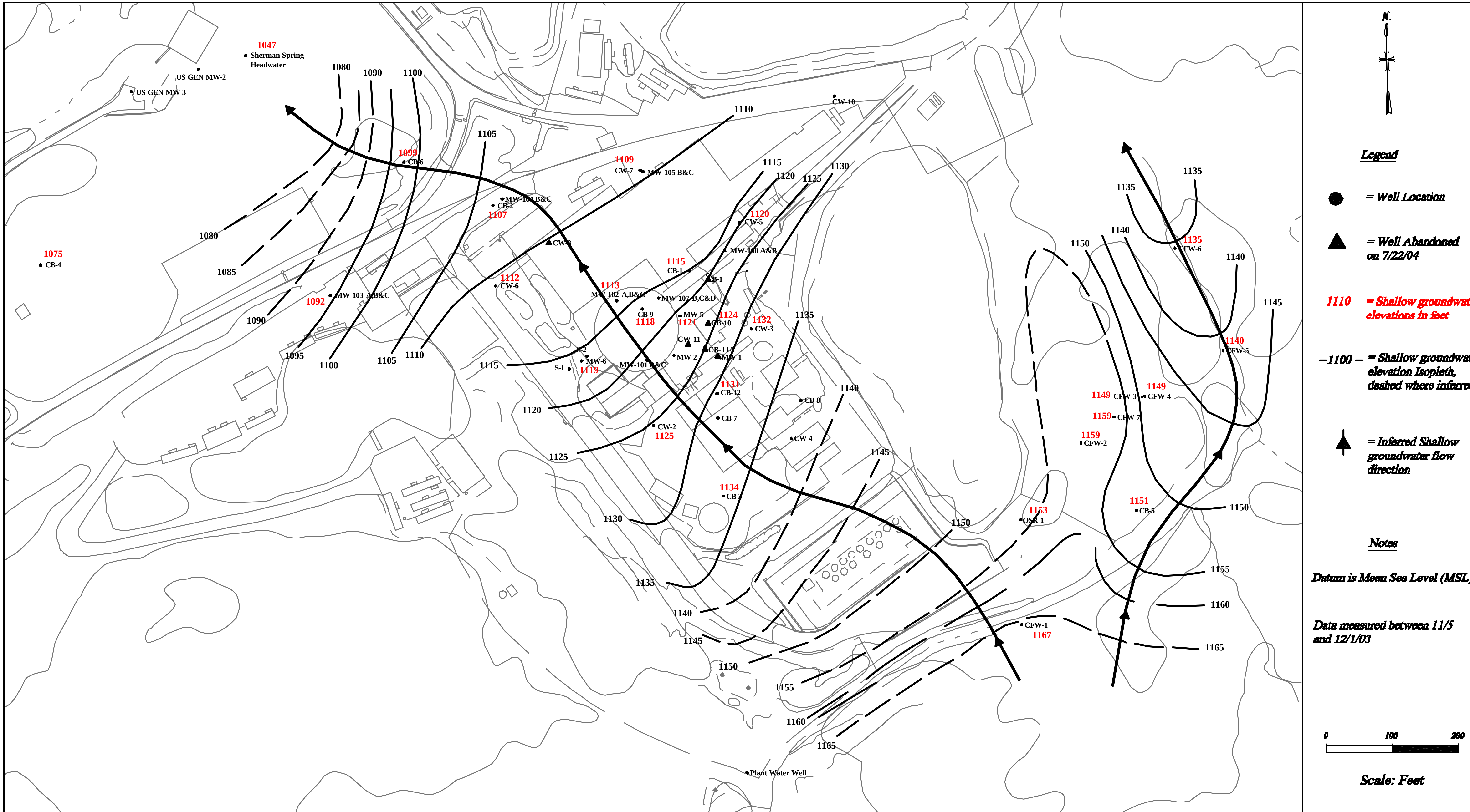
Yankee Nuclear Power Station
Shallow Groundwater Elevation Contour Map
for July 2003

YANKEE

Date: August 2004

Revision: 1

Figure: 2-11



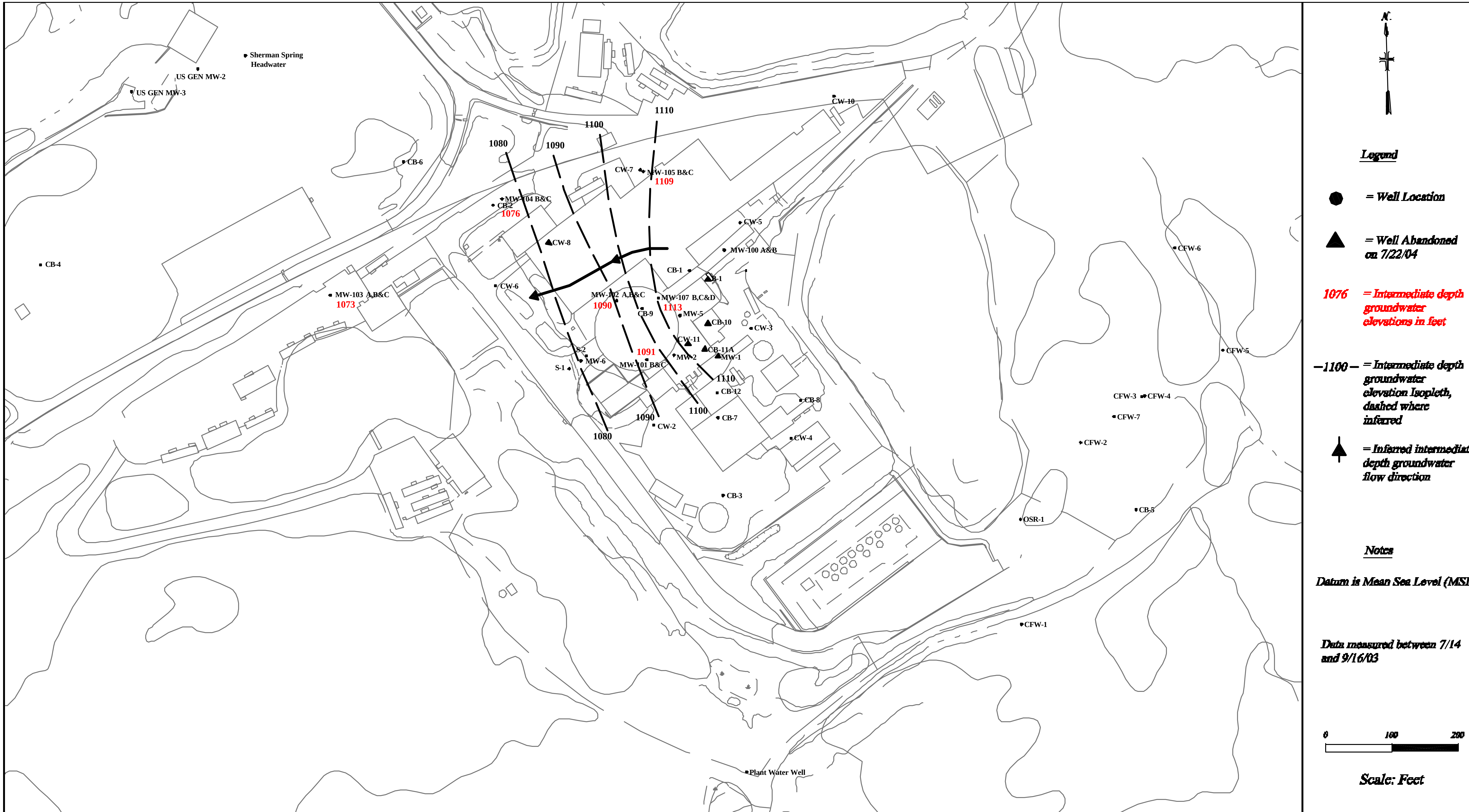
Yankee Nuclear Power Station
Shallow Groundwater Elevation Contour Map
for November 2003

YANKEE

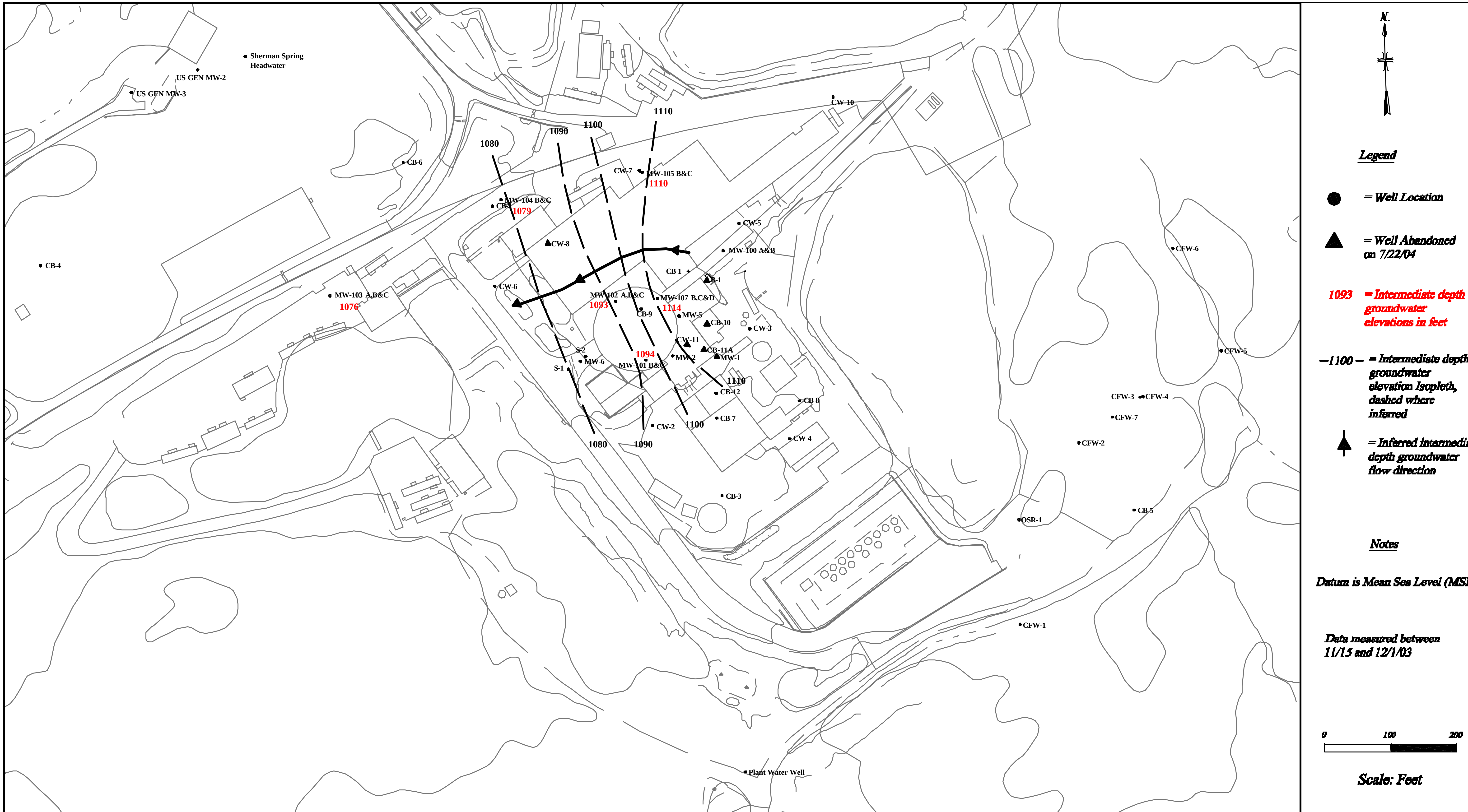
Date: August 2004

Revision: 1

Figure: 2-12



Yankee Nuclear Power Station Intermediate Depth Groundwater Elevation Contour Map for July 2003	Date: August 2004 Revision: 1	Figure: 2-13

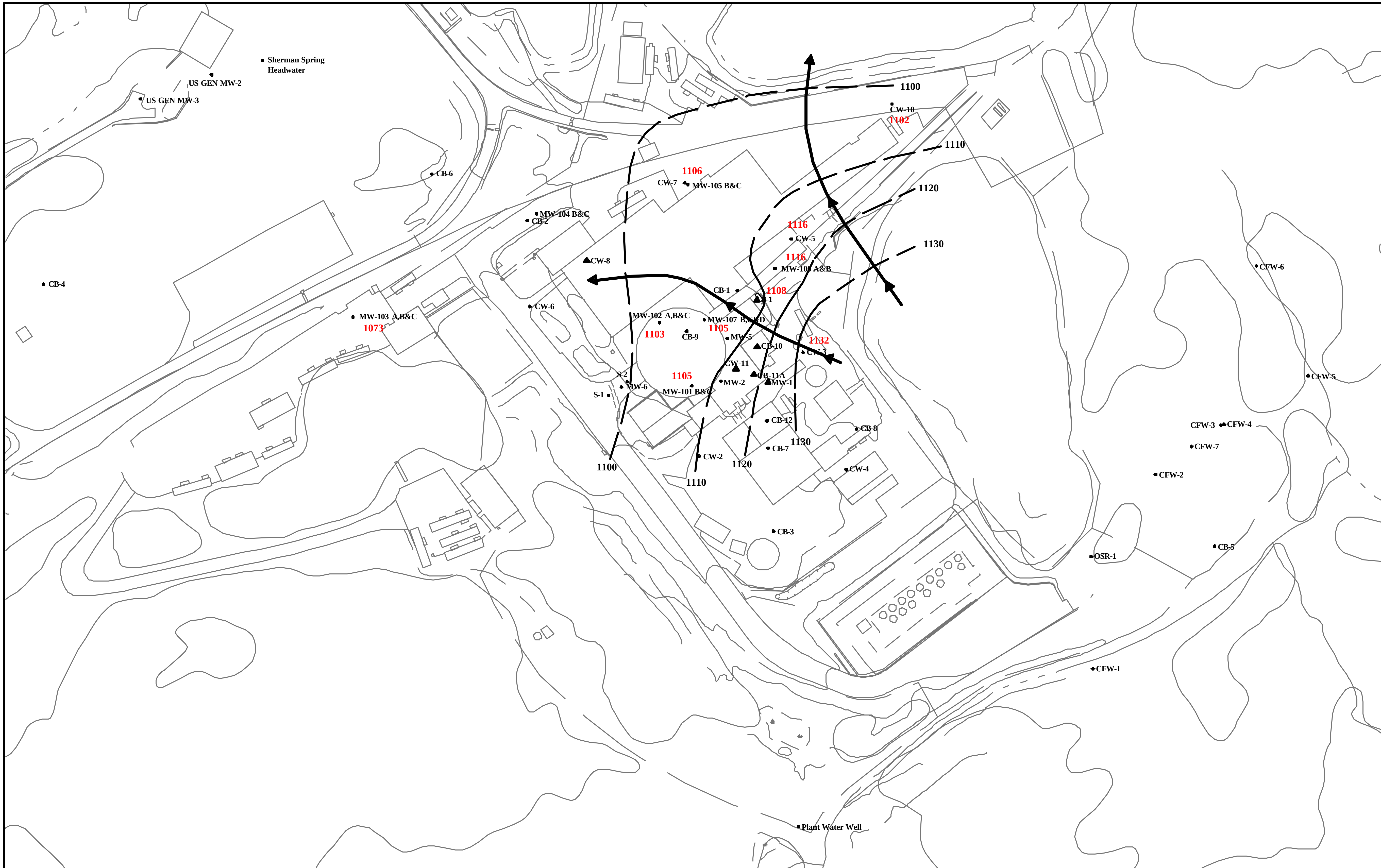


Yankee Nuclear Power Station
Intermediate Depth Groundwater Elevation Contour Map
for November 2003

Date: August 2004

Revision: 1

Figure: 2-14



Legend

= Well Location

= Well Abandoned on 7/22/04

1105 = Bedrock groundwater elevations in feet

—1100— = Bedrock groundwater elevation Isopleth, dashed where inferred

= Inferred bedrock groundwater flow direction

Notes

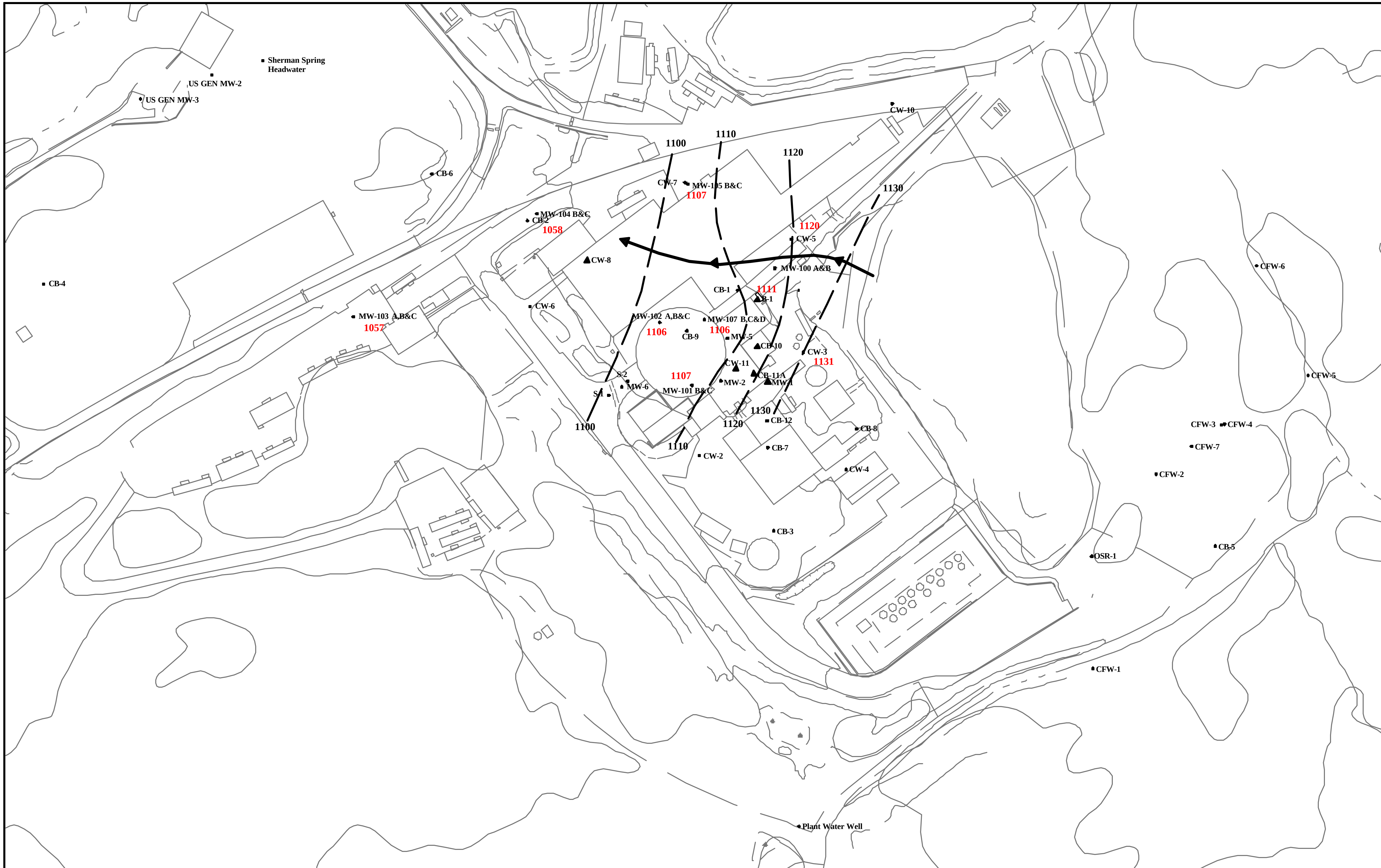
Datum is Mean Sea Level (MSL)

Data measured between 7/14 and 9/16/03

0 100 200

Scale: Feet

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



N

Legend

= Well Location

= Well Abandoned on 7/22/04

1106 = Bedrock groundwater elevations in feet

–1100– = Bedrock groundwater elevation Isopleth, dashed where inferred

= Inferred bedrock groundwater flow direction

Notes

Datum is Mean Sea Level (MSL)

Data measured between 11/15 and 12/1/03

0 100 200

Scale: Feet

Yankee Nuclear Power Station
Bedrock Groundwater Elevation Contour Map for November 2003

YANKEE

Date: August 2004

Revision: 1

Figure: 2-16

Appendix 2A

Summaries of the Significant Events Leading to Long-Term Contamination of the YNPS Site (Presented in LTP Table 2-3)

AOR 61-15: Radioactive Spill – 9/20/61
A half-liter container of reactor coolant water was dropped on the asphalt in the Potentially Contaminated Area between the Primary Auxiliary Building and the Waste Disposal Building. The sample contained approximately 35 μCi (specific radionuclide data not available). The spill was absorbed using absorbent paper and the area decontaminated by mopping. The fixed contamination remaining was approximately 0.05 mr/hr at 1 inch from the pavement.
Impacted Areas **NOL-02/ NOL-05**

AOR 63-12: Shield Tank Cavity Fill Water Spill – 9/18/63
A one-half inch sampling valve located over the IX Pit was inadvertently left open while filling the shield tank cavity. This resulted in a spill of approximately 10 gallons of water from the Safety Injection Tank. A portion of the spill ran off the deck of the pit and onto a section of the blacktop surface to the west of the pit. The radiation level in the immediate area was 70-100 mr/hr measured at one inch. Contamination levels were 10^6 to 10^7 dpm (specific radionuclide data not available) over areas of several square inches. Run off water resulted in contamination levels of 20-60,000 dpm/ft^2 (Sic).
Impacted Areas **NOL-01/NOL-02**
Impacted Structures **NSY-02**

AOR 63-17: De-watering Pump Packing Leakage – 10/8/63
A water leak from the fuel chute de-watering pump was routed, via a small utility hose, to a 30 gallon collection drum placed in a storm drain catch basin (ECB-005) located between the railroad tracks and the NE corner of the spent fuel pit. It was determined that the bottom rim of the barrel was corroded, and water was leaking from the bottom of the barrel. At the time the leak was identified, six to eight inches of water had accumulated in the barrel with activity of 6 x 10^{-5} μCi/ml (specific radionuclide data not available). It was believed only a small amount of water was leaked to the storm system.
Impacted Areas **OOL-05/OOL-06/NOL-01**
Impacted Sub-surface Areas/Structures **- East Storm Drain System**

AOR 64-08: Seal Water Tank Spill – 9/3/64
Shutdown cooling pump seals leaked reactor coolant water and back-flowed into the seal water tank. This caused the tank to overflow through the vent connection, into the common relief valve discharge line and onto the Primary Auxiliary Building roof. An estimated 35 gallons of water containing a total activity of 270 μCi (specific radionuclide data not available) was released. The Roof Drain System drained into the Storm Drain System via a sub-surface piping connection. A sample of the storm drain (WCB-009) was determined to contain 1 x 10^{-6} μCi/ml. The predominant isotopes were Co-58, Co-60 and Mn-54 (distribution of the radionuclides in the sample not available). Service Water was diverted to the storm drain to flush the system.
Impacted Areas **- AUX-02 Roof and Roof Drain System**
Impacted Sub-surface Areas/Structures **- West Storm Drain System**

AOR 64-13: Leakage from Ion Exchange Pit - 10/3/64
After filling the Ion Exchange Pit to its normal operating level, the operator failed to close the fill valve. Water continued to flow into the pit from the Primary Water Storage Tank by gravity feed. Later, the operator noticed water seeping through the blacktop on the west side of the pit,

diagnosed the cause and closed the valve. The water on the blacktop was sampled and was found to contain radioactivity. The radionuclides and concentrations identified were: Ag-110m at 5 x 10^{-7} µCi/ml and Co-60 at 1 x 10^{-6} µCi/ml. The blacktop was rinsed down with Service Water to the storm drain (ECB-005).
***Impacted Areas* NSY-02/NOL-01/OOL-05/OOL-06**
***Impacted Sub-surface Areas/Structures* - East Storm Drain System internal and external to piping (backfill) / SFP-02 sub-floor / NSY-09 / AUX-01 North external perimeter (backfill) / SFP-01 West external perimeter (backfill) / BRT-01 Eastern external perimeter**

AOR 66-7: Spent Fuel Pit Water Spill – 9/27/66
A two-inch priming valve for the Spent Fuel Pit (SFP) cooling and purification pump was left open; however an upstream valve isolating make up water to the Low Pressure Surge Tank (LPST) was correctly closed. The LPST make up pump was started to provide make up water to a hose connection located between the two valves to wash down a shipping cask as it was removed from the pit. Water flowed through the open priming valve to the SFP in sufficient quantity to result in actuation of the high level alarm. The reason for the high level alarm was not immediately determined and by the time the reason was identified water had overflowed from the SFP. Approximately 33 gallons of water flowed down the SFP exterior wall, over a small section of asphalt paving and into an immediately adjacent storm drain, ECB-005. A continuous service water flush of the east side culvert system (ECB-005) was initiated and continued for a 24 hour period. This occurrence resulted in a total release of 4 µCi gross β-γ and 670 µCi of tritium (more specific radionuclide data not available).
***Impacted Areas* SFP-01 North external wall /NOL-01/OOL-01**
***Impacted Sub-surface Areas/Structures* East Storm Drain System internal and external to piping (backfill between SFP-01 and ECB-005)**

AOR 66-8: Abnormal Activity in Storm Drain – 9/27/66
Water from the west storm drain culvert was sampled (the SFP water released discussed above discharged to the east side only). An average of two samples from the west side showed gross activity of 6.7 x 10^{-7} µCi/ml (specific radionuclide data not available). Investigation found a relief valve on the safety injection tank heating system to be slowly leaking into a floor drain in the PAB. The floor drains in that section of the building were traced to discharge to a storm drain located on the outside of the building (WCB-009). Further investigation indicated that the relief valve leak could not have existed for more than one day and that the maximum volume did not exceed eight gallons during that period. A sample of culvert water collected 24 hours after the occurrence indicated a gross activity of 1.2 x 10^{-8} µCi/ml and tritium activity of 5.1 x 10^{-5} µCi/ml. This occurrence resulted in a total release of 0.8 µCi gross β-γ and 3.32 mCi tritium.
***Impacted Area* - OOL-05/OOL-06**
***Impacted Sub-surface Areas/Structures* - West Storm Drain system**

AOR 66-9: Hose Failure – 11/1/66
The hose used for a routine draining of the fuel chute pump discharge line burst. Less than 10 gallons of contaminated water flowed into a storm drain served by the east culvert (ECB-005). Approximately 10 gallons of water with an activity of 3.0 x 10^{-3} µCi/ml (for a total of 113 µCi) was released. The spill area was flushed with service water. The east culvert was sampled after the spill.

***Impacted Areas* - NOL-01/OOL-01**
***Impacted Sub-surface Areas/Structures* - East Storm Drain system**

AOR 68-1: **Waste Holdup Tank Moat Spill – 1/16/68**
The suction line from the waste hold-up tank was found to be frozen. Approximately 200 gallons of water spilled from a valve bonnet failure caused by the freezing of the suction line. A total of 520 μCi β-γ and 698 mCi tritium were spilled into the moat. The spill was contained within the moat structure.
***Impacted Structures* - NSY-07**

PIR 75-7: Yard Area Contamination 7/16/75
An area of land near the Ion Exchange Pit was identified with a contamination level of approximately 500,000 dpm. Over the next few days, the entire restricted area was surveyed. Fourteen areas, ten of which were in areas previously identified as a "clean area," were found to be contaminated at levels greater than 1000 dpm/100 cm^2. Most of the contamination was removed, and the remaining contamination was sealed in place using asphalt sealer and covered with clean soil.
***Impacted Areas* - NOL-01 through NOL-06 and SVC-03**
***Impacted Sub-surface Areas/Structures* - SVC-03 beneath slab in old RCA access alley**

PIR 77-16: **Service Building Radioactive Sump Transfer Line Puncture – 12/21/77**
A boring bit inadvertently punctured the 2.5 inch stainless steel line leading from the Service Building Sump Tanks to the PAB while conducting core borings inside the Radiation Control Area. The sump line ran at a depth of 15 feet underground, where the damage occurred, and the boring depth was 61.5 feet. The damage was not detected until the next day when the sump pump started and water issued from the borehole. The sump pump ran through two cycles resulting in 20 gallons of water discharged from the rupture. The water contained the following:

Radionuclide	Total Activity, μCi	Concentration, μCi/ml	Fraction of MPC
I-131	16.50	2.18×10^{-4}	3.63
I-133	2.76	3.65×10^{-5}	0.18
Cs-134	0.34	4.46×10^{-6}	0.01
Cs-137	0.50	6.67×10^{-6}	0.02
Co-60	0.58	7.69×10^{-6}	0.01

No measurable levels of activity were released offsite or to the storm drain. The line was repaired, and a sand and concrete casing was poured around it.
***Impacted Areas* - NOL-02**
***Impacted Sub-surface Areas/Structures* - Soils surrounding perforation and transfer line backfill/Soils to a depth of 61.5 feet and below along the bore hole.**

PIR 80-9: Resin Spill - 8/6/80
A hose developed a pinhole leak, while pumping resin to a cask. The failure of the hose allowed the release of several gallons of water and one quart of resin. A 15 foot by 20 foot area of the RCA yard was contaminated. Radiation readings on contact with the resin were 1 mrad/hr and the spilled liquid reading were up to several hundred thousand dpm/100 cm^2 (sic) (specific

radionuclide data not available). Decontamination included removal and disposal of some of the blacktop.
Impacted Areas - **NOL-02/NSY-02**
Impacted Sub-surface Areas/Structures - **South and East exterior walls of NSY-02. The sub-slab area of NSY-02 (IX-pit) was also impacted due to transfer of contamination by surface water (i.e., water used in decontamination and rainwater) into cracks between asphalt and IX Pit walls.**

PIR 81-9: Contamination of Yard Area During Reactor Head Removal – 5/15/81
While positioning the reactor vessel head over the equipment hatch in preparation to lower the head through the equipment hatch, the reactor head made contact with the shield wall. This resulted in the spread of removable radioactivity outside of the Vapor Container (VC). Removable radioactivity immediately below the equipment hatch was 200 mrad/hr beta. The total activity released to the ground was approximately 250 μCi, with approximately 10μCi (specific radionuclide data not available) discharged to Sherman Pond. The area was cleaned, but due to rainfall trace radioactive material levels were detected in the east storm drains.
Impacted Areas - **NOL-01/NOL-06/OOL-12/OOL-13**
Impacted Sub-surface Areas/Structures - **BRT-01/in cracks and crevices under VC Equipment Hatch and along rails/ties in OOL-12 and OOL-13 and the East Storm Drain System due to surface water run-off.**

PIR 84-16: Drain Pipe Failure – 9/10/84
An excavated drainpipe from the Potentially Contaminated Area (PCA) storage building to the Waste Disposal building was found to be leaking. Soil samples from around the pipe identified the presence of Co-60 and Cs-137 and the excavation of the pipe continued. The area of maximum contamination was measured at 25-35 mR/hr (specific radionuclide data not available), with a hot spot of 29,300 pCi/gm Co-60 in this same area. The pipe from the edge of the old PCA (Potentially Contaminated Area) building to the edge of the waste disposal building and approximately 420 ft^3 of dirt and rock were removed as radioactive waste. The soil remaining at the bottom of the excavation contained Co-60 at an average concentration of 30 pCi/gm.
Impacted Areas – **WST-01/WST-02/WST-03**
Impacted Sub-surface Areas/Structures – **WST-02 at a depth in excess of 9 feet below grade, activity remains potentially in excess of the soil DCGL. WST-03 at ash dewatering sump in drumming pit. Decommissioning standards had not yet been developed at the time this partial remediation was performed. Radiological decay since 1984 may have reduced the radionuclide concentration below the soil DCGL. Further scoping data will be collected below the 9 foot clean backfill to confirm this evaluated condition.**

PIR 94-03 & 94-09.
Leakage from Frozen Fuel Chute Dewatering Line and NST Tell-tales
On February 17 and 18, 1994, a fuel chute dewatering line and a neutron shield tank telltale drain line ruptured due to freezing. A 3.5 liter sample from the fuel chute line indicated 1000 net cpm, and a sample from the NST telltale line indicated the presence of Co-60 and Cs-137. The ground below the rupture, as well as the area adjacent to the railroad tracks and pumpback house, showed no contamination. However, the snow pile along the south side of the rails by the new

fuel vault indicated the presence of Co-60, Cs-137 and Mn-54. All snow piles with positive radiation measurements were sent to the rad drains and the areas de-posted.

***Impacted Area* – NOL-01**

Appendix 2B

Impacted Area Assessments
Buildings, Structures and Open Land Areas Inside of the RCA

Buildings

<u>Old PCA Warehouse (WST-01)</u>

Description: WST-01 is a concrete block structure constructed on a reinforced concrete foundation. It contains a reinforced concrete tank/tub fitted with a drain that connects to the floor drain and continues to the Waste Disposal Building ash de-watering sump. It also had a locally-controlled ventilation system located in the northeast corner of the structure.

History: WST-01 was constructed for use as an equipment decontamination and storage facility. It was subsequently converted to a contaminated area used for radioactive material storage only. It was later decontaminated and is now used as a hazardous and mixed waste storage location. The decontamination tub was generally used for items considered heavily contaminated. These include control rod dash-pots and other components of moderate size from the primary systems. The glue in the joints of the drain line from this tub failed to hold over time and the use of the tub was discontinued. This drain line was partially remediated in 1984 during construction of the Radwaste Warehouse (WST-02). The area directly under the tub remains to be investigated.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area WST-01 are Co-60, Cs-137, Ag-108m, Ni-63, Sr-90 and H-3.
2. Media: Reinforced concrete structure (slab), sub-floor soil, sub-surface structures
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-floor soils

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in WST-01 include:
   - Closing of the tank/tub and floor drain system to inputs.
   - Removal of the local ventilation system.
   - Decontamination activities
   - Painting of the structure interior
2. Planned: Planned decommissioning activities for the WST-01 include demolition of walls to elevation 1035’-6”.
3. Anticipated End State Configuration: The end state configuration of WST-01 is anticipated to include:
   - Reinforced concrete structures (slab)
   - Subsurface concrete structures (foundations)
   - Sub-floor soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area WST-01 is identified as a Class 1 Area.

Radwaste Warehouse (WST-02)

Description: WST-02 is a steel frame and concrete block structure constructed on a reinforced concrete foundation. WST-02 is bounded by WST-04 and WST-03 on the north; NSY-07, NOL-03 and WST-01 on the east; NOL-04 on the south; and NOL-05 on the west.

History: WST-02 was constructed for use as a radioactive waste storage facility. However, it is normally maintained as a non-contaminated area. Contaminating events have occurred in WST-02.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area WST-02 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil.
3. Continued Investigation: Continued investigation will evaluate reinforced concrete surface structures, subsurface structures, systems and subsurface soil.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in WST-02.
2. Planned: Planned decommissioning activities for the WST-02 include demolition of walls to elevation 1035’-6.”
3. Anticipated End State Configuration: The end state configuration of WST-02 is anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area WST-02 is identified as a Class 1 Area.

Waste Disposal Building (WST-03)

Description: WST-03 is a steel frame and concrete block structure constructed on a reinforced concrete foundation. WST-03 is bounded by NOL-05 on the north, NSY-07 on the east, WST-02 on the south, and WST-04 on the west.

History: WST-03 was constructed for use as a radioactive waste processing and storage facility. It was normally maintained as a contaminated area. Contaminating events have occurred in WST-03.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area WST-03 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil and subsurface structures.

3. Continued Investigation: Continued investigation will evaluate reinforced concrete surface structures, subsurface structures, systems and subsurface soil.

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in WST-03 include:
   - Removal of all waste processing systems
   - Removal of floor drains
   - Removal of floors
   - Removal of sub-floor soils
   - Backfill of soil removal areas.
2. Planned: Planned decommissioning activities for the WST-03 include demolition of walls to elevation 1035'-6".
3. Anticipated End State Configuration: The end state configuration of WST-03 is anticipated to include:
   - Surface concrete structures (slabs)
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area WST-03 is identified as a Class 1 Area.

<u>Waste Compactor Building (WST-04).</u>

Description: WST-04 is a steel frame and concrete block structure constructed on a reinforced concrete foundation. WST-04 is bounded by NOL-05 on the north, WST-03 on the east, WST-02 on the south, and NOL-05 on the west.

History: WST-04 was constructed for use as a radioactive waste processing and storage facility. It was normally maintained as a non-contaminated area. Contaminating events have occurred in WST-04.

Contamination
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area WST-04 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil and subsurface structures.
3. Continued Investigation: Continued investigation will evaluate reinforced concrete surface structures, subsurface structures, systems and subsurface soil.

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in WST-04 include the removal of all waste processing systems
2. Planned: Planned decommissioning activities for the WST-04 include demolition walls to elevation 1035'-6".

3. Anticipated End State Configuration: The end state configuration of WST-04 is anticipated to include:
   - Surface concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area WST-04 is identified as a Class 1 Area.

Service Building RCA (SVC-02)

Description: SVC-02 is bounded by SVC-01 and SVC-03 on the north, SVC-03 and OMB-04 on the east, OOL-12 and OOL-01 on the south and NSY-01 and TBN-01 on the west. SVC-02 consists of a steel frame and concrete block structure. Sink and floor drain located in SVC-02 are contaminated and connect to the Liquid Waste Disposal System in NSY-11.

History: The systems present and the processes performed in SVC-02 did involve radioactive materials. Contaminating events did occur in SVC-02. SVC-02 has served as the primary entrance and egress from the RCA during most of the plant history.

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area SVC-02 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-surface soils

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in SVC-02 include the removal of equipment.
2. Planned: Planned decommissioning activities for the SVC-02 include demolition of walls to elevation 1022'-8".
3. Anticipated End State Configuration: The end state configuration of WST-04 is anticipated to include:
   - Surface concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history and as a result of the decommissioning activities performed to date, survey area SVC-02 is identified as a Class 1 Area.

East Primary Auxiliary Building (AUX-01)

Description: AUX-01 consists of that portion of the PAB designed to contain the radiological constituents resulting from operation of the primary (radioactive) systems of the plant. The design of the AUX-01 portion of the PAB provided for collection and control of radioactive liquid and gaseous spills or releases that occurred within this portion of the PAB. All areas within AUX-01 have floor drains that channel liquids to the radwaste system and are ventilated through the Primary Ventilation Stack. AUX-01 is bounded by NOL-01 on the north, NSY-02 on the east, NOL-02 on the south and AUX-02 on the west. The structure is constructed of reinforced concrete.

History: The PAB was identified as a contaminated area shortly after the initial criticality of the YNPS reactor, as a result of a pipe leak. Over the operating history of the YNPS this portion of the plant has been maintained as a contaminated area.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area AUX-01 are Co-60, Cs-137, Ag-108m, Sr-90, Fe-55, Ni-63, Am-241, and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-surface soils

Decommissioning/Decontamination Activities

1. Performed: The decommissioning activities performed have removed all radiologically contaminated piping, pumps, tanks, and other system components from AUX-01. In addition concrete surfaces have been de-contaminated via surface removal techniques.
2. Planned: Planned decommissioning activities for the PAB structure include demolition of the west, north and east walls to grade elevation on the north side of the building and demolition of the south wall to grade elevation on the south side of the building.
3. Anticipated End State Configuration: The anticipated end state configuration will consist of reinforced concrete floor slabs, foundations, surface structures below the north grade elevation and the south wall up to the south grade elevation (a difference of about 13 feet) and adjacent sub-surface soils.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area AUX-01 is identified as a Class 1 Area.

West Primary Auxiliary Building (AUX-02)

Description: AUX-02 consists of that portion of the PAB that was not designed to contain portions of the primary (radioactive) operating systems of the plant. The design of the AUX-02 portion of the PAB did not provide for collection and control of radioactive liquid and gaseous spills or releases, if they occurred within this portion of the PAB. All areas within AUX-02 had floor drains that channeled liquids to the storm drain system. These spaces are not ventilated through the Primary Ventilation System. AUX-02 is bounded by NOL-01 and NOL-06 on the north, AUX-01 on the east, NOL-05 on the south and NOL-06 and NSY-03 on the west. The structure consists of a steel frame and block wall construction.

History: The AUX-02 area of the PAB was identified as a contaminated as a result of a cross-contaminating event where water spilled from the seal water system vent. Contamination of AUX-02 also occurred when the Safety Injection Tank heating system pump leaked resulting in contamination of the floor and floor drains in the lower level of the PAB. Over the operating history of the YNPS, this portion of the plant has been decontaminated, in order to maintain it as a non-contaminated area.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area AUX-02 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-surface soils

Decommissioning/Decontamination Activities

1. Performed: All piping, pumps, tanks, and other system components have been removed from AUX-02 with the exception of the Roof Drain System. In addition concrete surfaces have been de-contaminated via surface removal techniques.
2. Planned: Planned decommissioning activities for the AUX-02 structure include demolition of the west, north and east walls to the north grade elevation and the demolition of the south wall to the south grade elevation (similar to AUX-01).
3. Anticipated End State Configuration: The anticipate end state configuration will consist of reinforced concrete floors, foundations, surface structures below the north grade elevation and the south wall below the south grade elevation including adjacent sub-surface soils.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area AUX-02 is identified as a Class 1 Area.

Spent Fuel Pit (SFP-01)

Description: SFP-01 is a steel frame and metal panel structure built atop the reinforced concrete Spent Fuel Pit. SFP-01 is bounded by NOL-01 on the north, SFP-02 on the east, NSY-02 on the south, and NSY-09 and NOL-01 on the west.

History: SFP-01 was constructed for use as a wet spent fuel storage facility. It was normally maintained as a contaminated area. Contaminating events have occurred in SFP-01 that resulted in contamination of the outside of the structure. This survey area also includes appurtenances such as the Fuel chute lower lock valve assembly ("Woodchuck hole") and the fuel chute de-watering pump pad.

Contamination
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area SFP-01 are Co-60, Cs-137, Ag-108m, Sr-90, C-14, Fe-55, Am-241, Pu-238, Pu-239/240, Pu-241 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil and subsurface structures.
3. Continued Investigation: Continued investigation will evaluate reinforced concrete surface structures, subsurface structures, systems and subsurface soil.

Decommissioning/Decontamination Activities
1. Performed: The fuel transfer chute has been isolated from the VC. The pit has been drained and a preliminary decontamination has been performed to allow removal of the stainless steel pit liner.
2. Planned: Planned decommissioning activities for the SFP-01 include demolition walls to elevation grade. Continued investigation of the extent of the residual concrete contamination may result in complete removal of this structure.
3. Anticipated End State Configuration: The end state configuration of SFP-01 is currently anticipated to include:
   - Surface concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area SFP-01 is identified as a Class 1 Area.

New Fuel Vault (SFP-02)

Description: SFP-02 is a concrete block structure built on a reinforced concrete foundation. SFP-02 is bounded by NOL-01 on the north and the east, NSY-02 and NOL-02 on the south, and SFP-01 on the west.

History: SFP-02 was constructed for use as a new fuel storage facility. It was normally maintained as a contaminated area.

Contamination
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area SFP-02 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil and subsurface structures.
3. Continued Investigation: Continued investigation will evaluate reinforced concrete surface structures, subsurface structures, systems and subsurface soil.

Decommissioning/Decontamination Activities
1. Performed: No decommissioning activities have been performed in SFP-02.
2. Planned: Planned decommissioning activities for the SFP-02 include demolition of walls to elevation 1022’-8”.

3. Anticipated End State Configuration: The end state configuration of SFP-02 is anticipated to include:
   - Surface concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area SFP-02 is identified as a Class 1 Area.

Yard Structures

VC/Reactor Support Structure (BRT-01)

Description: BRT-01 is enclosed by NOL-01 on the east and NOL-06 on the west. BRT-01 consists of reinforced concrete structures remaining after demolition of the Lower Pipe Chase, the Fuel Transfer Chute Support, Vapor Container (VC) and the Reactor Support Structure (RSS). This includes the following:

- The two, reinforced concrete RSS leg bases that protrude out of the RSS mat foundation.
- The six, reinforced concrete RSS leg bases that protrude out of the RSS ring beam foundation.
- The sixteen, reinforced-concrete bases that support the VC legs
- The Lower Pipe Chase Support and foundation.
- The Fuel Transfer Chute Support and foundation.

The VC formerly contained the primary reactor systems such as the reactor vessel and steam generators. All of these primary system components have been removed leaving, as of September 2003, the concrete shield tank cavity structure surrounded by the steel sphere of the VC. The VC and support legs will be removed from site as radioactive waste leaving only the items listed above as an end state condition subject to these residual structures meeting the license termination criteria.

History: All the structures within BRT-01 have the same potential for being contaminated by work activities performed in the area. With the exception of the six leg RSS bases on the ring beam, the structures that comprise BRT-01 are of original plant construction. The six leg RSS bases on the ring beam received a seismic upgrade modification in 1979.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area BRT-01 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete.

Decommissioning/Decontamination Activities:

1. Performed: No decommissioning activities have been performed on BRT-01. Primary systems have been removed from the VC.
2. Planned: Planned decommissioning activities for the BRT-01 include the demolition of the BRT-01 related structures down to grade (elevation 1022'-8"). The VC and supporting legs will be removed.
3. Anticipated End State Configuration: The anticipate end state configuration will consist of reinforced concrete support structures below 1022' -8".

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities planned, survey area BRT-01 is identified as a Class 1 Area.

North and South Decon Pads and Fuel Transfer Enclosure (NSY-01)

Description: NSY-01 consists of the following portions of the Service Building: the former north and south decontamination rooms and the recent addition to the structure south of the Hot Machine Shop (all of which are now referred to as the Fuel Transfer Enclosure, or FTE). The former north decon room consists of a reinforced concrete floor and concrete block walls. The former south decon room consists of a reinforced concrete floor surrounding a steel clad decontamination pad, with a drain trench around the perimeter. The walls of the former south decon room were removed and replaced with insulated metal panel and steel frame construction. The addition south of the former hot machine shop consists of the reinforced concrete floor and insulated metal panel and steel frame walls.

History: The FTE was used for closure of the NAC Nuclear Fuel Transportable Storage Canister in preparation for placement into the Vertical Concrete Casks (VCCs). Portions of the FTE were maintained as a contaminated areas. Prior to construction of the FTE the north and south decon rooms were used to decontaminate and perform maintenance on plant components, tools and equipment. This area was also used for preparation of waste shipping containers/casks. The north and south decon rooms were generally maintained as a contaminated area.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-01 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in NSY-01
2. Planned: Planned decommissioning activities for the FTE include demolition the structure down to elevation 1022'-8" and decontamination or removal of the decon pads.

3. Anticipated End State Configuration: The end state configuration of the FTE is anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area NSY-01 is identified as a Class 1 Area.

Ion Exchange Pit (NSY-02)

Description: NSY-02 consists of the concrete structure that contained the purification system ion exchange vessels and filter capsules in a water-filled shield tank and an adjoining valve gallery and pipe chase that connected the Ion Exchange Pit (IX Pit) to the PAB. In addition, survey area NSY-02 includes the stairway leading to the foyer of the east PAB cubicle corridor access. The north wall of the IX Pit and the south wall of the Spent Fuel Pit (SFP-01) are a common wall. The east wall of the IX Pit abuts NOL-02. The south line of the IX Pit also abuts NOL-02. The west line of the IX Pit abuts AUX-01 and NOL-01.

History: Survey area NYS-02 (IX Pit) became contaminated as a result of purification system leakage into the shield water in the IX Pit and as a result of inadvertent misalignment of valves. The IX Pit itself leaked as a result of a flawed concrete joint in the northwest corner where it attaches to the SFP and the VC elevator foundation. This leak was repaired in 1965. It was also contaminated by spills during ion exchange resin transfers.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-02 are Co-60, Cs-137, Ag-108m, Sr-90, C-14, Fe-55, Am-241, Pu-238, Pu-239/240, Pu-241 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil.
3. Continued Investigation: Reinforced concrete surface and subsurface structures, sub-surface soil.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NSY-02 include:
   - Removal of the purification system valves, piping and pipe supports
   - Concrete shield blocks
   - Ion exchange vessels and filter capsules
   - Decontamination via surface removal of the interior surfaces of the IX Pit and the valve gallery
2. Planned: Planned decommissioning activities for the IX Pit included demolition the structure down to elevation 1022'-8" along the north and west walls, and to 1035'-6" on the east and south walls. An investigation of the impact of the IX Pit leakage during early plant operations on the adjacent open land survey areas (NOL-01 and NOL-02) with regard to the path of leakage into subsurface soils and into the groundwater will be

conducted in accordance with section 2-5 (Continuing Investigation of Subsurface Contamination) and section 2-6 (Continuing Investigation of Groundwater Contamination) concurrent with the subsurface investigation of the Spent Fuel Pit (SFP-01).

3. Anticipated End State Configuration: The end state configuration of the IX Pit is anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-02 is identified as a Class 1 Area.

<u>Safety Injection and Diesel Generator Building (NSY-03)</u>

Description: NSY-03 consists of the remainder of the Safety Injection/Diesel Generator Building (SI/DG) and includes the #3 Battery and MCC rooms. NSY-03 is bounded by NOL-06 on the north, south, and west and AUX-02 on the east. The original storm drain system and an electrical duct bank ran under the SI/Diesel building.

History: The SI/Diesel Building was constructed in 1970, adjacent to the northeast corner of the PAB. This location is suspected of having been contaminated prior to construction of the SI/Diesel Building. The SI/Diesel building contained radioactive systems that caused minor contamination of the floor area. The safety injection pumps leaked to a pump pedestal drain that was connected to a sump that was pumped to the gravity drain tank in the PAB. This drain system leak radioactive liquids into the surrounding soils under the SI/Diesel building floor.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-03 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete structures and subsurface structures, systems and soil

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NSY-03 include:
      - Removal of the Safety injection system piping, valves, pumps and controls
      - Removal of the floor drain and surrounding soils
      - Removal of the Diesel Generators and support systems
      - Removal of the #3 Battery and MCC
      - Removal of the electrical distribution systems in manhole #3
      - Removal of the walls and roof of the SI/Diesel building and the #3 Battery Room and MCC.

2. Planned: Planned decommissioning activities for NSY-03 included demolition the structure down to grade.
3. Anticipated End State Configuration: The end state configuration of NSY-03 is anticipated to include:
   - Reinforced concrete structures (floor slabs)
   - Subsurface concrete structures (foundations, electrical duct banks)
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-03 is identified as a Class 1 Area.

Safe Shutdown System Building (NSY-04)

Description: NSY-04 consists of the Safe Shutdown System (SSS) Building. The SSS Building was constructed in 1985 in a portion of the RCA that had been temporarily cleared to facilitate its construction in a clean area. NSY-04 is bounded entirely by NOL-05.

History: Prior to 1985, the location of the SSS building was part of the RCA that was down grade from the radwaste storage area. The Safe Shutdown System Building became contaminated as a result of a radioactive liquid spill in 1985. The spill was cleaned-up and the building was subsequently maintained as a non-contaminated area.

Contamination
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-04 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in NSY-04 include:
   - Removal of the SSS diesel generator and support systems
   - Removal of the SSS control panel and electrical distribution system.
   - Removal of the SSS pumps, piping and tanks.
   - Removal of a portion of the floor and contaminated soil under the floor.
2. Planned: Planned decommissioning activities for NSY-04 include demolition of the structure to elevation 1034’-0”.
3. Anticipated End State Configuration: The end state configuration of NSY-04 anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-04 is identified as a Class 1 Area.

Firewater Storage Tank and Diesel Fire Pump House (NSY-05)

Description: NSY-05 consists of the Firewater Storage Tank and Diesel Fire Pump House, constructed in 1979 in a portion of the RCA. NSY-05 is bounded entirely by NOL-04.

History: Prior to 1979, the location of the Firewater Storage Tank and Diesel Fire Pump House were on the edge of the RCA, down slope from the Radwaste Storage Area. The Firewater Storage Tank and Diesel Fire Pump House have not been surveyed on a routine basis. The Firewater Storage Tank and Diesel Fire Pump House, although located in the RCA, are not considered radioactively contaminated structures.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-05 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil subsurface systems.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in NSY-05.
2. Planned: Planned decommissioning activities for NSY-05 include removal of the firewater storage tank, diesel driven pump and pump house.
3. Anticipated End State Configuration: The end state configuration of NSY-05 anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area NSY-05 is identified as a Class 1 Area.

New PCA Storage Building (NSY-06)

Description: NSY-06 consists of a pre-fabricated metal building that was constructed in 1975 in a portion of the RCA. NSY-06 is bounded by NOL-03 on the north, south, and east and bounded by NOL-04 on the west.

History: Prior to 1975, the location of the New PCA Storage Building was on the edge of the RCA and down slope from the radwaste storage area. NSY-06 was used as a radioactive material storage area and occasionally as a contaminated work area.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-06 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil subsurface systems.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in NSY-06.
2. Planned: Planned decommissioning activities for NSY-06 include removal of the metal structure.
3. Anticipated End State Configuration: The end state configuration of NSY-06 anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area NSY-06 is identified as a Class 1 Area.

<u>Radioactive Waste Storage Tank Moat Area (NSY-07)</u>

Description: NSY-07 consists of a concrete structure that provided a secondary containment function for radioactive liquids and provided shielding from radioactive liquids stored in the tanks. A drain valve isolated the moat area from the east storm drain system. NSY-07 is bounded by NOL-02 on the north, NOL-03 on the east and south, and WST-03 on the west.

History: NSY-07 is part of the original plant structure. NSY-07 was contaminated by a pipe leak during early plant operations.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-07 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities

1. Performed: Decommissioning work activities performed under Decommissioning Work Packages (DWPs) include removal of tank 31 (Waste Hold-up Tank), tank-32 (Activity Dilution and Decay Tank).
2. Planned: Planned decommissioning activities for the NSY-07 includes demolition the structure down to grade.
3. Anticipated End State Configuration: The end state configuration of the NSY-07 is anticipated to include:
   - Reinforced concrete structures

- Subsurface concrete structures
- Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-07 is identified as a Class 1 Area.

New Safety Injection Tank Pad (NSY-08)

Description: NSY-08 consists of the New Safety Injection (SI) Tank Pad, constructed in 1991 in a portion of the RCA. NSY-08 is bounded entirely by NOL-05.

History: Prior to 1991, the location of the New SI Tank Pad was at the edge of the RCA and down slope from the Radwaste Storage Area. The new SI tank developed a leak from a temperature monitoring well located on the eastside of the tank. This leak resulted in minor contamination of the side of the tank and a portion of the tank pad.

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-08 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil subsurface systems.

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in NSY-08 include:
    a. Removal of the New SI Tank
    b. Removal of the SI Tank piping.
2. Planned: Planned decommissioning activities will depend on the results of the continuing investigation.
3. Anticipated End State Configuration: The end state configuration of NSY-08 is anticipated to include:
    - Reinforced concrete structures
    - Subsurface concrete structures
    - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-08 is identified as a Class 1 Area.

VC Elevator Foundation (NSY-09)

Description: NSY-09 consists of the foundation of the VC elevator structure.

History: NSY-09 is part of the original plant structure. The interior surface of NSY-09 was contaminated by the presence of loose contamination within the elevator shaft. The exterior of NSY-09 was likely contaminated by a leak from the Ion Exchange Pit (NSY-02).

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-09 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities
1. Performed: No decommissioning activities have been performed in NSY-09.
2. Planned: Planned decommissioning activities for the NSY-09 includes demolition the structure down to elevation 1022’-8”.
3. Anticipated End State Configuration: The end state configuration of the NSY-09 is anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area NSY-09 is identified as a Class 1 Area.

ISFSI Pad (NSY-10)

Description: NSY-10 is the ISFSI Pad, constructed in 1999 on the former location of the Pole Barn. NSY-10 is bounded entirely by NOL-07. The design and function of the VCC is such that no contamination of the ISFSI should result from their presence on the ISFSI.

History: Prior to 1999, this location was used for storage of materials and equipment some of which were radioactive materials. During construction of the ISFSI pad, a radiological assessment of some areas north of the pad (notably the NOL-03 and NOL-04 yard areas and the above grade exterior walls of structures within them) was performed using a technologically advanced method. The assessment was performed in anticipation that area background would be impacted by transfer of the fuel to the ISFSI pad. The ISFSI pad is now occupied by loaded VCC. The transportation of the loaded VCC was performed under strict controls to ensure that the transport process would not contaminate the ISFSI. The ISFSI is surveyed on a routine basis and it is anticipated to remain non-contaminated as a result of the presence of the VCC. Should future surveys identify the presence of contamination on the ISFSI pad then the survey area may be re-classified.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-10 are Co-60, Cs-137, Sr-90.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil subsurface systems.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning work performed under DWPs included removal of the Pole Barn and re-grading of the surface to facilitate ISFSI pad and road construction. Soils removed from the area were deposited primarily in Survey Areas OOL-07 and OOL-09. Soils from the roadway approach area were deposited in Survey Areas OOL-02 and OOL-10. .
2. Planned: Planned decommissioning activities will depend on the results of the investigation conducted when the ISFSI is taken out of service.
3. Anticipated End State Configuration: The end state configuration of NSY-10 anticipated to include:
   - Reinforced concrete structures
   - Subsurface concrete structures
   - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-10 is identified as a Class 3 Area.

<u>Chem-waste Transfer Pump Pit (NSY-11)</u>

Description: NSY-11 consists of a concrete vault, which houses the liquid waste transfer pumps that support the decon-room drains, the RP control point drains and the chemistry laboratory drains. NSY-11 is bounded entirely by NOL-01.

History: NSY-11 is part of the original plant structure. NSY-11 was contaminated by leaks and/or spills that occurred during early plant operations.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-11 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in NSY-11
2. Planned: Decommissioning activities for the NSY-11 will depend upon the results of the continuing investigation.
3. Anticipated End State Configuration: The end state configuration of the NSY-11 is anticipated to include:
    - Reinforced concrete structures
    - Subsurface concrete structures
    - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area NSY-11 is identified as a Class 1 Area.

<u>Tank-1 Base and Pipe Chase (NSY-12)</u>

Description: NSY-12 consists of the base for Tank-1 (TK-1) and a subsurface pipe chase that connects the TK-1 base to the Auxiliary Boiler Room in the Turbine Building. NSY-12 is bounded entirely by NOL-06.

History: NSY-12 is part of the original plant structure. There is no documentation indicating that NSY-12 is contaminated; however, there is information that indicates that the area around NSY-12 is potentially contaminated.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-12 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NSY-12 include removal of TK-1 and related systems.
2. Planned: Decommissioning activities for the NSY-12 will depend upon the results of the continuing investigation.
3. Anticipated End State Configuration: The end state configuration of the NSY-12 is anticipated to include:
    - Reinforced concrete structures
    - Subsurface concrete structures
    - Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history and as a result of the decommissioning activities performed to date, survey area NSY-12 is identified as a Class 1 Area.

Tank-39 Base Demineralized Water Storage Tank (NSY-13)

Description: NSY-13 consists of the base for Tank-39 (TK-39). NSY-13 is bounded entirely by NOL-02.

History: NSY-13 is part of the original plant structure. There is a history of tritium being detected in the tank water but no other radionuclides. The tank has recently been drained. There is information that indicating that the area around NSY-13 (tank base) is potentially contaminated.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NSY-12 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, sub-surface soil
3. Continued Investigation: Reinforced concrete, surface soil, sub-surface soil

Decommissioning/Decontamination Activities

1. Performed: No Decommissioning activities have been performed in NSY-13.
2. Planned: Decommissioning activities planned for NSY-13 will include the removal of Tank-39. Disposition of the concrete tank base will depend upon the results of the continuing investigation.
3. Anticipated End State Configuration: The end state configuration of the NSY-13 is anticipated to include:

- Reinforced concrete structures
- Subsurface concrete structures
- Subsurface soil.

Classification Statement: Based upon the radiological condition of this survey area identified in the operating history, survey area NSY-13 is identified as a Class 1 Area.

Open Land Areas

Eastern Lower RCA Yard (NOL-01)

Description: NOL-01 is the land area within the RCA that is bounded by NOL-06, the FTE and Service Building on the north; the east boundary of the RCA (OOL-12) to the east; NOL-02, the New Fuel Vault/Spent Fuel Pit and the PAB on the south; and NOL-06 on the west. The bounds of NOL-01 were established such that it is appropriately sized as a Class 1 survey unit according to MARSSIM. Subsurface structures within NOL-01 will be surveyed as a survey unit within the survey area.

History: In addition to the normal migration of minor levels of contamination in the RCA NOL-01 was contaminated by the following events:

- Overfilling of the Spent Fuel Pit.
- Leaks associated with fuel transfer chute pump.
- A Reactor Head removal contamination event.
- Leakage from the IX Pit during early plant operations

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-01 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil and sub-surface concrete.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures, systems and the extent of contamination in soil from known spill events as described in Sections 2.2.3 and 2.2.4.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NOL-01 include the construction of the landing pads for reactor vessel removal and fuel transfer casks, construction of the Spent Fuel Pit Security Blast Shield Wall (this entailed some remediation of contaminated soils disposed of as radioactive waste), and installation of Auxiliary Service Water System..
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey are sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history and as a result of the decommissioning activities performed to date, survey area NOL-01 is identified as a Class 1 Area.

Northeast Upper RCA Yard (NOL-02)

Description: NOL-02 is the land area within the RCA that is bounded by the Exchange Pit/New Fuel Vault and NOL-01 on the north, the east boundary of the RCA (OOL-11) to the east, NOL-03 and the Liquid Waste Storage Tanks (NSY-07) on the south and the NOL-05 and Waste Disposal on the west. The bounds of NOL-02 were established such that it is appropriately sized as a Class 1 survey unit according to MARSSIM. Subsurface structures identified within NOL-02 will be surveyed as a survey unit within the survey area.

History: In addition to the normal migration of minor levels of contamination in the RCA NOL-02 was contaminated or affected by the following events:

- A resin spill during resin transfer operation
- The inadvertent severing of a buried radwaste transfer piping
- Leak from piping associated with Test Tanks

- Release of Test Tank liquids during sample collection
- A subsurface break in the fire protection piping.
- Leakage from the IX Pit during early plant operations

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-02 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil and sub-surface concrete.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures, systems and the extent of contamination in soil from known spill events as described in sections 2.2.3 and 2.2.4.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NOL-02 include:
    - Removal of the Test Tanks
    - Removal of the Monitor Tanks
    - Removal of waste transfer piping.
    - Removal of contaminated soils identified in area of the test tanks.
    - Backfill of excavations with surveyed clean soil.
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey will be sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area NOL-02 is identified as a Class 1 Area.

Southeast Upper RCA Yard (NOL-03)

Description: NOL-03 is the land area within the RCA that is bounded by the NOL-02 and the Liquid Waste Storage Tanks (NSY-07) on the north, the east boundary of the RCA (OOL-11) to the east, OOL-10 on the south and the NOL-04 and the radwaste warehouse complex on the west. The bounds of NOL-03 were established such that it is appropriately sized as a Class 1 survey unit according to MARSSIM. Subsurface structures and system identified within NOL-03 will be surveyed as a survey unit within the survey area.

History: In addition to the normal migration of minor levels of contamination in the RCA NOL-03 was contaminated by the storage of radioactive material.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-03 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil and sub-surface concrete.

3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures, systems and the extent of contamination in soil.

Decommissioning/Decontamination Activities:

1. Performed: Decommissioning activities performed in NOL-03 have removed contaminated soils identified in area of radioactive material storage. Excavations were backfilled with surveyed clean soil.
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey will be sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area NOL-03 is identified as a Class 1 Area.

<u>Southwest Upper RCA Yard (NOL-04)</u>

Description: NOL-04 is the land area within the RCA that is bounded by the NOL-05 and the radwaste warehouse on the north, NOL-03 and NSY-06 on the east, OOL-10 on the south and west. NOL-04 is appropriately sized as a Class 1 survey unit according to MARSSIM. Subsurface structures and system identified within NOL-04 will be surveyed as a survey unit within the survey area.

History: In addition to the normal migration of minor levels of contamination in the RCA NOL-04 was contaminated by temporary storage of packaged radioactive material awaiting shipment.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-04 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil and sub-surface concrete.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures, systems and the extent of contamination in soil.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in NOL-04
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey will be sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area NOL-04 is identified as a Class 1 Area.

Northwest Upper RCA Yard (NOL-05)

Description: NOL-05 is the land area within the RCA that is bounded by the NOL-06 and the PAB on the north, NOL-02 and the waste disposal and radwaste warehouse on the east, NOL-04 on the south and OOL-10 on the west. NOL-05 is appropriately sized as a Class 1 survey unit according to MARSSIM. Subsurface structures and systems identified within NOL-05 will be surveyed as a separate survey unit within the survey area.

History: In addition to the normal migration of minor levels of contamination in the RCA NOL-05 was contaminated by radioactive liquid leakage from the original plant Safety Injection Tank.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-05 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil and sub-surface concrete.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures, systems and the extent of contamination in soil.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NOL-05 include:
    - Removal buried piping connecting the Safe Shutdown System Building to the PAB
    - Removal of both the original and new Safety Injection Tanks
    - Removal of the piping connecting the Safety Injection Tanks to the PAB.
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey will be sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area NOL-05 is identified as a Class 1 Area.

Western Lower RCA Yard (NOL-06)

Description: NOL-06 is the land area within the RCA that is bounded by the OOL-10 and the Turbine Building on the north; the FTE, NOL-01 and the PAB on the east; NOL-05 on the south; and OOL-10 on the west. The bounds of NOL-06 were established such that it is appropriately sized as a Class 1 survey unit according to MARSSIM. Subsurface structures and system identified within NOL-06 will be surveyed as a survey unit within the survey area.

History: NOL-06 was contaminated by the normal migration of minor levels of contamination in the RCA.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-06 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil and sub-surface concrete.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures, systems and the extent of contamination in soil.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in NOL-06 include:
   - Remediation of mixed waste along the south wall of the SI/Diesel Building.
   - Construction of the Fuel Transfer Haul road under the VC.
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey will be sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area NOL-06 is identified as a Class 1 Area.

<u>ISFSI RCA Yard (NOL-07)</u>

Description: NOL-07 is the land area that bounds the ISFSI pad and bounded entirely by OOL-10

History: NOL-07 was constructed at the same time as the ISFSI. A comprehensive radiological assessment of this area was performed prior to construction of the ISFSI. Previously this area was used as a material storage area. Some of this material was later identified as radioactive material. A survey of this area under the guidelines of NUREG/CR-5849 was conducted prior to grading. Samples have been taken of each load of soils removed from the area. These samples showed no detectable activity. All soils removed from the area were deposited in survey areas OOL-07 (Class 2) and OOL-09 (Class 3).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area NOL-07 are Co-60, Cs-137, and Sr-90.
2. Media: Surface and subsurface soil.
3. Continued Investigation: Continued investigation will not be performed until the spent fuel and waste stored on the ISFSI has been removed.

Decommissioning/Decontamination Activities

1. Performed: Dismantlement of a pole barn structure and non-rad material storage area. The area was then graded in preparation for construction of the ISFSI pad. New concrete was used in the structure. Fuel Storage Casks have been placed on the pad and are in their final configuration.
2. Planned: Future decommissioning activities are dependent upon the results of continued investigations
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey will be sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area NOL-07 is identified as a Class 3 Area. It is not expected that any radioactive material will leave the confines of the fuel casks and residual contamination after removal of the fuel casks is anticipated to be a small fraction of the DCGLs.

Appendix 2C
Impacted Area Assessments
Buildings, Structures and Open Land Areas Outside of the RCA

Buildings and Structures

Screen-well Pump House (OMB-01)

Description: OMB-01 is a YNPS structure located on U S Gen owned property. OMB-01 is located within the bounds of survey area OOL-03, a Class 3 land survey area. OMB-01 consists of reinforced concrete that forms the intake and screen-well structure below grade and steel frame and block structure that housed the pump motors and controls above grade. The intake structure connects to Sherman Reservoir through a corrugated metal pipe. The pump discharge connects to the turbine building through 84 inch diameter concrete pipe.

History: The systems present and the processes performed in OMB-01 did not involve radioactive materials. There is no information that identifies the presence of radioactive materials in OMB-01. Access to OMB-01 is through OOL-03 a Class 3 land survey area, there is a potential that contamination may have been translocated to OMB-01 from OOL-03.

Draft NUREG/CR-5849 based surveys performed 9/2/98 identified no licensed radioactivity present.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OMB-01 are Co-60, Cs-137, Sr-90 and H-3 resulting from the intake of waters and sediments at Sherman Reservoir. Sherman Reservoir receives the discharge of the circulating water system, which includes permit released liquid radioactive effluents. The East Storm Drain System also discharges to Sherman pond, which is known to contain radioactivity from surface run-off from within the RCA.
2. Media: Reinforced concrete, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-surface soils. Potential for migration of radioactivity exists from groundwater movement along the backfill around the circulating water system piping located below the Turbine Building. This will be investigated by core bore sampling of soils adjacent to and under the circulating water system piping under the Turbine Building.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OMB-01 include:
   - Removal of the circulating water pump motors and impellers.
   - Removal of the circulating water pipes within the structure.
   - Removal of the traveling screen equipment in the Intake Structure
   - Removal of the service water pumps and pipes within the structure.
   - Installation of the Auxiliary Service Water System (ASWS)
2. Planned: No further decommissioning activities are planned at this time.

3. Anticipated End State Configuration: OMB-01, if present, will be surveyed as it currently exists: Reinforced concrete, concrete blocks, and structural steel. This structure may be removed in its entirety, subject to FERC approval.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the HSA, survey area OMB-01 is identified as a Class 3 Area.

Security Gatehouse and Diesel Generator Building (OMB-02)

Description: OMB-02 is located on YAEC owned property. OMB-02 is bounded by OOL-02 on the north, east, south and OOL-06 on the west. OMB-02 consists of reinforced concrete and block structures. OMB-02 functions as the access control point for the YNPS site. It also now houses the YNPS control room. Survey area OMB-02 also includes the Security Diesel Generator Building that supplies emergency power to the Gatehouse and the ISFSI. A portion of the West Storm Drain System runs under OMB-02. The potable water and sanitary sewer systems connect to OMB-02.

History: The use of radioactive materials in OMB-02 involved electro-plated or sealed check sources for instrument response verification. There is information that identifies events involving radioactive material present in OMB-02 resulting from infrequent and unintentional translocation of plant related radioactivity into OMB-02 from within the RCA. Contamination monitors were operated at the gatehouse as a final check for radioactivity on personnel leaving the industrial area. When contamination was identified at the OMB-02 monitors it was cleaned-up and a post decontamination survey performed to verify no detectable residual contamination. It is anticipated that any residual contamination, if present in OMB-02, would not exceed a small fraction of the appropriate DCGLs.

Contamination
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OMB-02 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, concrete block, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-surface soils

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in OMB-02 include the relocation of the Control Room into the gatehouse.
2. Planned: No further decommissioning activities are needed at this time.
3. Anticipated End State Configuration: OMB-02 will be surveyed as it currently exists, a reinforced concrete and concrete block structure.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the HSA, survey area OMB-02 is identified as a Class 3 Area.

Administration Building (OMB-03)

Description: OMB-03 is located on YAEC owned property. OMB-03 is bounded entirely by OOL-06. OMB-03 consists of a metal frame and panel structure set on a reinforced concrete pad. OMB-03 functions as the Administration Office Building.

History: The systems present and the processes performed in OMB-03 did not involve use of radioactive materials. Over its history as a visitor center and training center, various radioactive materials were present in the building including electro-plated and sealed check sources and examples of naturally occurring radioactive materials and consumer products.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OMB-03 are Co-60 and Cs-137.
2. Media: Reinforced concrete, surface soil.
3. Continued Investigation: Continued investigation will evaluate the structure as it currently exists.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in OMB-03.
2. Planned: No further decommissioning activities are needed at this time.
3. Anticipated End State Configuration: OMB-03 will be surveyed as is; Reinforced concrete, structural steel and generic building materials.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OMB-03 is identified as a Class 3 Area.

Warehouse and Loading Dock (OMB-04)

Description: OMB-04 is located on YAEC-owned property. OMB-04 is bounded by OOL-02 on the north and east, OOL-12 on the south and SVC-03 on the west. OMB-04 consists of a metal frame and panel structure set on a concrete pad.

History: Although a single structure now, OMB-04 previously consisted of two structures: the original warehouse and a separate two bay garage. The warehouse and garage were connected by construction of an addition that spanned the gap between the east end of the warehouse and the west end of the garage. The construction of the Service Building Annex connected to the warehouse to the Service Building. A two-inch thick layer of concrete was poured over the existing floor of the warehouse as part of a loading dock modification.

OMB-04 was used as a storage location for plant equipment and materials and was not intended for storage of radioactive materials. There were incidents where radioactively contaminated equipment was inadvertently stored in OMB-04. The contamination consisted of loose radioactive material, resulting from the unintentional translocation of contaminated equipment into OMB-04 from the RCA. When these events were identified the radioactive contamination was cleaned-up with the results of decontamination verified through survey. It is anticipated that any residual contamination, if present in OMB-04, would not exceed a small fraction of the appropriate DCGLs.

Survey area OMB-04 is adjacent to a Class 1 open land area (OOL-12). The mode of contamination of OOL-12 was via surface water run-off from the inside the RCA. The entire surface of survey area OMB-04 is elevated above the prevailing grade of the surface water run-off pathway in survey area OOL-12. Consequently survey area OMB-04 was not impacted by this mode of contamination spread. Residual contamination in survey area OOL-12 is embedded into crevices of the rail bed and is not available for translocation by foot traffic.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OMB-04 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Concrete.
3. Continued Investigation: Continued investigation will evaluate the structure as is and the backfill surrounding the recently installed ASWS.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OMB-04 included:
   - Installation of the Auxiliary Service Water System.
   - The steel frame and panel structure has been demolished and removed from site.
2. Planned: none
3. Anticipated End State Configuration: Concrete floor slab and reinforced concrete loading dock structure.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OMB-04 is identified as a Class 3 Area.

Furlon House (OMB-05)

Description: OMB-05 is located on YAEC owned property. OMB-05 is bounded entirely by OOL-16. OMB-05 is a wood frame structure set on a stone and concrete foundation and was constructed prior to construction and operation of YNPS.

History: OMB-05 was used for storage of emergency response equipment trunks containing sealed packages of radioactive material (respirators, protective clothing, etc) The trunks also contained radioactive material in the form of electro-plated and sealed check sources used to verify instrument operability. The sealed packages were surveyed prior to placement into the trunks in storage at OMB-05.

After 9/11/01 OMB-05 was designated as the shipping and receiving location for the YNPS site. Radioactive material packages to be shipped are brought to OMB-05 in a condition ready for shipment, no preparation of packaged radioactive material is performed in OMB-05. No radioactive material packages are opened in OMB-05. Radioactive material packages received at the YNPS site are surveyed in accordance with transportation regulations to verify radioactive material package integrity prior to opening. Packages are opened inside the RCA.

Draft NUREG/CR-5849 based surveys performed in 1998 identified no licensed radioactivity detectable. Only naturally occurring radionuclides were identified during the scans of the lower walls and floors and total surface contamination measurements. No exposure rate or loose surface contamination measurements were obtained.
In August of 2003 the foundation was repaired on the south wall. Soils excavated from the work area were deposited in OOL-07.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OMB-05 are Co-60 and Cs-137. This radioactivity would have been reintroduced to the area after 1998 and, if present, would be a small fraction of the DCGL.
2. Media: Generic Building Materials.
3. Continued Investigation: Continued investigation will evaluate the structure as is.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in OMB-05.
2. Planned: No further decommissioning activities are planned.
3. Anticipated End State Configuration: The structure will remain as is.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OMB-05 is identified as a Class 3 Area.

<u>Seal Pit (OMB-06)</u>

Description: OMB-06 is a YNPS structure located on U S Gen owned property. OMB-06 is bounded by survey area OOL-03 on the east south and west and by survey area OOL-01 on the north. OMB-01 consists of reinforced concrete that forms the discharge structure of the circulating water system. The discharge structure is located at the edge of Sherman Reservoir and is the terminus of the 84 inch diameter concrete pipe returning circulating system water to Sherman Reservoir.

History: The circulating water system is the ultimate discharge point for 10 CFR 20 permitted releases of liquid radioactive effluents. Access to OMB-06 is through OOL-03 a Class 3 land survey area, there is a potential that contamination may have been translocated to OMB-06 from OOL-03.

Draft NUREG/CR-5849 based surveys performed in 1998 identified licensed radioactivity present in scale built up on the circulating water pipes upstream from OMB-06 and in pond sediment samples collected in the bay of Sherman Reservoir (OOL-01) in front of OMB-06. The radioactive material concentrations detected in the pond sediment as well as sediments taken from inside the structure after the circulating water system was deactivated, is below the proposed DCGLs for soil. The circulating water piping will be surveyed as part of the continuing characterization investigations and will either be free released or will be removed and disposed of as low-level radioactive waste. The accumulation of sediments within structure will be removed. It is expected that any residual contamination, if present in OMB-06, would not exceed a small fraction of the appropriate DCGLs.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OMB-06 are Co-60, Cs-137, Sr-90 and H-3 resulting from the permitted release of liquid radioactive effluents into the circulating water system, discharge to Sherman Reservoir.
2. Media: Reinforced concrete, accumulated sediment within the structure.
3. Continued Investigation: Continued investigation will evaluate reinforced concrete and sediments. Potential for migration of radioactivity exists from groundwater movement through the backfill around the outside of the circulating water system piping located below the Turbine Building. This will be investigated by core bore sampling of soils adjacent to and under the circulating water system piping under the Turbine Building.

Decommissioning/Decontamination Activities

1. Performed: None
2. Planned: No decommissioning activities are planned at this time.
3. Anticipated End State Configuration: OMB-06 will be surveyed as is: Reinforced concrete.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the HSA, survey area OMB-06 is identified as a Class 3 Area.

Turbine Building and Portions of Service Building Outside of the RCA

<u>Turbine Building and Offices (TBN-01)</u>

Description: TBN-01 is bounded by OOL-02 on the north, SVC-01 on the east, NOL-06 on the south and OOL-10 and OOL-02 the west. The TBN-01 consists of a steel frame and concrete block lower structure with a steel frame and metal panel upper structure.

History: The systems present and the processes performed in TBN-01 were not intended to involve radioactive materials. There is information that identifies conditions and events where radioactive material was present in the TBN-01.

A portion of the Turbine Building became contaminated in 1967 while a main coolant pump was being refurbished on the turbine deck. At that time the area was decontaminated. The event was incorporated into plans for decommissioning activities and survey plans developed for this area.

The condensate system contained radioactive materials as a result of primary to secondary system leakage that occurred in the steam generators. Contamination from this condition was identified in the condensate piping and components, in the floor drain system and in the soil around and under the floor drains. Additional contaminated concrete surfaces and soil below the concrete floor were identified near turbine support pedestal #4. All of these identified subsurface locations have undergone a successful mitigation process and have been backfilled to grade. The interior of the structure and slab were surveyed under NUREG/CR-5849 criteria after phase 1 decommissioning activities were complete.

The general sub-surface conditions are the subject of continuing investigation.
Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area TBN-01 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Reinforced concrete, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below-grade reinforced concrete and adjacent sub-surface soils

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in TBN-01 include:
   - Removal of secondary systems.
   - Removal of equipment.
   - Removal of sub-floor systems (floor and equipment drains, service water piping).
   - Removal of soil from around the sub-floor systems.
   - Soil excavations backfilled.
2. Planned: Planned decommissioning activities for the TBN-01 include demolition of the entire structure to grade.
3. Anticipated End State Configuration: Reinforced concrete structure (floor slab), sub-floor soils, sub-grade structures.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area TBN-01 is identified as a Class 3 Area.

<u>Non-Rad Service Building (SVC-01)</u>

Description: SVC-01 is bounded by OOL-02 on the north, by SVC-03 on the east, by SVC-02 on the south and TBN-01on the west. The SVC-01 consists of a steel frame and concrete block structure built on reinforced concrete floor slab and foundations.

History: The systems present and the processes performed in SVC-01 did not involve the use of radioactive materials other than radioactive electro-plated and sealed check sources used to test instrumentation operability. SVC-01 is adjacent to the radiation protection (RP) control point and was maintained as a clean area. There is information concerning events involving radioactive material contamination identified present in SVC-01. The contamination consisted of loose contamination, resulting from inadvertent translocation of radioactivity into SVC-01from the RCA at the control point. When these events were identified the radioactive contamination was cleaned-up and the area surveyed. It is anticipated that any residual contamination, if present in SVC-01, would not exceed a small fraction of the appropriate DCGLs.

Contamination

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area SVC-01 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, sub-surface soil
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete structures and adjacent sub-surface soils

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in SVC-01 include:
   - Removal of secondary systems.
   - Removal of equipment.
2. Planned: Planned decommissioning activities for the SVC-01 include demolition of entire structure to elevation 1022'-8".
3. Anticipated End State Configuration: Reinforced concrete structure (floor slab).

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area SVC-01 is identified as a Class 3 Area.

Service Building Addition (SVC-03)

Description: SVC-03 is bounded by OOL-02 on the north, OOL-02 and OMB-04 on the east, SVC-02 on the south, and SVC-02 and SVC-01on the west. SVC-03 consists of a steel frame and concrete block structure. The ground floor corridor that runs north and south located in the southwest corner of the Service Building Addition and the south most room adjacent to it, are not included in SVC-03 but are included in SVC-02.

History: The systems present and the processes performed in SVC-03 did not involve use of radioactive materials. There is information that identifies inadvertent transmigration of plant-related radioactivity into SVC-03. The contamination consisted of loose radioactive material, resulting from inadvertent translocation of radioactivity into SVC-03 from the RCA. When these events were identified the radioactive contamination was cleaned-up with the results of decontamination verified through survey. It is anticipated that any residual contamination, if present in SVC-03, would not exceed a small fraction of the appropriate DCGLs.

A portion of the SVC-03 was built on top of what was a portion of the RCA from the time prior to its construction. This circumstance will be investigated as part of the continuing investigation of subsurface locations; however, it is anticipated that any residual contamination, if present beneath the poured slab, would not exceed a small fraction of the appropriate DCGLs.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area SVC-03 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Reinforced concrete, surface soil, subsurface soil.
3. Continued Investigation: Continued investigation will evaluate below grade reinforced concrete and adjacent sub-surface soils.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in SVC-03 include:
   - Removal of secondary systems.
   - Removal of equipment.
   - Demolition and removal of the non-RCA portion of the structure.
2. Planned: Planned decommissioning activities for the SVC-03 include demolition of walls to grade.
3. Anticipated End State Configuration: The end state configuration of WST-04 is anticipated to include:
   - Surface concrete structures (floor slab)
   - Subsurface concrete structures (foundations)
   - Subsurface soil.

Classification Statement: Based upon the current/best information indicating the radiological conditions and upon conditions and events identified in the operating history and activities performed during decommissioning, survey area SVC-03 is identified as a Class 3 Area.

Open Land Areas Outside of the RCA (OOL)

Sherman Pond Sediment (OOL-01)

Description: OOL-01 consists of the sediment layers in Sherman Pond and is bounded by the continuation of Sherman Pond on the north, OOL-15 on the east, OOL-13 and OOL-03 on the south and OOL-03 on the west. Sherman Pond is owned by US Gen.

History: OOL-01 has received surface run-off from the east end of the RCA via OOL-12 and OOL-13 and also discharge of the east storm drain system. It also received the permitted liquid waste discharge effluent that was released from the site via the circulating water system. A significant amount of sediment sampling was performed over the life of the plant under the Radiological Environmental Monitoring Program (REMP) with no impact being noted. Additional sediment sampling has been performed in OOL-01 following the cessation of power operations. Scoping samples of pond sediment indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-01 are, Cs-137, Sr-90 and H-3. PCBs have also been identified in the sediments of Sherman Pond.
2. Media: Sediment
3. Continued Investigation: Continued investigation will be necessary to support possible PCB sediment removal.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in OOL-01
2. Planned: Future decommissioning activities may include sediment removal for non-radioactive concerns.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-01 is identified as a Class 3 Area.

Yankee Non-Rad Yard Area (OOL-02)

Description: OOL-02 consists of the land area (owned by YAEC), in the yard area within the current industrial area of the YNPS site. Survey Area OOL-02 is bounded by the YAEC/US Gen property line on the north; OOL-12 and OOL-13 on the east, the Warehouse/Service Building/Turbine Building complex, plus OOL-10 and OOL-08 on the south and OOL-06 on the west. Subsurface systems present in OOL-02 include the east storm drain system, security lighting and video conduit runs, sanitary sewer system, fire protection water system and the circulating water system.

History: The west end of Survey Area OOL-02 received surface run-off from OOL-10, a Class 2 survey area. On the east end OOL-02 is located upslope from survey areas OOL-12 and OOL-13 and so was not subject to run-off from the RCA. OOL-02 has been the main travel path for all material, including radioactive material received at or shipped from the YNPS site. Scoping samples of various survey media in OOL-02 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-02 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface and subsurface soil.
3. Continued Investigation: Continued investigation will be necessary to assess subsurface structures and systems.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OOL-02 include:
   - Removal of subsurface system components that traverse OOL-02
   - Installation of the Auxiliary Service Water system.
2. Planned: Future-decommissioning activities may include removal of certain subsurface structures and systems.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey are sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history and as a result of the decommissioning activities performed to date, survey area OOL-02 is identified as a Class 3 Area.

Sherman Reservoir Dam and South Shoreline (OOL-03)

Description: OOL-03 consists of the land area owned by US Gen. Survey Area OOL-03 is bounded by the Deerfield River and Sherman Reservoir on the north, OOL-13 on the east, the YAEC, US Gen property line (OOL-02) on the south and OOL-04 on the west. Subsurface systems present in OOL-02 include the east storm drain system, security lighting and video conduit runs, sanitary sewer system fire protection water system and the circulating water system.

History: Survey Area OOL-03 has received surface run-off from OOL-02 a Class 3 survey area. OOL-03 has been used as a path of travel for radioactive material received at and shipped from the YNPS site. The HSA has identified that there are no contaminating events associated with OOL-03. Scoping samples of various survey media in OOL-03 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-03 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface and subsurface soil.
3. Continued Investigation: Continued investigation will be necessary to assess subsurface structures and systems.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-03.
2. Planned: Future-decommissioning activities may include removal of certain subsurface structures and systems
3. Anticipated End State Configuration: A soil surface configuration suitable for survey. Subsurface structures requiring survey are sufficiently exposed to allow survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-03 is identified as a Class 3 Area.

US Gen Sherman Station (OOL-04)

Description: OOL-04 consists of the land area owned by US Gen. Survey Area OOL-04 is bounded by the Deerfield River on the north, OOL-03 on the east, the YAEC/US Gen property line (OOL-02) on the south and OOL-05 on the west.

History: Survey Area OOL-04 has received surface run-off from the OOL-02 a Class 3 survey area. The groundwater within Survey Area OOL-04 is suspected of containing radioactivity originating from the operations at YNPS. Sherman Spring, located in Survey Area OOL-04, has been determined to contain plant related radioactivity (tritium). Scoping samples of various survey media in OOL-04 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-04 are Co-60, Cs-137, Ag-108m, Sr-90 and H-3.
2. Media: Surface and subsurface soil, surface water and groundwater.
3. Continued Investigation: Continued investigation will be necessary to assess subsurface soil, surface water and groundwater.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-04.
2. Planned: Future-decommissioning activities may include removal of certain soils depending upon the results of the continuing investigation.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey and access to surface water and groundwater.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-04 is identified as a Class 3 Area.

<u>US Gen Deerfield River Frontage Property (OOL-05)</u>

Description: OOL-05 consists of the land area owned by US Gen. Survey Area OOL-05 is bounded by the Deerfield River on the north, OOL-04 on the east, the YAEC, US Gen property line (OOL-06) on the south, and by non-impacted US Gen owned property on the west.

History: Survey Area OOL-05 has received surface run-off from OOL-06 and the west storm drain system of YNPS. The septic waste disposal systems associated with YNPS are located within the bounds of OOL-05. The original septic system leach field was abandoned in place after it became clogged with solids. A radiological assessment of the leach field identified the presence of low levels of Co-60. Scoping samples of various survey media in OOL-06 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-05 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface and subsurface soil, surface water and groundwater.
3. Continued Investigation: Continued investigation will be necessary to assess surface and subsurface soil, surface water and groundwater.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-05.

2. Planned: Future-decommissioning activities may include removal of certain soils depending upon the results of the continuing investigation and actions required to discontinue use of or close the leach fields.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey and access to surface water and groundwater.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-05 is identified as a Class 3 Area.

YNPS Western Access (OOL-06)

Description: OOL-06 consists of the land area owned by YAEC and is bounded by the YAEC/US Gen property line (OOL-05) on the north, OOL-02 and OOL-08 on the east, OOL-08 on the south and non-impacted YAEC owned property on the west. OOL-06 contains within its bounds survey areas OOL-07 and OMB-03. Subsurface systems present in OOL-06 include the west storm drain system, the site electrical supply conduits and the sanitary sewer system associated with OMB-03. Adjacent to OOL-07 is the location of the YNPS trash compactor and salt/sand shed, both of which are temporary structures. The surface area of OOL-06 is indigenous soils and asphalt of the parking lots area and roadways. There are numerous temporary structures present in OOL-06.

History: Survey Area OOL-06 has received surface run-off from the OOL-02 and is the outfall of the west storm drain system. OOL-06 contains the primary access point for the YNPS site and is a travel path for material, including radioactive material received at or shipped from the YNPS site. There is an abandoned leach field as well as an active leach field both associated with the administration building located within the bounds of OOL-06. Scoping samples of various survey media in OOL-06 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-06 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface and subsurface soil, surface water and groundwater.
3. Continued Investigation: Continued investigation will be necessary to assess surface and subsurface soil, surface water and groundwater.

Decommissioning/Decontamination Activities
1. Performed: No decommissioning activities have been performed in survey area OOL-06.
2. Planned: Future decommissioning activities may include removal of certain soils depending upon the results of the continuing investigation.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey and access to surface water and groundwater.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-06 is identified as a Class 3 Area.

Spoils Deposit Area (OOL-07)

Description: OOL-07 consists of the land area owned by YAEC. OOL-07 is bounded entirely by the survey area OOL-06. Survey area OOL-07 consists of a deposit of soils excavated from the area of the ISFSI and ISFSI haul road. The soil deposited in OOL-07 partially covers the footprint of a septic system leach field that serves the Administration Building/Training Center.

History: Survey Area OOL-07 has received excavation spoils from certain YNPS site modifications performed over the history of the YNPS site. Although a majority of the spoils were assessed for radioactive material content prior to deposition in OOL-07 with a "no detectable activity" result, no location specific data has been collected in this Survey Area.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-07 are Co-60, Cs-137, Sr-90 and Ag-108m.
2. Media: Surface and subsurface soil
3. Continued Investigation: Continued investigation may be necessary to assess surface and subsurface soil.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OOL-02 include the addition of soil excavated during construction of the ISFSI haul road.
2. Planned: Future-decommissioning activities may include removal of certain soils depending upon the results of the continuing investigation.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history and as a result of the decommissioning activities performed to date, survey area OOL-07 is identified as a Class 2 Area.

YNPS Site Impacted Perimeter Zone (OOL-08)

Description: OOL-08 consists of the land area owned by YAEC. OOL-08 is bounded by OOL-06, OOL-02, OOL-10, OOL-09, OOL-11, OOL-12, OOL-14 and OOL-15 on the north, and by the non-impacted area on the east, west and south. The surface of OOL-08 is indigenous soils.

History: Survey Area OOL-08 represents that portion of the YNPS site that may have been impacted by wind born transmigration of radioactivity from the YNPS site that is not captured within the bounds of another survey area. OOL-08 forms a wide buffer zone between the plant industrial area and that portion of the site designated as non-impacted. Scoping samples of various survey media in OOL-08 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-08 are Co-60, Cs-137, Sr-90, Ag-108m and H-3.
2. Media: Surface soil.
3. Continued Investigation: Continued investigation may be necessary to assess surface soil.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-08.
2. Planned: Future-decommissioning activities may include removal of certain soils depending upon the results of the continuing investigation.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-08 is identified as a Class 3 Area.

<u>Southeast Construction Fill Area (OOL-09)</u>

Description: OOL-09 consists of the land area owned by YAEC. OOL-09 is bounded on the north, east and south by survey area OOL-08 and on the west by survey area OOL-10.

History: Survey Area OOL-09 has received construction spoils and certain discarded material previously used at YNPS. A comprehensive radiological assessment of this Survey Area has been performed with subsurface objects being located by ground penetrating radar. These objects were exhumed and surveyed for radioactive material, in addition numerous test pits were excavated and assessed. No radioactive material was discovered in the material, soils or groundwater obtained from this area. The area is currently operated as a landfill and ground water is being monitored. Scoping samples of various survey media in OOL-09 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-09 are Co-60, Cs-137, Sr-90, Ag-108m and H-3.
2. Media: Surface and subsurface soil, surface water and groundwater.
3. Continued Investigation: Continued investigation will be necessary to assess surface and subsurface soil, surface water and groundwater.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OOL-09 include the addition of soil excavated during construction of the ISFSI and haul road.
2. Planned: Future decommissioning activities may include removal of certain soils and materials depending upon the results of the continuing investigation and requirements for further cleanup of the area related to non-radioactive materials.

3. Anticipated End State Configuration: A soil surface configuration suitable for survey and access to surface water and groundwater.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-09 is identified as a Class 3 Area.

<u>ISFSI Pad Access Zone (OOL-10)</u>

Description: OOL-10 consists of the land area owned by YAEC. OOL-10 is bounded by OOL-02, NOL-06, NOL-05, NOL-04, NOL-03 and OOL-11 on the north, OOL-08 and OOL-09 on the east, OOL-08 on the south and also on the west.

History: Survey Area OOL-10 is the buffer zone around the RCA and, as such, has the potential to have become contaminated.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-10 are Co-60, Cs-137, Sr-90, Ag-108m and H-3.
2. Media: Surface and subsurface soil, surface water and groundwater.
3. Continued Investigation: Continued investigation will be necessary to assess surface and subsurface soil surface water and groundwater.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OOL-10 consist of soil removal to adjust the grade of the ISFSI fuel transfer haul road.
2. Planned: Future-decommissioning activities may include removal of certain soils and materials depending upon the results of the continuing investigation.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey and access to surface water and groundwater.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-10 is identified as a Class 2 Area.

<u>East RCA Buffer Zone (OOL-11)</u>

Description: OOL-11 consists of the land area owned by YAEC. OOL-11 is bounded by OOL-12 on the north, OOL-08 on the east, OOL-10 on the south and NOL-02 and NOL-03 on the west.

History: Survey Area OOL-11 is the buffer zone around the RCA and, as such, has the potential to have become contaminated.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-11 are Co-60, Cs-137, Sr-90, Ag-108m and H-3.

2. Media: Surface and subsurface soil.
3. Continued Investigation: Continued investigation will be necessary to assess surface soil.

Decommissioning/Decontamination Activities
1. Performed: No decommissioning activities have been performed in OOL-11.
2. Planned: Future-decommissioning activities may include removal of certain soils and materials depending upon the results of the continuing investigation.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-11 is identified as a Class 2 Area.

Warehouse Rail Spur (OOL-12)

Description: OOL-12 consists of the land area owned by YAEC, in the yard area within the current industrial area of the YNPS site extending from the east line of the RCA to the YAEC, US Gen property line. Survey Area OOL-12 is bounded by the Service Building and warehouse, OOL-02 and the YAEC, US Gen property line on the north, OOL-13 and OOL-14 on the east, OOL-08 and OOL-11 on the south and NOL-01 on the west.

History: Survey Area OOL-12 has received surface run-off from the east end of the RCA and has been a travel path for radioactive material received at or shipped from the YNPS site. Contaminated surface soil has been removed from OOL-12 during plant operations.

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-12 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface and subsurface soil.
3. Continued Investigation: Continued investigation will be necessary to assess subsurface structures and systems.

Decommissioning/Decontamination Activities
1. Performed: Decommissioning activities performed in OOL-02 include the installation of the Auxiliary Service Water system.
2. Planned: Future decommissioning activities may include removal of certain surface and subsurface structures and systems
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-12 is identified as a Class 1 Area.

US Gen Rail Spur Terminus (OOL-13)

Description: OOL-13 consists of the land area owned by US Gen. Survey Area OOL-13 is bounded by Sherman Reservoir on the north, OOL-15 on the east, the YAEC survey Area OOL-14 on the south and OOL-12 on the west.

History: Survey Area OOL-13 has received surface run-off from the OOL-12 and has been used as a path of travel for radioactive material received at and shipped from the YNPS site.

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-13 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface and subsurface soil.
3. Continued Investigation: Continued investigation will be necessary to assess surface and subsurface soil.

Decommissioning/Decontamination Activities
1. Performed: No decommissioning activities have been performed in survey area OOL-13.
2. Planned: Future-decommissioning activities may include removal of certain soils.
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-13 is identified as a Class 1 Area.

US Gen Wheeler Brook Frontage (OOL-14)

Description: OOL-14 consists of the land area owned by US Gen. Survey Area OOL-14 is bounded by OOL-13 on the north, OOL-15 and OOL-08 on the east, OOL-08 on the south and also on the west.

History: Survey Area OOL-14 at one time was included within the security fence of the YNPS site OOL-14 serves as a buffer zone between survey areas OOL-12 and OOL-13. Scoping samples of various survey media in OOL-14 indicate mean levels of radioactivity to be a small fraction of the proposed soil DCGLs (see Table 2-5).

Although OOL-14 abuts class 1 area OOL-13, the mode of contamination of OOL-13 was by surface water run-off from the RCA. OOL-14 is above the grade level of OOL-13 and was not impacted by the surface run-off transmigration vector.

Contamination:
1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-14 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface soil.
3. Continued Investigation: Continued investigation will be necessary to assess surface soils.

Decommissioning/Decontamination Activities

1. Performed: Decommissioning activities performed in OOL-02 include placement of LP Gas storage tanks.
2. Planned: Future decommissioning activities include removal the LP Gas tanks from survey area OOL-14
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-14 is identified as a Class 3 Area.

<u>US Gen Sherman Reservoir East Shoreline (OOL-15)</u>

Description: OOL-15 consists of the land area owned by US Gen. Survey Area OOL-15 is bounded by Sherman Reservoir (OOL-01) on the north, US Gen owned non-impacted area on the east, OOL-08 on the south and OOL-14 and OOL-13 on the west.

History: Survey Area OOL-15 serves as a buffer zone to survey area OOL-13.

Although OOL-15 abuts class 1 area OOL-13, the mode of contamination of OOL-13 was by surface water run-off from the RCA. OOL-15 is above the grade level of OOL-13 and beyond the Wheeler Brook surface run-off terminus. OOL-15 was not impacted by the surface run-off transmigration vector that impacted OOL-13.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-15 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface soil.
3. Continued Investigation: Continued investigation will be necessary to assess surface soils.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-15.
2. Planned: No decommissioning activities are anticipated for survey area OOL-15
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-15 is identified as a Class 3 Area.

<u>Yankee House Parking (OOL-16)</u>

Description: OOL-16 consists of the land area owned by YAEC and is entirely bounded by non-impact area.

History: Survey Area OOL-16 received soil from the YNPS site for the purpose of leveling the parking area. Although the soil originated in areas that are impacted (class 3 areas), soils from these areas typically show levels of radioactivity at a small fraction of the proposed soil DCGLs.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-16 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface soil.
3. Continued Investigation: Continued investigation will be necessary to assess surface soils.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-16.
2. Planned: No decommissioning activities are anticipated for survey area OOL-16
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-16 is identified as a Class 3 Area.

<u>Asphalt Brick and Concrete Storage Area (OOL-17)</u>

Description: OOL-17 consists of the land area owned by YAEC. Survey Area OOL-17 is bounded entirely by non-impact area.

History: Survey Area OOL-17 received asphalt and concrete from the YNPS site. Materials deposited in this area were subjected to radiological survey for free release prior to being transported to OOL-17. Based on the origin of this material the area must be classified as impacted. It is anticipated that any residual radioactivity, if present would not exceed a small fraction of the proposed soil DCGLs.

Contamination:

1. Radionuclides Potentially Present: The primary radionuclides of concern for survey area OOL-17 are Co-60, Cs-137, Sr-90 and H-3.
2. Media: Surface soil.
3. Continued Investigation: Continued investigation will be necessary to assess surface soils.

Decommissioning/Decontamination Activities

1. Performed: No decommissioning activities have been performed in survey area OOL-17.
2. Planned: No decommissioning activities are anticipated for survey area OOL-17
3. Anticipated End State Configuration: A soil surface configuration suitable for survey.

Classification Statement: Based upon the current/best information indicating the radiological conditions and on conditions and events identified in the operating history, survey area OOL-17 is identified as a Class 3 Area.

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3 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction and General Considerations

In accordance with 10CFR50.82(a)(9)(ii)(B), Reference 3-1, the License Termination Plan (LTP) must identify the major dismantlement activities that remain. Included in this information are estimates of occupational radiation dose associated with those activities and estimates of projected volumes of radioactive waste. These activities are undertaken pursuant to the current 10CFR50 license, are consistent with the PSDAR (Reference 3-2), and do not depend upon approval of the LTP to proceed.

YAEC intends to release the YNPS site for unrestricted use, and its primary goals are to decommission the YNPS safely and to maintain continued safe storage of spent fuel, until it is removed from the site. YAEC will decontaminate and dismantle YNPS in accordance with the DECON alternative, as described in the NRC's Final Generic Environmental Impact Statement (NUREG-0586 and its supplements, Reference 3-3). Completion of the DECON option is contingent upon continued access to one or more low-level waste disposal sites. Currently YNPS has access to low-level waste disposal sites in Barnwell, South Carolina, and South Clive, Utah.

Decommissioning activities at YNPS are being conducted in accordance with the YNPS PSDAR, YDQAP, FSAR, Technical Specifications, Part 50 license, and the requirements of 10CFR50.82(a)(6) and (a)(7). As such, the conduct of the decommissioning activities described herein is not dependent upon approval of the LTP. In addition, YAEC does not foresee any of the specific decommissioning activities described herein as resulting in the need for prior NRC approval upon evaluation under 10CFR50.59. These activities are being conducted in accordance with existing program and procedures which have been reviewed by the NRC, including: YNPS Radiation Protection Program, Occupational Safety Program, Radioactive and Non-Radioactive Waste Management Programs and the Decommissioning Quality Assurance Plan. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during plant operation and refueling. Nonetheless, if any activity requires prior NRC approval under 10CFR50.59(c)(2) or a change to the YNPS Technical Specifications or license, a submittal will be made to the NRC for review and approval before implementing the activity in question.

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.

- Detailed decommissioning work documents 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. 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.

- After completion of decommissioning and/or remediation activities 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 before remediating substructure and soils beneath the structures. Measures, as described in LTP Section 5.4.5, will be implemented to prevent recontamination of surveyed areas prior to final status survey.

General decontamination and dismantlement considerations are given in Section 3.2; however, specific decontamination and dismantlement considerations for applicable systems, structures, and components are given in the following sections. The contamination status for the remaining systems is provided in Table 3-1. Also, the description and status of remaining SSCs are presented in Sections 3.2.2.1 (Systems and Components) and 3.2.2.2 (Structures).

3.2.2.1 Systems and Components

3.2.2.1.1 Temporary Waste Water Processing System

The Temporary Waste Water Processing System receives, contains, treats, and safely disposes of liquid radioactive wastes. Waste water generated as a result of decommissioning activities is routed to a 20,000 gallon waste water storage tank (TK-81). The tank currently accepts water from the radioactive lab sump discharge line. The waste water is pumped to this storage tank and is then transferred to the evaporator equipment enclosure for processing and eventual discharge. Per the NPDES permit, discharge of construction waste water can also be via the storm drain system.

Most of the major equipment (the 20,000 gallon tank, the equipment enclosure, two 5,000-gallon test tanks, and the package pool boiler) is located in the area east of the Spent Fuel Pit Building, adjacent to Fire Hose House 15.

Decontamination and dismantlement considerations for the Temporary Waste Water Processing Island are as follows:

- The Temporary Waste Water Processing Island should be isolated at the connections to the Plant Ventilation System, Auxiliary Service Water System, and the Rad Lab Sump System.

- Sludge will be removed from the Rad Lab Sump tanks and the 20,000 gallon storage tank prior to dismantlement of the system.

The Temporary Waste Water Processing System will be dismantled and disposed of as radioactive waste.

3.2.2.1.2 Radiation Monitoring System

The Radiation Monitoring System monitors plant radiological conditions through two subsystems:

- Process Radiation Monitoring Subsystem
- Area Radiation Monitoring Subsystem

The following components of the Process Radiation Monitoring Subsystem are required to support plant operation during the dismantlement period:

Auxiliary Service Water/Liquid Waste Effluent Channel: This channel monitors Auxiliary Service Water (ASW) and also monitors liquid effluent from the Temporary Waste Water Processing Island System before it is discharged to Sherman Reservoir. If any of the following conditions occur, the release will be terminated:

- A high or failure alarm from the ASW/liquid effluent radiation monitor

- Loss of power to the ASW/liquid effluent radiation monitor or control circuit.

The Offsite Dose Calculation Manual (ODCM) has provisions for a liquid effluent release with the ASW radiation monitor out of service. Operation of the ASW System for release of liquid effluent will continue, as required by the ODCM, through decommissioning to support dismantlement and decontamination activities.

Primary Vent Stack Channel: This channel monitors airborne releases from ventilated areas of the primary side of the plant before release to the environment. Airborne release monitoring will continue throughout the decommissioning until ventilated areas are sufficiently decontaminated and/or demolished. This monitoring will be conducted until no longer required by the ODCM.

The following Area Radiation Monitoring channels, located in potentially contaminated areas, will be used to monitor conditions during system and component dismantling activities:

- Spent Fuel Pit manipulator crane during component movement activities
- Radiation Control Area Control Point
- Primary Auxiliary Building fan room.

As systems are dismantled, the associated monitoring equipment will also be removed. The Area Radiation Monitoring equipment will remain in operation until contaminated process systems have been removed from the area or are no longer required for demolition activities. It will then be removed prior to the commencement of area and building decontamination activities. Detector locations may be changed to facilitate removal activities if the new location provides comparable detection capability. Radiation Monitoring System (RMS) equipment in uncontaminated areas of the plant will be removed as part of the site dismantlement and restoration process.

Decontamination and dismantlement considerations for the Radiation Monitoring System include removal of the system in uncontaminated areas of the plant, as part of the site dismantlement and restoration process.

3.2.2.1.3 Ventilation Systems (Including VC Ventilation and Purge System)

The Ventilation System includes equipment associated with the collection, monitoring, filtration and discharge of potentially radioactive gaseous effluents from specific plant areas. The Ventilation System provides for the controlled airborne ventilation and discharge function. It is used to ventilate and discharge exhaust air via fixed ductwork from the Vapor Container, Spent Fuel Pit Building, Fuel Transfer Enclosure and Fan Room. The Ventilation System also ventilates and discharges exhaust air via temporary ducting from the Radioactive Waste Evaporator System, and other areas of the plant as needed to support specific decontamination activities. Potentially radioactive airborne effluents are collected by the Ventilation System, filtered through pre-filters and HEPA filters, and discharged through the Primary Vent Stack (PVS). Instrumentation channels monitor the effluent release through the PVS for noble gases, and sample for tritium and particulates. The Ventilations System components and equipment are located in or on the Primary Auxiliary Building, Vapor Container, and Yard Area.

Ventilation System components and equipment are located in or on the Primary Auxiliary Building, the VC, and the Yard Area. Potential airborne releases from these areas shall be processed (filtered) and monitored prior to release, as specified in the Yankee Decommissioning Quality Assurance Program and the ODCM.

The following are decontamination and dismantlement considerations specific to the Ventilation System:

- Fans and motors will be separated from their associated baseplates before removal,

- Filter units will be dismantled into manageable sections.

Heating and ventilation systems at the Gatehouse will remain to support fuel storage and monitoring at the ISFSI.

Contaminated portions of the Ventilation System will be dismantled and disposed of as radioactive waste.

3.2.2.1.4 Fuel Handling Equipment System

The Fuel Handling Equipment System supported the handling of fuel and irradiated components in the SFP. The system consists of the Spent Fuel Pit manipulator crane and yard area crane, fuel inspection equipment, grappling fixtures, fuel storage racks, and the necessary associated controls and instrumentation.

The fuel handling equipment is no longer used to move spent fuel, since all spent fuel has been removed from the SFP. Elements of this system may be used during the demolition of the SFP and SFP Building.

There are currently no decontamination or dismantlement considerations specific to the Fuel Handling Equipment System. The Fuel Handling System will be dismantled and disposed of as radioactive waste.

3.2.2.1.5 SFP Cooling and Purification System

The SFP Cooling and Purification System cooled and purified Spent Fuel Pit water. The spent fuel has been removed from the pool, and there is no longer any need for the cooling function of this system.

The SFP Cooling and Purification System is not required to support dismantlement activities. Resins from the system will be removed and sent to a radioactive waste disposal facility. The remaining components will be dismantled and disposed of as radioactive waste

Decontamination and dismantlement considerations for the SFP Cooling and Purification System include the isolation of the system at the connections to the Demineralized Water and Auxiliary Service Water System.

3.2.2.1.6 Auxiliary Service Water System

The Auxiliary Service Water (ASW) System supports plant operations by supplying water from the Sherman Reservoir to dilute waste water releases.

The system consists of one auxiliary service water pump and the necessary associated valves, piping, fittings and instrumentation. The pump, installed in the Screenwell House, circulated the Sherman Reservoir water through the SFP Cooling System heat exchanger and discharged it back into Sherman Reservoir. An effluent radiation monitor is installed downstream of the heat exchanger.

The ASW System and components are located in the Screenwell House, the SFP Building, and Yard Area. The system is being used to support decontamination and dismantlement activities.

Once the ASW System is no longer required for demolition activities, it will be isolated at the connections to the SFP Cooling System and the Temporary Waste Water Processing Island. Components and piping will be removed and disposed of.

3.2.2.1.7 Demineralized Water System

The Demineralized Water System supports plant operations by providing demineralized water for decontamination activities.

The system consists of a water storage tank, one make-up pump, and the necessary associated valves, piping, fittings, hoses and instrumentation. The system may be used to support decontamination and dismantlement activities. During decontamination and dismantlement, the Demineralized Water System will be isolated at the connections to plant systems, as they are being isolated and dismantled.

3.2.2.1.8 Compressed Air System

The Compressed Air System provides air for plant use. The system consists of portable electric and/or diesel-driven air compressors, receiver tanks, and the necessary associated valves, piping, fittings, and instrumentation.

The Compressed Air System is required to support dismantlement activities. The system and components are located in various areas of the plant. The system will remain in service to support decommissioning activities. Portions of the Compressed Air System will be isolated, dismantled, and removed as the systems and areas that it supports are dismantled and removed from service.

3.2.2.1.9 Electrical System

The onsite electrical system is powered by the 13.8kV Massachusetts Electric Line. The system consists of transformers, switchboards, motor control centers, distribution panels, and associated instrumentation and controls.

The 13.8 kV Massachusetts Electric Line provides power to the Furlon House, the Training Center, and the Trash Compactor. The electrical system at YNPS provides power to equipment that will remain energized during the final phase of decommissioning.

All onsite electrical equipment is powered from two 480VAC switchboards via two 13.8kV/480V, 100kVA transformers. The 1600 amp Secondary Side Switchboard power equipment is located on the Secondary side of the plant. The Gatehouse and the ISFSI are supplied from the Secondary Side Switchboard. The 1200 amp Primary Side Switchboard powers equipment on the Primary side of the plant. The Spent Fuel Pit motor control center (MCC) and the Fuel Transfer Enclosure Switchboard are powered from the Primary Side Switchboard through a manual transfer switch located near the Primary Side Switchboard. Backup power for portions of the plant electrical system is provided by a manually-started and loaded 175 kW Security Diesel Generator.

Electrical System components associated with the Gatehouse and ISFSI will remain to support storage and monitoring of spent fuel at the ISFSI. There are currently no decommissioning or dismantlement considerations specific to the Electrical System.

3.2.2.1.10 Heating System

The Heating System consists of permanent and temporary electric heater units. The Heating System may be used during the dismantlement period. The system and its components are located in various plant buildings. The system will remain operable to support environmental heating requirements during contaminated system removal activities. Temporary heating may be required during area and building dismantlement activities. Heating System components associated with the Gatehouse will remain to support storage and monitoring of spent fuel at the ISFSI.

There are currently no decommissioning or dismantlement considerations specific to the Heating System.

3.2.2.1.11 Fire Protection and Detection System

The Fire Protection and Detection System provides the equipment needed to detect and to respond to fires that could occur in the plant. The system consists of electric and diesel-driven fire pumps, a pressure maintenance system, hydrants, hoses, detectors, the necessary associated valves, piping, fittings and instrumentation.

Portions of the Fire Protection and Detection System are required to support plant operations during the dismantlement period. The Fire Protection Technical Requirements Manual presents system availability requirements. The Fire Protection Technical Requirements Manual describes the locations of the Fire Protection System components. Portions of the system, that are no longer required to support fire suppression requirements, may be disconnected, isolated and removed.

Modifications to the Fire Protection and Detection System require review and modifications, as necessary, to the YNPS Fire Protection Plan.

3.2.2.2 Structures

3.2.2.2.1 Yard Area Crane and Support Structure

The Yard Area Crane Support Structure is a braced steel frame structure that supports a crane that services the Ion Exchange (IX) Pit, SFP, and Decontamination Room. The crane support structure is approximately 34 feet by 151 feet by 73 feet high, with a design capacity of 80 tons.

The Yard Area Crane and Support Structure may be used to support activities associated with the demolition of the SFP and IX Pit and other heavy lifts. The support columns will be removed to the top of the concrete foundations.

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.

- The 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
- Secondary Shielding
- Auxiliary Shielding.

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

- Auxiliary shielding will be decontaminated and dismantled as part of the area and building decontamination and dismantlement activities.

- Supplemental shielding may be decontaminated and dismantled at any time.

Radiation Shielding will be demolished or dismantled and disposed of as radioactive waste.

3.2.2.2.6 Pipe Chases

There are two pipe chases between the Primary Auxiliary Building and the VC:

Lower Pipe Chase: The Lower Pipe Chase is a corridor that runs between the second story of the PAB and the VC lower hemisphere. The chase is constructed of reinforced concrete.

Upper Pipe Chase: The Upper Pipe Chase is a corridor that runs form the PAB roof to the VC lower hemisphere. The chase is constructed of concrete masonry units and is supported by the lower pipe chase.

The piping in both Pipe Chases has been removed and the VC cut to allow for easier removal of equipment and components from the VC and to serve as an alternate personnel access to the VC. The pipe chases will be removed and disposed of as radioactive waste. The associated support columns will be removed to the level of the PAB ground floor level (elevation 1022’-8”).

There are currently no decontamination or dismantlement considerations specific to the Pipe Chases.

3.2.2.2.7 Fuel Transfer Chute

The Fuel Transfer Chute was used to transfer new and spent fuel, as well as irradiated components, between the SFP and the VC. The chute was a series of stainless steel pipe sections connected by bolted flanges enclosed in a reinforced concrete tunnel. The chute is structurally isolated from the VC by a metal bellows expansion joint. The Fuel Transfer Chute was accessed through a below-grade manhole tank.

The Fuel Transfer Chute has been isolated by:

- Re-supporting the Fuel Transfer Chute/SFP penetration assembly to the SFP using the latch mechanism,

- 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 liner 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 minimizes neutron activation, thereby generating minimal radioactive waste. This waste should qualify for disposal at a low-level radioactive waste disposal site.

As indicated in Section 1 of the LTP, YAEC may decide to remove some portions of the site from the license before license termination. For those areas the process outlined in Section 1.5 will be followed. Termination of the license will occur after the last stage of final status survey and independent NRC verification (i.e., on the grounds and SSCs associated with the ISFSI).

3.3 Decommissioning Schedule

Figure 3-1 provides an overview of the decommissioning schedule. Updates will be provided to the NRC through current interactions with the NRC Region I personnel.

3.4 Radiological Impacts of Decommissioning

The decommissioning activities are being conducted under the provisions of the YNPS Radiation Protection Program and Radioactive Waste Management Program. These programs continue to be implemented as described in the YNPS FSAR. The Radiation Protection Program implements the regulatory requirements of 10CFR20 through approved plant procedures established to maintain radiation exposures ALARA. The Radioactive Waste Management Program controls generation, characterization, processing, handling, shipping and disposal of radioactive wastes per the approved YNPS Radiation Protection Program, Process Control Program, and plant procedures.

The current Radiation Protection Program (described in FSAR Section 507), Waste Management Program (FSAR Section 508) and Offsite Dose Calculation Manual will be used to protect workers and the public during the various decontamination and decommissioning activities. These well-established programs are routinely inspected by the NRC to ensure that workers, the public, and the environment are protected during facility decommissioning activities. It is also important to note that most decommissioning activities involve very similar radiation protection and waste management considerations as those encountered during plant operations. As described in the PSDAR, the YNPS decommissioning will be accomplished with no significant adverse environmental impacts in that:

- The postulated impacts associated with the method chosen, DECON, have already been considered in the Final Generic Environmental Impact Statement (FGEIS).

- There are no unique aspects of the plant or decommissioning techniques to be utilized that would invalidate the conclusions reached in the FGEIS.

- The methods to be employed to dismantle and decontaminate the site are standard construction based techniques fully considered in the FGEIS.

- The site-specific person-rem estimate for all decommissioning activities has been conservatively calculated using methods similar to and consistent with those in the FGEIS.

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.

Table 3-1

Remaining Contaminated Plant Systems

(as of July 2003)

System	Internally Contaminated?	Externally Contaminated?	Extent of Contamination
Radiation Monitoring System	No	Yes	Entire System
VC Ventilation and Purge System	Yes	Yes	Entire System
Fuel Handling Equipment System	Yes	Yes	Entire System
SFP Cooling and Purification System	Yes	Yes	Entire System
Auxiliary Service Water System	No	Yes	Partial System
Demineralized Water System	No	Yes	Partial System
Compressed Air System	No	Yes	Partial System
Electrical System	No	Yes	Partial System
Heating System	Yes	Yes	Partial System
Ventilation System	Yes	Yes	Entire System
Fire Protection and Detection System	No	Yes	Partial System

Table 3-2

Status of Plant SSCs as of July 2003

SSC	Status
Reactor Vessel	Removed.*
Steam Generators	Removed.
Main Coolant System	Removed.
Pressure Control and Relief System	Removed.
Charging and Volume Control System	Removed.
Chemical Shutdown System	Removed.
Purification System	Removed.
Component Cooling System	Removed.
Primary Plant Corrosion Control System	Removed.
Primary Plant Sample System	Removed.
Waste Disposal System	Original system removed, replaced with temporary liquid waste system.
Shutdown Cooling System	Removed.
Primary Plant Vent and Drain System	Removed.
Emergency Core Cooling System	Removed.
Radiation Monitoring System	Partially removed, portions in service.
VC Ventilation and Purge System	Partially removed, portions in service.
VC Heating and Cooling System	Removed.
Post-Accident Hydrogen Control System	Removed.
Containment Isolation System	Removed.
Fuel Handling Equipment System	Partially removed, portions in service
SFP Cooling and Purification System	Modified for SFP Island
Main Steam System	Removed.
Feedwater System	Removed.
Steam Generator Blowdown System	Removed.
Emergency Feedwater System	Removed.
Service Water System	Partially Removed, ASW installed for SFP Island.
Demineralized Water System	Partially removed, portions in service.
Compressed Air System	Original system removed, temporary system provided for SFP.
Electrical System	Partially removed, portions in service.

* "Removed" SSCs have been physically removed from the site and disposed of in appropriate disposal facilities.

Table 3-2

Status of Plant SSCs as of July 2003

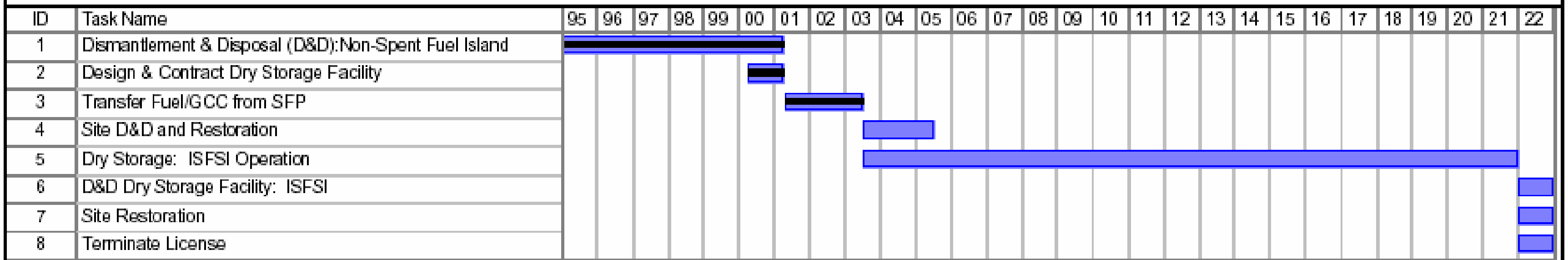
SSC	Status
Heating System	Partially removed.
Ventilation System	Partially removed, portions in service.
Fire Protection and Detection System	Partially removed, portions in service
Primary Pump Seal Water System	Removed.
Safe Shutdown System	Removed.
Water Cleanup System	Removed.
Vapor Container	Placed in lay-up condition.
Reactor Support	Placed in lay-up condition.
Vapor Container Polar Crane	Placed in lay-up condition.
Radiation Shielding	Partially removed/decontaminated.
Neutron Shield Tank	Removed.
Pipe Chases	Placed in lay-up condition.
Fuel Transfer Chute	Partially removed/decontaminated
Yard Area Crane and Support Structure	In service.
Ion Exchange Pit	Partial decontamination in 1997, full decon after fuel removed from SFP. North wall required structurally for SFP.
Primary Vent Stack	In service.
Spent Fuel Pit and Spent Fuel Pit Building	In service.
New Fuel Vault	To be decontaminated after fuel removed from SFP. West wall required structurally for SFP.
Primary Auxiliary Building	Partially decontaminated.
Diesel Generator Building	Building demolished.
Waste Disposal Building	Partially decontaminated.
Safe Shutdown System Building	Partially decontaminated.
Potentially Contaminated Area (PCA) Storage Buildings 1 and 2 and Warehouse	PCA Bldgs No. 1 and 2 and Warehouse to be decontaminated.
Compactor Building	To be decontaminated.
Service Building	Portions of building within the Radiation Control Area are in service.
Miscellaneous Tanks	Most removed; one tank remaining to be decontaminated.
Meteorological Tower	Function no longer required. Tower at ISFSI provides necessary wind speed and direction information.

Table 3-3

Radiation Exposure Projections

Activity	Exposure (Person-rem)	
	Original Estimate, Reference 3-5	**Revised Estimate, Reference 3-6**
Component Removal Project		
• Asbestos Abatement	73	76
• Steam Generators and Pressurizer	62	59
• Reactor Vessel Internals	25	92
Subtotal	**160**	**227**
Fuel Transfer	41	41
Dismantlement		
• Reactor Vessel	48	33
• Main Coolant System	50	36
• Other Systems in Vapor Container	84	48
• Balance of Plant Systems	98	48
• Asbestos Abatement	90	55
• Structures	50	28
• Miscellaneous	82	56
Subtotal	**502**	**304**
Transportation	41	7
Plant Effluents	<1	<1
Total	**744**	**579**

Figure 3-1
YNPS Decommissioning
Summary Schedule



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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.

4.2.3 Surface Water and Groundwater

Characterization data available to date indicated that no remediation of surface or ground waters will be required at YNPS to meet the site release criteria.

4.3 ALARA Evaluations

When dismantlement and decontamination actions are completed, residual radioactivity may remain on building surfaces and on site soils. Residual radioactivity must satisfy the provisions of 10CFR20, Subpart E. As depicted on Figure 4-1, the ALARA cleanup levels for the YNPS decommissioning may be established at one of two levels:

(1) a pre-defined generic ALARA screening, or
(2) a survey unit-specific ALARA evaluation.

In either case, the ALARA evaluation uses an action level. This action level corresponds to a residual radioactivity concentration at which the averted collective radiation dose converted into dollars is equal to the costs of clean-up (e.g., risk of transportation accidents converted into dollars, worker and public doses associated with the action converted into dollars, and the actual costs to perform the activity).

If the dollar-value of further dose reduction from additional clean-up is greater than the "costs" of the action, then the action being evaluated is cost-effective and should be performed. Conversely, if the dollar-value of dose reduction associated with further clean-up is less than the costs of that action, the current level of residual radioactivity is already considered to be ALARA, and further action would not be required. The methodology and equations used for calculating remediation levels are consistent with those provided in Appendix N of NUREG-1757, Volume 2. (Reference 4-3). These are provided in Appendix 4.A of the LTP. Documentation of ALARA evaluations will be included in the final status survey report for each survey area.

4.3.1 Generic ALARA Screening Levels

As discussed in Appendix N to NUREG-1757, Volume 2, clean-up of soils beyond the DCGLs is not likely to be cost-beneficial due to the high costs of waste disposal. A generic ALARA evaluation for soils will be developed to determine if this is the case for YNPS. If clean-up of soils beyond the DCGL is determined not to be cost-beneficial, soils meeting the site-specific DCGLs, determined in Section 6, will be considered to be at levels that are "as low as reasonably achievable."

For structures, a generic ALARA screening level will be calculated using conservative estimates for building clean-up costs. This generic ALARA screening level will be calculated using the guidance of Appendix N to NUREG-1757, Volume 2, and documented. This value will 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

4.4 References

4-1 Title 10 to the Code of Federal Regulations, Part 50.82. "Termination of licenses."

4-2 Title 10 to the Code of Federal Regulations, Subpart E to Part 20, "Radiological Criteria for License Termination."

4-3 NUREG-1757, Vol. 2 "Consolidated NMSS Decommissioning Guidance," dated September 2003.

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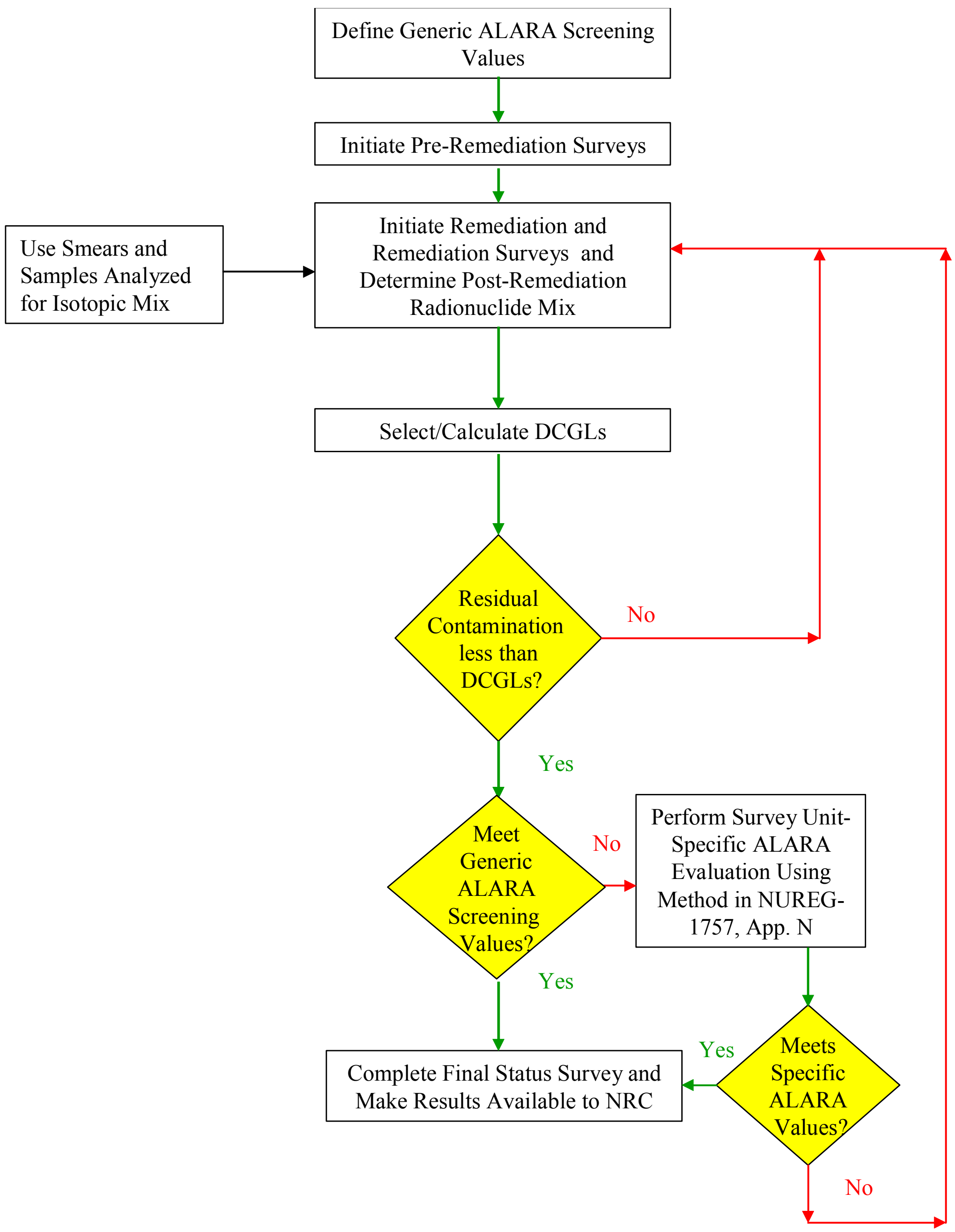
Figure 4-1
Survey Unit ALARA Evaluation Process



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Appendix 4A
ALARA Evaluations

4A.1 Determining ALARA Action Levels

Per Appendix N of NUREG-1757, the residual radioactivity level that is as low as reasonably achievable, or ALARA, is the concentration, Conc, at which the benefit from additional clean-up exceeds the cost of that clean-up. If the total clean-up cost, $Cost_T$, is set equal to the present worth of the collective doses averted, the ratio of the concentration (Conc) to the $DCGL_W$ is as follows:

$$AL = \frac{Conc}{DCGL_w} = \frac{Cost_T}{\$2000 \times PD \times 0.025 \times F \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}} \quad \text{(Equation 4A-1)}$$

Where

AL	=	ALARA action level, as a fraction of $DCGL_W$
Conc	=	Average concentration of residual activity in the area being evaluated
DCGLw	=	Derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 25 mrem/yr to the average member of the critical group
$Cost_T$	=	Total cost of clean-up action, in dollars
\$2000	=	Monetary value of one person-rem averted (NUREG-1757, Appendix N, Table N.2)
PD	=	Population density for critical group scenario, people/m^2
0.025	=	Annual dose to average member of critical group from residual radioactivity at DGCLW concentration, rem/yr
A	=	Area being evaluated, m^2
F	=	Removable fraction for clean-up action being evaluated
r	=	Monetary discount rate, yr^{-1}
λ	=	Radiological decay constant, yr^{-1}
N	=	Number of years over which the collective dose is calculated

Acceptable values for population density (PD), monetary discount rate (r), and the number of years over which the collective dose is calculated (N) are given in NUREG -1757, Appendix N, Table N.2 and are provided in Table 4A-1

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,

$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose}$ (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:

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

Where

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

The cost of workplace accidents, $Cost_{ACC}$, is calculated using the following equation:

$$Cost_{ACC} = \$3{,}000{,}000 \times F_W \times T_A$$ (Equation 4A-4)

Where

\$3,000,000	=	Monetary value of a fatality equivalent to \$2000 per person-rem (NUREG-1757, Appendix N)
F_W	=	Workplace fatality rate, in fatalities per hour worked, or 4.2×10^{-8}/hr (NUREG -1757, Appendix N)
T_A	=	Worker time required for clean-up, person-hours

The cost of traffic fatalities incurred during the shipment of waste, $Cost_{TF}$, is calculated using the following equation:

$$Cost_{TF} = \frac{\$3{,}000{,}000 \times V_A \times F_T \times D_T}{V_{ship}}$$ (Equation 4A-5)

Where

\$3,000,000	=	Monetary value of a fatality equivalent to \$2000 per person-rem (NUREG-1757, Appendix n)
V_A	=	Volume of waste produced, m^3
F_T	=	Fatality rate per truck-kilometer traveled, in unites of fatalities per truck-kilometer or 3.8×10^{-8}/km (NUREG -1757, Appendix N, Table N.2)
D_T	=	Distance traveled, km
V_{ship}	=	Volume of waste shipped per truckload, or 13.6 m^3 from NUREG-1757, Appendix N, Table N.2

The cost of clean-up worker dose, $Cost_{WDose}$, is calculated using the following equation:

$$Cost_{WDose} = \$2000 \times D_R \times T_R$$ (Equation 4A-6)

Where

\$2,000	=	Dollars per person-rem from Appendix N, Table N.2)
D_R	=	Total effective dose equivalent to workers, rem/hr
T_R	=	Time worked to remediate area, person-hours

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

4A.3 Example

The following example is one provided in NUREG -1757. The values for the cost of the clean-up activity and the clean-up action effectiveness are those presented in NUREG-1757. At the time that an actual ALARA evaluation is performed, site-specific costs and clean-up action effectiveness will be used.

The following example considers a building with residual Cs-137 radioactivity (λ=0.023/yr). The clean-up activity being evaluated is washing a floor of 100 m^2 area. The estimated total cost is \$400 and may remove 20% (F = 0.2) of the residual radioactivity. Using these values in Equation 4A-1 gives:

$$\mathrm{AL} = \frac{\$400}{\$2000 \times 0.09 \times 0.025 \times 0.2 \times 100\,\mathrm{sq\ m}} \times \frac{0.07 + 0.023}{1 - e^{-(0.07+0.023)\times 70}}$$

AL = 0.41

Thus, the determination to perform the additional clean-up would be based on an AL of 0.41. If the residual radioactivity on the building floor is demonstrated to be less than 0.41 DCGL, then washing would not be necessary.

5 FINAL STATUS SURVEY PLAN

5.1 Introduction

The FSS (FSS) Plan describes the methods for planning, designing, conducting, and evaluating FSS 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 FSS 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 FSS 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.

5.3 Summary of FSS Process

The FSS provides data to demonstrate that radiological parameters satisfy the established guideline values and conditions. The primary objectives of the FSS are to:

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

The FSS process consists of four principal elements:

- planning,
- design,
- implementation, and
- assessment.

Survey planning includes review of the Historical Site Assessment (HSA) and other pertinent characterization information to establish the radionuclides of concern and the survey unit classifications. Survey units are fundamental elements for which FSSs are designed and executed. The classification of a survey unit determines how large it can be in terms of surface area. If any of the radionuclides of concern are present in background, planning may include establishing appropriate reference areas to be used to establish baseline concentrations and their variability for these radionuclides. A reference coordinate system is used for documenting locations where measurements were made and to allow replication of survey efforts if necessary.

Before the survey process can proceed to the design phase, concentration levels that represent the maximum annual dose criterion of 10CFR20.1402 must be established. These concentrations are established for either surface contamination or volumetric contamination. They are used in the survey design process to establish the minimum sensitivities required for the available survey instruments and techniques, and in some cases, the spacing of fixed measurements or samples to be made within a survey unit. Surface or volumetric concentrations corresponding to the maximum annual dose criterion are referred to as Derived Concentration Guideline Levels, or DCGLs. A DCGL for the average residual radioactivity in a survey unit is called a $DCGL_W$. Values of the $DCGL_W$ may then be increased through the use of area factors to obtain a DCGL that represents the same dose to an individual for residual radioactivity over a smaller area within a survey unit. This scaled value is called the $DCGL_{EMC}$, where EMC stands for elevated measurement comparison.

After the $DCGL_W$ is established, a survey design is developed that selects the appropriate survey instruments and techniques to provide adequate coverage of the unit through a combination of scans, fixed measurements, and sampling. This process ensures that data of sufficient quantity and quality are obtained to make decisions regarding the suitability of the survey design assumptions and whether the unit meets the release criterion. Approved site procedures will direct this process to ensure consistent implementation and adherence to applicable requirements.

Survey implementation is the execution of the survey plan for a given survey unit. This process may consist of a combination of scan measurements, fixed measurements, and/or collection and analysis of samples.

The Data Quality Assessment (DQA) approach is applied to FSS results to ensure their validity and to demonstrate that the objectives of the FSS are met. Data assessment includes data verification and validation (V&V), review of survey design bases, and data analysis. For a given survey unit, the survey data are evaluated to determine if the residual activity levels in the unit meet the applicable release criterion and if any areas of elevated activity exist. In some cases, data evaluation will simply serve to show that all of the measurements made in a given survey unit were below the applicable $DCGL_W$. In that case, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where there are measurement results both above and below the $DCGL_W$. In these cases, statistical tests must be performed to determine whether the unit meets the release criterion. The statistical tests that might be required to make decisions regarding the residual activity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

MARSSIM specifies two non-parametric statistical tests to be applied to FSS data to evaluate whether a set of measurements demonstrates compliance with the release criterion for a given survey unit. These statistical tests are discussed in detail in Section 5.7.

Quality assurance and control measures, satisfying the criteria of Appendix B to 10CFR50 as applicable, are employed throughout the FSS process to ensure that decisions are made on the basis of data of acceptable quality. Quality assurance and control measures are applied to ensure:

- the plan is correctly implemented as prescribed,
- Data Quality Objectives are properly defined and derived,
- data and samples are collected by individuals with the proper training following approved procedures,
- instruments are properly calibrated,
- collected data are validated, recorded, and stored in accordance with approved procedures,
- required documents are properly maintained, and,
- if necessary, corrective actions are prescribed, implemented and assessed.

These measures apply to any services provided in support of FSS.

Survey results will be converted to appropriate units (i.e., either dpm/100 cm^2 or pCi/g) and compared to investigation levels to determine appropriate follow-up action. Measurements exceeding investigation levels will be verified and investigated and, following confirmatory measurement(s), the affected area may be remediated and/or re-classified and a re-survey performed consistent with the guidance in MARSSIM (Section 8.5.3, "If the Survey Unit Fails") and commensurate with the classification and extent of contamination.

It is anticipated that FSS results will be documented and made available to the NRC for survey areas rather than for individual survey units. Reports will be compiled after FSS activities for all of the survey units for a given area are completed. The information to be contained in the FSS report is specified in Section 5.8 of the LTP. This approach should minimize the submittal of redundant historical assessment information and provide for a logical approach to perform reviews and independent verification.

5.4 FSS Planning

5.4.1 Data Quality Objectives

The Data Quality Objective (DQO) process is incorporated as an integral component of the data life cycle at YNPS. The DQO process is used in the planning phase for scoping, characterization, remediation, and FSS plan development using a graded approach. Survey plans that are complex or that have a higher level of risk associated with an incorrect decision (such as FSSs) require significantly more effort than a survey plan used to obtain data relative to the extent and variability of a contaminant. This process, described in MARSSIM, is a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. Data Quality Objectives allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions. Furthermore, the DQO process is flexible in that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. Finally, the DQO process is iterative allowing the survey planning team to incorporate new knowledge and modify the output of previous steps to act as input to subsequent step. A FSS Quality Assurance Project Plan will be developed that provides a detailed description of the application of the DQO process to the different elements of the FSS.

The DQO process consists of performing the following seven steps:

- State the Problem
- Identify the Decision
- Identify the Inputs to the Decision
- Define the Boundaries of the Decision
- Develop a Decision Rule
- Specify Tolerable Limits on Decision Errors
- Optimize the Design for Obtaining Data

The actions taken to address these DQO process steps during the planning of a FSS for a particular survey area are addressed below.

- **State the Problem**

  The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated and the estimated resources. The problem associated with an FSS is to determine whether an area meets the radiological release criterion of 10CFR20.1402.

- **Identify the Decision**

  This step of the DQO process consists of developing a decision statement based on a principal study question (i.e., the stated problem) and determining alternative actions that may be taken based on the answer to the principal study question. Alternative actions identify the potential measures to resolve the problem. The decision statement combines the principal study question and the alternative actions into an expression of choice among multiple actions. For FSS the principal study question could be, "Does residual radioactive contamination present in the survey unit exceed the release criteria?" The alternative actions may include no action, investigation, re-survey, remediation and re-classification.

- **Identify Inputs to the Decision**

  The information required depends on the type of media under consideration and whether existing data are sufficient or new data are needed to make the decision. If the decision can be based on existing data, then the data source(s) will be documented and evaluated to ensure reasonable confidence that the data are acceptable. If new data are needed, then the type of measurement (e.g., scan, direct measurement and sampling) will need to be determined.

  Sampling methods, sample quantity, sample matrix, type(s) of analyses and analytic and measurement process performance criteria, including detection limits, are established to ensure sensitivity is appropriate for the action level and to minimize bias. Action levels provide the criteria for choosing among alternative actions (e.g., whether to take no action or perform confirmatory sampling). These action levels may be radioactivity concentration (pCi/g) or measurement device response (count rate corrected for background). Typical investigation levels for FSS are derived from Table 5-2, depending upon the final classification of the survey unit. FSS will include survey unit specific action levels and their bases.

- **Define the Boundaries of the Study**

  This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame 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 are 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.

  The probability that a survey unit does not meet the release criteria may be graphed and 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 FSS 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. Because 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 such modifications in order to determine whether prior notification to the NRC is required. RAI#5
Survey areas have been determined as described in Section 2.1.1 of this LTP.

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 FSS. 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 FSS 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, impacted areas of YNPS 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) Land areas	 ≤ 100 m^2 ≤ 2,000 m^2
Class 2: Structures (floor area) Land areas	 100 m^2 < area ≤ 1,000 m^2 2,000 m^2 < area ≤ 10,000 m^2
Class 3: Structures (floor area) Land areas	 no limit 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 FSS 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 FSS plan 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 to prepare an FSS 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 FSS. 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 FSS. 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 an FSS 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. This is because the average background 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 FSS. 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 FSS has not been precisely determined due to certain ongoing decommissioning activities at the site. This source of gamma radiation will be evaluated and appropriately accounted in the design of the FSS in adjacent areas. However, some land area surrounding the ISFSI will exhibit a gamma radiation field that will be above the criteria for performing an FSS while the fuel is stored onsite. This portion of the site will not be released or surveyed until the fuel is removed from the site.

5.4.5 Area Preparation: Isolation and Control

5.4.5.1 Area Preparation

Before FSS activities can begin in an area, a transition must occur where planned decommissioning activities are completed and the area is subsequently assessed to scope the required isolation and control measures. This includes establishing if the area is ready for final survey activities and identifying any work practice issues that must be addressed in survey planning and design. Determination of readiness for FSS will be based on characterization and/or remediation surveys indicating that the residual radioactive material is likely to comply with the DCGLs and the removable contamination is below 1000 dpm/100 cm^2 (beta-gamma). Following this assessment, isolation and control measures will be implemented to prevent the introduction of plant-related contamination to soils or structures in the area, prior to, during and after final survey activities. These control measures will include posting (e.g., with a placard or sign) areas that have been turned over for FSS. Isolation and control measures are implemented for areas such as an entire building or large, open areas, for which there should not be any impact from on-going decommissioning activities. In the event that additional remediation is required in an area following the implementation of isolation and control measures, local contamination control measures such as tents, HEPA filters, or vacuums will be employed as appropriate.

Prior to transitioning an area from decommissioning activities to isolation and control, a walkdown may be performed to identify access requirements and to specify the required isolation and control measures. The physical condition of the area will also be assessed, with any conditions that could interfere with final survey activities identified and addressed. If any support equipment needed for final survey activities, such as ladders or scaffolding, are in place, it will be evaluated to ensure that it does not pose the potential for introducing radioactive material into the area. Industrial safety and work practice issues, such as access to high areas or confined spaces, will also be identified during the pre-survey evaluation. Operational health physics or decontamination support data, if available, will be reviewed to identify any potential areas where additional decontamination may be required prior to commencing final survey activities. In some instances, turnover surveys may be performed to verify that an area is ready for final survey.

The following criteria must be met for an area to be deemed ready for isolation and control:

- planned decommissioning activities, in support of license termination, in the area are complete;

- planned decommissioning activities, in support of license termination, in areas either adjacent to the area to be isolated or that could otherwise affect it, are either complete or are deemed not to have any reasonable potential to spread plant-related radioactive material to the area;

- tools and equipment, which are not needed for final survey activities and could interfere with final survey activities, are removed;

- equipment to be used for final survey activities is evaluated to ensure it does not pose the potential for introducing plant-related radioactive material into the area; and

- where practical, transit paths to or through the area, except those required to support final survey activities, are eliminated or re-routed.

Once the area meets the isolation and control criteria, isolation and control will be achieved through:

- a combination of personnel training, physical barriers and postings, and site notices as appropriate, to prevent unauthorized access to an isolated area;

- implementation of provisions to prevent the introduction of plant-related radioactive material by persons authorized to enter the area; and

- measures to prevent the introduction of plant-related radioactive material through the air or through other paths, such as systems or piping.

For buildings, measures to prevent against the introduction of plant-related radioactive material by persons entering an isolated area may include personnel frisking stations at the entry point, the use of "sticky pads," or other such routine methods. Isolation from airborne material may include sealing off openings, including doors and ventilation ducts. Though not likely to be encountered, if a potential for waterborne material is deemed to exist (e.g., floor drains or penetrations left by decommissioning activities), similar measures will be taken to be sure such sources are sealed off from the isolated area. In addition to these physical controls, access points to buildings will be posted with signs that include information pertaining to the proper individual to contact prior to conducting plant-related activities in the area. An administrative process will be used to evaluate, approve (or deny), and document plant related activities conducted in these open land areas during and following FSS.

For open land areas, access roads and trails will be posted with signs that include information pertaining to the proper individual to contact prior to conducting plant-related activities in the area. An administrative process will be used to evaluate, approve (or deny), and document plant related activities conducted in these open land areas during and following FSS. For land areas that do not have positive access control (i.e., areas that have passed FSS but are not surrounded by a fence), the area will be inspected annually and any material that has been deposited since the last inspection will be investigated (i.e., scanned and/or sampled).

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 FSS.

To provide additional assurance that land areas and structures that have undergone successful 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 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. RAI#10 This investigation survey will be more 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 survey areas. These periodic surveys, and any follow-up actions, will continue until the FSS activities for all available survey areas have been successfully completed.

5.4.6 Selection of DCGLs

Residual levels of radioactive material that correspond to allowable radiation dose standards are calculated by analysis of various pathways (direct radiation, inhalation, ingestion, etc.), media (concrete and soils) and scenarios through which exposures could occur. These derived levels, known as derived concentration guideline levels (DCGLs), are presented in terms of surface or mass activity concentrations. DCGLs usually refer to average levels of radiation or radioactivity above appropriate background levels. DCGLs applicable to building or other structural surfaces are expressed in units of activity per surface area (typically dpm/100 cm^2). When applied to soil, sediments or structural materials where the radionuclides are distributed throughout, DCGLs are expressed in units of activity per unit of mass (typically pCi/g).

Section 6 of this plan describes in detail the modeling performed to develop the radionuclide-specific DCGLs for soil, building surfaces and volumetrically-contaminated concrete. These values will be used to establish DCGLs for survey units in cases where measurements are made that are not radionuclide specific or when difficult-to-measure radionuclides are present that necessitate the need for a surrogate radionuclide. In such cases, DCGLs will be established based on a representative radionuclide mix established for each survey unit. In cases where measurable activity still exists, it is expected that the radionuclide mix will be established based on gamma-ray spectroscopy and alpha spectroscopy (where conditions warrant) or equivalent analyses on representative samples, with scaling factors used to establish the activity contribution for any difficult-to-measure radionuclides that might be present. Scaling factors will be selected from available composite waste stream analyses or similar assays. Such analyses are performed periodically and documented in support of waste characterization needs.

In the case of a survey unit for which there is not measurable activity distinguishable from background at the time of FSS design, a representative radionuclide mix (e.g., relative concentration of radionuclides) will be selected based upon historical characterization information from that survey unit or from a unit with a similar history and physical characteristics (e.g., adjacent areas). This representative mix may be used to determine a gross activity DCGL or surrogate ratio DCGL, and to determine the MDC and the number of sample points. Alternatively, a conservative DCGL could be selected as the basis for FSS activities.

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 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.

5.5 Final Status Survey Design

The general approach prescribed by MARSSIM for FSSs requires that at least some minimum number of measurements or samples be taken within a survey unit, so that the non-parametric statistical tests used for data assessment can be applied with adequate confidence. Decisions regarding whether a given survey unit meets the applicable release criterion are made based on the results of these tests. Scanning measurements are used to check the design basis for the survey by evaluating if any small areas of elevated activity exist that would require re-classification, tighter grid spacing for the fixed measurements, or both. However, MARSSIM also recognizes that alternatives to this general approach for FSSs exist. Specifically, MARSSIM states that if the equipment and methodology used for scanning are capable of providing data of the same quality as fixed measurements (e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of fixed measurements, provided that results are documented for at least the number of locations that would have been necessary had fixed measurements been used.

FSSs for the YNPS surface soils and structures will be designed, following MARSSIM guidance, using combinations of fixed measurements, traditional scanning surveys, and other advanced survey methods, as appropriate, to evaluate survey units relative to their applicable release criteria. As MARSSIM does not directly address FSS for subsurface soils, the principles of MARSSIM will guide the design of these surveys. Subsurface survey considerations can be found in Section 5.6.3.2.2.

Under MARSSIM, the level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Class 3 survey units receive judgmental scanning and randomly located measurements or samples. Class 2 survey units receive scanning over a portion of the survey unit based on the potential for contamination, combined with fixed measurements or sampling performed on a systematic grid. Class 1 survey units receive scanning over 100% of the survey unit combined with fixed measurements or sampling performed on a systematic grid. Depending on the sensitivity of the scanning method, the grid spacing may need to be adjusted to ensure that small areas of elevated activity are detected.

For combinations of fixed measurements and traditional scanning, MARSSIM methodology is to select a requisite number of measurement locations to satisfy the decision error rates for the non-parametric statistical test to be used for data evaluation and to account for sample losses or data anomalies. The purpose of scanning is to confirm that the area was properly classified and that any small areas of elevated activity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, the number of fixed measurement locations may need to be increased so the spacing between measurements is reduced. Details on selecting the number and location of fixed measurements are the subject of Section 5.5.1 and subsequent subsections of this plan. The coverage requirements that will be applied for scans performed in support of FSSs for the YNPS site are:

- 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 FSSs 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 FSSs 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 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 when advanced technologies are 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 $DCGL_W$ being acceptable. These same criteria will be used when deciding if advanced survey techniques can be used instead of fixed measurements and traditional scans for a given survey unit. MDCs for advanced techniques will be computed using background count rates obtained using appropriate reference materials.

With respect to the survey methods and techniques discussed above, the survey design criteria that will be employed for FSSs for the YNPS site are summarized below. Note that "fixed measurements" is used to refer to measurements or samples taken at specific locations.

- For Class 1 or Class 2 survey units, advanced survey technologies may be used exclusively only in survey units for which the above coverage requirements can be achieved and MDCs are no greater than 50% of the applicable $DCGL_W$.

- For Class 1 or Class 2 survey units for which advanced technologies would have an acceptable MDC, but the above coverage requirements cannot be achieved, advanced technologies may be used over 100% of the accessible area with a combination of fixed measurements and traditional scans used over the remainder of the area as specified in Section 5.5.1 and subsequent subsections of this plan.

For any survey units for which advanced survey techniques are impractical, fixed measurements and traditional scans will be used exclusively in accordance with this plan.

5.5.1 Selecting the Number of Fixed Measurements and Locations

The MARSSIM methodology for evaluating whether a survey unit meets its applicable release criterion using fixed measurements plus scans is based on using non-parametric statistical tests for data assessment. Specifically, the methods of MARSSIM are based on two non-parametric tests: the Wilcoxon Rank Sum (WRS) test and the Sign test, as discussed in Section 5.7. Selection of the required minimum number of data points depends on which statistical test is going to be used to evaluate the data, and thus depends on what type of measurements are to be made (gross measurement, net measurement or radionuclide specific) and if the radionuclide(s) of interest appear(s) in background.

5.5.1.1 Establishing Acceptable Decision Error Rates

One input to the process of selecting the required number of data points for a given survey, which does not depend on the statistical test applied, is the selection of the acceptable decision error rates. Decision errors refer to making false decisions by either rejecting a null hypothesis when it is true (a Type I error) or accepting a null hypothesis when it is false (a Type II error). With respect to FSSs, the null hypothesis is that the survey unit of interest contains residual contamination in excess of the applicable release criterion. Thus, a Type I error refers to concluding that an area meets the release criteria when in fact it does not. The probability of making a Type I error is referred to as alpha (α). Likewise, a Type II error refers to concluding a unit does not meet the release criteria when it actually does. The probability of making a Type II error is denoted beta (β). Selecting values of α that are too low will result in an excessive number of fixed measurements being required. Likewise, selecting a β value that is too large can 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 FSSs 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 FSS.

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:

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

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:

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

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 data points 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.

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:

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

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:

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

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 FSS. 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.

The following example shows how 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 as 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:

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

HTD radionuclides are included by using the surrogate ratio in determining the $DCGL_{EMC}$.

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 FSS activities. RAI #19

The following example shows how 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
ε_i = 0.3 for Co-60
ε_i = 0.38 for Cs-137
ε_s = 0.25 for Co-60
ε_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 for reasons such as an obstruction or inaccessibility.

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 FSS. 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 FSS Plan.

5.5.3 Data Investigations

5.5.3.1 Investigation Levels

An important aspect of the FSS is the selection and implementation of investigation levels. Investigation levels are levels of radioactivity used to indicate when additional investigations may be necessary. Investigation levels also serve as a quality control check to determine when a measurement process begins to deviate from expected norms. For example, a measurement that exceeds an investigation level may indicate a failing instrument or an improper measurement.

However, in general, investigation levels are used to confirm that survey units have been properly classified.

When an investigation level is exceeded, the first step is to confirm that the initial measurement/sample actually exceeds the particular investigation level. Depending on the results of the investigation actions, the survey unit may subsequently require re-classification, remediation, and/or re-survey. Investigation levels are established for each class of survey unit. The investigation levels (criteria), to be employed for the YNPS FSS effort, are given in Table 5-2.

Table 5-2
Investigation Levels

Survey Unit Classification	For fixed measurements or samples, perform investigation if:	For scan measurements, perform investigation if:
Class 1	> $DCGL_{EMC}$ or > $DCGL_W$ and a statistical outlier.	> $DCGL_{EMC}$
Class 2	> $DCGL_W$	> $DCGL_W$ or > MDC_{scan} if MDC_{scan} is greater than the $DCGL_W$
Class 3	> $0.5 \times DCGL_W$	Detectable over background.

For Class 1 survey units, measurements above the $DCGL_W$ are not necessarily unexpected. However, such a result may still indicate a need for further investigation if it is significantly different than the other measurements made within the same survey unit. Thus, some additional evaluation criterion is needed to assess if results from fixed measurements or samples in a Class 1 survey unit that exceed the $DCGL_W$ warrant further attention. Measurements in Class 1 survey units that exceed the $DCGL_W$ and differ from the mean of the remaining measurements by more than three standard deviations will therefore be investigated. Measurements in Class 1 units that exceed the $DCGL_W$, but do not differ from the mean by as much may still be investigated on the basis of professional judgment, as may any measurements that differ significantly from the rest of the measurements made within a given survey unit.

In Class 2 or Class 3 areas, neither measurements above the $DCGL_W$ nor areas of elevated activity are expected. Thus, any fixed measurements or sampling results that exceed the $DCGL_W$ in these areas will be investigated. In the case of Class 3 areas, where any residual radioactivity would be unexpected, fixed measurement or sample results that are greater than 0.5 x $DCGL_W$ will be investigated. Because the survey design for Class 2 and Class 3 survey units is not driven 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 FSS, 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 FSS 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 FSS 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 FSS design will be carried out in accordance with the SOPs and the QA plan and will result in the generation of raw data. The product of the survey design process is a survey package, which addresses various elements of the survey, including, but not limited to:

- maps of the survey area showing the survey unit(s) and measurement/sample locations, as appropriate;
- applicable DCGLs
- instrumentation to be used;
- types and quantities of measurements or samples to be made or collected;
- investigation criteria;
- QA/QC requirements (e.g., replicate measurements or samples);
- personnel training;
- applicable health and safety procedures;
- approved survey procedures; and
- applicable operating procedures.

An important element of the survey design process is establishing the DCGLs for the measurements to be made. The DCGLs will be determined as described in Section 5.4.6 based on characterization data for the survey unit(s) being considered. Isotopic mix, material backgrounds, and the variability of these will be considered. The detection limit requirements dictated by the DCGLs affect the selection of both the instrumentation to be used for a given survey and the survey method(s) to be employed (advanced survey methods, fixed measurements, sampling; or combinations thereof).

5.6.1 Survey Methods

The survey methods to be employed in the FSSs will consist of combinations of advanced technologies, scanning, fixed measurements, sampling, and other methods as needed to meet the survey objectives. Additional methods may be used if such become available between the time this plan is adopted and the completion of final survey activities. However, any new technologies must still meet the applicable requirements of this plan. Note that in some cases, the same instrument may be used for more than one type of survey. For instance, a sodium-iodide (NaI) detector may be used in either a scanning mode or for fixed spectroscopic measurements.

5.6.1.1 Scanning

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., ground, wall, floor, equipment).

The term scanning survey is used to describe the process of moving portable radiation detectors across a surface with the intent of locating residual radioactivity. Investigation levels for scanning surveys are determined during survey planning to identify areas of elevated activity. Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action. These investigation levels may be based on the $DCGL_W$ or the $DCGL_{EMC}$.

Table 5-3 gives the areal coverage requirements when scanning is used with fixed measurements.

Table 5-3
Traditional Scanning Coverage Requirements

Survey Unit Classification	Required Scanning Coverage Fraction
Class 1	100%
Class 2	Outdoor areas, floors, or lower walls of buildings: 10% to 100% Upper walls or ceilings: 10% to 50%
Class 3	Judgmental

5.6.1.2 Fixed Measurements

Fixed measurements are taken by placing the instrument at the appropriate distance above the surface, taking a discrete measurement for a pre-determined time interval, and recording the reading. Fixed measurements may be collected at random locations in a survey unit or may be collected at systematic locations and supplement scanning surveys for the identification of small areas of elevated activity. In addition, fixed measurements may be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify locations for fixed measurements to further define the areal extent of contamination. Locations for fixed measurements specified by a given survey design will be established as discussed in Section 5.5.

5.6.1.3 Advanced Technologies

In the context of the License Termination Plan, advanced technologies refer to survey instruments or methods that create a spatially-correlated log of the measurements made as the detector is passed over an area. This logging of the measurements allows quantitative assessments of activity levels to be made, thus serving the same role as fixed measurements. Having the measurements logged allows statistical analyses to be made using a large number of samples, which provides for enhanced detection sensitivity relative to traditional scanning. The sensitivity achieved using advanced survey methods may, in some cases, be small enough relative to the $DCGL_W$ that the advanced method alone will allow a decision to be made as to whether a survey unit meets the release criterion without the need for additional fixed measurements. The fact that the instrument records every measurement made over the entire area it covers inherently addresses the issue of small areas of elevated activity. Average and 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 FSS 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 FSSs include, but are not limited to, *in situ* gamma spectrometry, *in situ* object counting systems, and systems capable of traversing ducting or piping. Similar to 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. 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 FSS activities that apply to onsite and offsite laboratories employed to analyze samples as a part of the FSS 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.

Instruments must be stable and reliable under the environmental and physical conditions where they will be used, and their physical characteristics (size and weight) should be compatible with the intended application. This has been the case for typical radiation detection instrumentation used at YNPS for operational surveys as well as scoping and characterization surveys.

The radiation detectors to be used for final survey activities at the YNPS Plant can be divided into three general classes:

- gas-filled detectors,
- scintillation detectors, and
- solid-state detectors.

Gas-filled detectors include ionization chambers, proportional counters (both gas-flow and pressurized) and Geiger-Mueller (GM) detectors. Scintillation detectors include plastic scintillators, zinc-sulfide (ZnS) detectors and sodium-iodide (NaI) detectors. Solid-state detectors include both n-type and p-type intrinsic germanium detectors.

Finally, the DQO process must evaluate, depending on the type of radiation of interest, and on the application, the ability of instrumentation to measure levels that are less than the DCGL. In some cases instruments used for scanning may have detection limits that are greater than the $DCGL_W$. This is recognized by MARSSIM as an acceptable approach as long as the grid spacing (for Class 1 survey units) and investigation levels used are in accordance with Sections 5.5.1.5, 5.5.1.6 and 5.5.3.1, respectively, of this plan. The DQO process for instrument selection is performed in the planning phase for an FSS activity and is typically documented by a technical support document, which is referenced in the survey plan.

5.6.2.2 Calibration and Maintenance

Instrumentation used for measurements to demonstrate compliance with the radiological criterion for license termination at the YNPS Plant will be calibrated and maintained under approved plant procedures and the Yankee Decommissioning Quality Assurance Program (YDQAP) or vendor QA plan that satisfies the requirement of the YDQAP. Instruments will be calibrated for normal use under typical field conditions at the frequency specified by vendor instructions or by approved plant procedures (at least annually). Calibration standards will be traceable to the National Institute of Standards and Technology (NIST). If external vendors are used for instrument calibration or maintenance, these services must be approved and conducted under the YDQAP. Calibration records will be maintained as required by plant procedures and the YDQAP.

Instruments used to measure gross beta surface activity will be calibrated using radionuclides such as Tc-99, Co-60, or Cs-137 so as to represent the beta energies for the beta-emitting radionuclides that will be encountered during final survey activities. Likewise, if direct measurements are performed for alpha-emitting radionuclides, radionuclides such as Pu-239 or Th-230 will likely be used to calibrate instruments used to assess alpha surface activity so the alpha energies of the TRU radionuclides that may be encountered are adequately represented.

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 is given to show how 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 FSS.

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 of 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

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

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

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 FSS 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[TM] 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 in a calculated exposure rate equal to 1.307 μR/h. The scan MDC is calculated by dividing the MDER by the localized contamination exposure rate conversion factor as follows:

$$Cs-137 scanMDC = 5pCi/g\frac{1.33\mu R/h}{1.307\mu R/h} = 5.1pCi/g$$

The scan MDCs will be documented prior to performing the FSS.

5.6.2.5 Scan MDCs for Hot Particles

The scan MDC and scan methodologies for instruments used for structure surfaces (beta sensitive detectors) and land areas (gamma sensitive detectors) are capable of detecting very small areas of elevated radioactivity that could be present in the form of small particles (i.e. hot particles). The minimum detectable particle activity for these scanning instruments and methods correspond to a small fraction of the TEDE limit provided in 10CFR20 subpart E.

5.6.2.6 Typical Instrumentation and MDCs

Table 5-4 provides nominal data for the types of field instrumentation anticipated for use in the final survey efforts for YNPS. The efficiencies listed in Table 5-4 are the total efficiencies in counts/disintegration, and the background count-rates shown are nominal values for generic materials. This table is provided to show the relative sensitivity of some of the types of instruments that will be used during the FSS and allow the readers to compare the sensitivities to the DCGLs in Section 6 of the LTP. The instrument efficiency (ε_I) and source efficiency (ε_s) will be evaluated for instruments used for FSS measurements and documented as part of the calibration records. This evaluation will include the effects of surface to detector distances, surface coatings and the depth of contamination in material (e.g., concrete) on instrument performance. Instrument calibration sources will be chosen that are appropriate for use for the radionuclides expected to be present post remediation. Instrument readings will be converted to activity by selecting conservative efficiency factors based upon the building surface conditions (including the depth of contamination in concrete).

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 Survey Considerations

The available complement of survey instrumentation and techniques will be evaluated to select an integrated approach that will effectively measure residual radioactivity for a given survey unit. The survey design must rely on both the historical site assessment and pertinent data from characterization or remediation support surveys to ensure a complete survey approach. Considerations that will be addressed in the selection of survey instrumentation and techniques include, but are not limited to:

- the types of measurements required;
- suitability for the expected physical and environmental conditions;
- MDCs for advanced survey methods, traditional scanning surveys, fixed measurements, and sampling relative to the $DCGL_W$ and the $DCGL_{EMC}$;
- radionuclide mix, including difficult-to-detect and alpha-emitting radionuclides;
- expected spatial variability of any suspected residual contamination;
- accessibility of areas (may impact coverage for scanning surveys); and
- the need for any judgmental assessments to address areas believed to have a higher potential for contamination or situations such as potential sub-surface contamination where prudence would dictate some additional sampling.

5.6.3.1 Survey Considerations for Buildings and Structures

The condition of surfaces following decontamination activities can affect the choice of survey instruments and techniques. Removing contamination that has penetrated a surface usually involves removing the surface material. As a result, the floors and walls of decontaminated facilities can be scarred or broken up and uneven. Such surfaces are more difficult to survey because it is not possible to maintain a fixed distance between the detector and the surface. In addition, scabbled or porous surfaces may attenuate radiation - particularly alpha and low-energy beta particles, and pose an increased risk of damage to detector probe faces. Surface irregularities may also cause difficulty in rolling or maneuvering detector systems on wheels.

Part of the planning for the FSS of a particular survey unit will include an evaluation of the surfaces to be monitored. For conventional instrumentation, surface anomalies will be identified as part of this process and will be taken into account when selecting efficiencies to convert instrument readings to activity and in the calculation of the corresponding MDCs. Conservative values will be chosen based upon surface conditions. If the condition of the surface in the area changes in a more conservative direction (e.g. shorter detector to surface distance), the effect on the MDC will be assessed but may not be re-derived. If the condition of the surface changes in a non-conservative direction (e.g. different construction material which has higher natural radioactivity) the MDC will be assessed and re-derived.

Expansion joints, stress cracks, floor/wall interfaces, and penetrations into floors and walls for piping, conduit, anchor bolts, etc., are potential sites for accumulation of contamination and pathways for migration into sub-floor soil and hollow wall spaces. Roof surfaces and drainage points are also important survey locations. In some cases, it may be necessary to core, drill, or use other methods as necessary to gain access to areas for sampling.

5.6.3.1.1 Activity Beneath Surfaces

Floors and walls of structures may have surface irregularities such as cracks and crevices that require special consideration in the survey process. Such considerations may consist of fixed measurements, longer count times, adjustments to counting efficiencies, sampling of material, or any combination of these approaches.

Plant areas where residual radioactive material beneath a painted surface is known or suspected to be present will also require special consideration. Sampling will be performed, as appropriate, to confirm or deny the presence of residual activity. If activity is found, the samples should be used to determine both the radionuclides that are present and the density-thickness of the paint layer(s) in order to assess the need for correction factors for counting efficiencies. Such corrections, if required, will be determined following the guidance given in Section 5 of NUREG-1507. The effect of any such corrections on instrument MDCs will be assessed to ensure that measurements can still be performed with the required sensitivity relative to the applicable DCGLs.

5.6.3.1.2 Exterior Surfaces of Building Foundations

Exterior surfaces of below-grade foundations will be evaluated using the historical site assessment and other pertinent records to determine the potential for sub-surface contamination on the exterior surfaces of below-grade foundations. One method available to evaluate the exterior surfaces is the use of core bores through foundation or walls and the taking of soil samples at locations having a high potential for the accumulation and migration of radioactive contamination to sub-surface soils. These biased locations for soil and concrete assessment could include stress cracks, floor and wall interfaces, penetrations through walls and floors for piping, run-off from exterior walls, and leaks or spills in adjacent outside areas, etc. If the soil is found to be free of residual radioactivity at the biased locations, it will be assumed that the exterior surface of the foundation is also free of residual activity. Otherwise, additional sampling may be necessary to determine the extent of decontamination and remediation efforts. Another method available for evaluating the exterior surfaces of below-grade foundations is gamma well logging. Soil in biased locations next to the exterior of the buildings may be evaluated using this technique. This technique can provide for rapid isotopic analysis of soils without sampling.

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 performed 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 the 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 HTD 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, as well as 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 will be deferred until later in the decommissioning process. As a part of survey planning, borehole logs will be reviewed, when available.

The DQO process for subsurface areas will be similar to the DQO process used for other surveys at YNPS (e.g., FSS for surface soils). However, there may be differences in design input parameters as necessary to satisfy the objectives of the plan. Additional detail regarding subsurface input parameters and methodology are provided below. Surveys (i.e., characterization, remediation and FSS) for subsurface areas will be performed under a documented survey plan developed using the DQO process. The level of effort with which the DQO process is used as a planning tool is commensurate with the type of survey and the necessity of avoiding a decision error. This is the graded approach of defining data quality requirements as described previously in the LTP. For example, characterization survey plans intended to collect data might only require a survey objective and the instrumentation and analyses specifications necessary to meet that survey objective. Remediation and final status plans which require decisions would need additional effort during the planning phase according to the level of risk of making a decision error and the potential consequences of making that error.

Evaluation of subsurface soil at YNPS during FSS will be a combination of systematic and biased measurements. Measurements may be either *in-situ* gamma spectroscopy by well logging or other advanced technology, provided the MDC meets the criteria discussed in Section 5.6.2.5, or by sampling. If advanced technology instrumentation is selected for use, a technical support document will be developed to describe 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 FSS activities. Sample locations will use a random start, systematic grid, supplemented with biased measurements. Biased measurements will be obtained at the locations of localized contamination. Where samples are taken, each 3-meter core will be homogenized and measured.

The horizontal extent of contamination will be investigated by judgmental sampling in areas that exceed the $DCGL_w$ and for samples within a systematic sampling area that exceed the $DCGL_{EMC}$. For the case where the $DCGL_{EMC}$ comparison is made, the value used for the area factor will be determined from the area bounded by the adjacent samples or by the area bounded by the locations that exceed the $DCGL_W$. Thus, for samples that exceed the $DCGL_{EMC}$, the investigation criteria will be the $DCGL_W$. This approach is consistent with the model used to calculate DCGLs in Section 6.

As discussed in Section 2.6, a portion of the YNPS industrial area has been identified as requiring additional investigation of subsurface soils. Twenty-five (25) measurement locations will be sampled in this area (see Figure 2-6 for the area of additional subsurface investigation). Biased measurements or samples will be obtained in these areas based upon characterization data and professional judgment. If a calculated sample location falls on a building foundation, a sample will be obtained at that location unless the building is in contact with bedrock. All samples will be evaluated against the soil DCGLs by using either the Sign or WRS test.

Investigation levels applicable to surface soils (given in Table 5-2) will be applied to subsurface soils. Similarly the area factors for surface soils will be applied to subsurface soils. That is, no sample can exceed the $DCGL_{EMC}$ without an investigation being performed. These investigations would be similar to those performed for surface soils.

Samples will be obtained to a depth of 3 meters or bedrock, whichever is reached first. These samples will be homogenized over the entire depth of the core obtained. In cases where refusal is met because of bedrock, the sample will be used "as is." In cases where a non-bedrock refusal is met prior to the 3-meter depth, the available sample will be used to represent the 3-meter sample, if the viable sample is at least 1.5 meters in depth. If a non-bedrock refusal is met before the 1.5-meter depth, then a new sample will be obtained within a 3-meter radius from the original location. Samples will be analyzed by gamma spectrometry. A minimum of 5% of the samples will be analyzed for HTD radionuclides. During specific investigations, analysis of a larger percentage of samples for HTD radionuclides will be performed.

5.6.3.2.3 Paved areas

Paved areas that remain at the YNPS following decommissioning activities may require surveys for residual radioactivity on the surface, beneath the surface, or both. As part of the survey design and planning process, historical information will be reviewed to determine whether radiological incidents or plant alterations have occurred in the survey unit. Where indications are that impacted soil could have been mixed by grade work prior to paving, this will be factored into final survey design to establish a reasonable depth of disturbed soil for evaluation. If it is determined that the soil beneath pavement has been impacted, the FSS will incorporate appropriate surveys and sampling.

If residual radioactivity is primarily on or near the surface of the paved area, for purposes of surveying, measurements will be taken as if the area were surface soil. If the residual radioactivity is primarily beneath the paving, it will be treated, for purposes of surveying, as subsurface residual radioactivity.

5.6.3.2.4 Groundwater

Assessments of any residual activity in groundwater at the YNPS will be via groundwater monitoring wells. The monitoring wells installed at the site will monitor groundwater at both deep and shallow depths. Section 2.7 describes the groundwater monitoring to be conducted.

The data collected from the monitoring wells, across multiple aquifers, will be used to ensure that the concentration of well water available, based upon the well supply requirements assumed in Section 6 for the resident farmer, is below the EPA MCLs (e.g., 20,000 pCi/l for H-3). This will ensure that the dose contribution from groundwater is a small fraction of the limit in 10CFR20.1402.

5.6.3.2.5 Sediments

Sediments will be assessed by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments, as appropriate. Such samples will be collected using approved procedures based on accepted methods for sampling of this nature. Sample locations will be established using the methods of Section 5.5.1 of this plan. Scanning in such areas is not applicable.

Sediment samples will be evaluated against the DCGLs for soil. This is considered appropriate given that the action that would result in the greatest radiological impact to future inhabitants of the site would be to dredge up the sediment and use it for farming. If the sediment is left in place, then use of the soil DCGLs is conservative since many of the pathways considered in developing the soil DCGLs (direct exposure, uptake by plants, etc.) would not apply.

Assessment of residual activity levels in surface water drainage systems will be via sampling of sediments and/or fixed measurements, taking measurements at traps and other appropriate access points where activity levels should be representative or bound those on the interior surfaces.

5.7 Final Status Survey Data Assessment

The Data Quality Assessment (DQA) process is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process will include a review of the DQOs and survey plan design, will include a review of preliminary data, will use appropriate statistical testing when applicable (statistical testing is not always required, e.g., when all sample or measurement results are less than the $DCGL_W$), will verify the assumptions of the statistical tests, and will draw conclusions from the data. Application of DQAs will be described in greater detail in the YNPS FSS (FSS) Quality Assurance Project Plan.

Prior to evaluating the data collected from a survey unit against the release criterion, the data are first confirmed to have been acquired in accordance with applicable procedures and QA/QC requirements. Any discrepancies between the data quality or the data collection process and the applicable requirements are resolved and documented prior to proceeding with data analysis. Data assessment will be performed, by trained personnel, using approved site procedures.

The first step in the data assessment process is to convert the survey results to DCGL units. Next, the individual measurements and sample concentrations will be compared to DCGL levels for evidence of small areas of elevated activity or results that are statistical outliers relative to the rest of the measurements (see Section 5.5.3.1). Graphical analyses of survey data that depict the spatial correlation of the measurements are especially useful for such assessments and will be used to the extent practical. The results may indicate that additional data or additional remediation and re-survey may be necessary. If this is not the case, the survey results will then be evaluated using direct comparisons or statistical methods, as appropriate, to determine if they exceed the release criterion. If the release criterion has been exceeded or if results indicate the need for additional data points, appropriate further actions will then be determined.

Interpreting the results from a survey is most straightforward when all measurements are higher or lower than the $DCGL_W$. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the $DCGL_W$.

The first step in evaluating the data for a given survey unit is to draw simple comparisons between the measurement results and the release criterion. The result of these comparisons will be one of three conclusions: 1) the unit meets the release criterion; 2) the unit does not meet the release criterion; or 3) no conclusion can be drawn from simple comparisons and thus one of the non-parametric statistical tests must be applied. The initial comparisons made for the results for a given survey unit depend on whether or not the results are to be compared against a background reference area.

If the survey data are in the form of gross (non-radionuclide-specific) measurements or if the radionuclide of interest is present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation will be as described in Table 5-5.

Table 5-5
Initial Evaluation of Survey Results
(Background Reference Area Used)

Evaluation Result	Conclusion
Difference between the maximum concentration measurement for the survey unit and the minimum reference area concentration is less than the $DCGL_W$	Survey unit meets the release criterion
Difference between the average concentration measured for the survey unit and the average reference concentration is greater than the $DCGL_W$	Survey unit does not meet the release criterion
Difference between any individual survey result and any individual reference area concentration is greater than the $DCGL_W$ and the difference between the average concentration and the average for the reference area is less than the $DCGL_W$	Conduct either the Wilcoxon Rank Sum test or the Sign test; and the EMC test

If the survey data are in the form of radionuclide-specific measurements and the radionuclide(s) of interest is not present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation will be as described in Table 5-6.

Table 5-6
Initial Evaluation of Survey Results
(Background Reference Area Not Used)

Evaluation Result	Conclusion
All measured concentrations less than the $DCGL_W$	Survey unit meets the release criterion
Average concentration exceeds the $DCGL_W$	Survey unit does not meet the release criterion
Individual measurement result(s) exceeds the $DCGL_W$ and the average concentration is less than the $DCGL_W$	Conduct the Sign test and the EMC test

5.7.1 Wilcoxon Rank Sum Test

Gross activity measurements or measurements for which the radionuclide of interest exists in background in concentrations that are a relevant fraction of the $DCGL_W$ may be evaluated using the Wilcoxon Rank Sum (WRS) test. In the WRS test, comparisons are made between the survey results for a given survey unit and reference (background) data for comparable materials. However, for survey units which contain multiple materials having different backgrounds, it may be advantageous to background-subtract gross activity measurements (using paired observation) and apply the Sign test (see Section 5.7.2).

The WRS test tests the null hypothesis that the median concentration in the survey unit exceeds that in the reference area by more than the $DCGL_W$. The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected. The other possibility is that the median concentration in the survey unit exceeds that in the reference area by less than the $DCGL_W$. Note that some or all of the survey unit measurements may be larger than some reference area measurements, while still meeting the release criterion. Indeed, some survey unit measurements may exceed some reference area measurements by more than the $DCGL_W$. The result of the hypothesis test determines whether or not the survey unit as a whole is deemed to meet the release criterion. The EMC is used to screen individual measurements.

The WRS test is applied as described in the following steps:

1. List the survey measurements.
2. Adjust the reference area measurements by adding the $DCGL_W$ to each one.
3. Pool the adjusted reference area measurements and the sample (survey unit) measurements and rank them in increasing order from 1 to the total number of data points (reference measurements plus sample measurements).
4. For any measurements that have the same value, the rank assigned to that set of measurements is the average of their ranks.
5. Sum the ranks of the adjusted reference area measurements.
6. Compare the sum of the adjusted reference area measurements (W_r) with the critical value from Table I.4 of the MARSSIM for the appropriate values of m (the number of

reference measurements), n (the number of sample measurements), and α (the decision error rate).

If the value W_r determined from steps 1 through 6 above exceeds the critical value from Table I.4 of the MARSSIM, then the null hypothesis is rejected and the alternate accepted. In other words, the results show that the survey unit meets the release criterion.

Note that the WRS test described in steps 1 through 6 above assumes that there are no “less than” results in the data set, i.e., that all of the data points have a quantitative value rather than “background” or “less than MDC.” Though it is not anticipated that data of this nature would be among that collected for an FSS, if it is encountered and must be used, the method described in Section 8.4.2 of the MARSSIM will be used to assign rank to these values. If more than 40% of the data collected for an FSS are “less than” values, then the WRS test cannot be used.

5.7.2 Sign Test

Radionuclide specific measurements for which the radionuclide(s) of interest either does not exist in background or is not present in a concentration that is a relevant fraction of the $DCGL_W$ will be evaluated using the Sign test. In addition, the Sign test may be used to evaluate gross activity measurements from survey units containing multiple materials by subtracting the appropriate background using paired measurements.

The null hypothesis tested by the Sign test is the same as that used for the WRS test. As with the WRS test, some individual survey unit measurements may exceed the $DCGL_W$ even when the survey unit as a whole meets the release criterion. In fact, a survey unit average that is close to the $DCGL_W$ might have almost half of its individual measurements greater than the $DCGL_W$. Such a survey unit may still not exceed the release criterion. As with the WRS test, the EMC is used to screen individual measurements.

The Sign test is applied as described in the following steps:

1. List the survey measurements.
2. For each survey unit measurement, subtract the measurement from the $DCGL_W$ and record the differences.
3. Discard any difference that is exactly zero and reduce the total number of measurements (N) by the number of zero differences.
4. Count the number of positive differences. This value is the test statistic S+.
5. Compare the number of positive difference (S+) to the critical values from Table I.3 of MARSSIM for the appropriate values of N (total measurements) and α (decision error rate). (A positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the release criterion.)

If S+ is greater than the critical value in Table I.3 of MARSSIM, then the null hypothesis is rejected and the alternate accepted.

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 an FSS 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.:

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

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 - β, 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 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 #40

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 might be found to be less than 1. The planning team would adjust the LBGR, evaluate the impact on power and accept 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. RAI #41

5.8 Final Status Survey Reports

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

- An overview of the results of the FSS;
- A discussion of any changes that were made in the FSS 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 $DCGL_w$;
- A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity;

- If a survey unit fails, a description of the investigation conducted to ascertain the reason for the failure and a discussion of the impact that the failure has on the conclusion that the facility is ready for final radiological surveys; and

- If a survey unit fails, a discussion of the impact that the reason for the failure has on other survey unit information.

In most cases, FSS results will be made available to the NRC for survey areas rather than for individual survey units. Where appropriate, FSS reports may address multiple survey areas. This approach should minimize the incorporation of redundant historical assessment information and provide for a logical approach to review and independent verification in that a more complete description of the final radiological status of an area will be provided.

5.9 Final Status Survey Quality Assurance and Quality Control Measures

5.9.1 Introduction

YAEC has developed and is implementing a comprehensive Quality Assurance Program to ensure conformance with the established regulatory requirements of the Nuclear Regulatory Commission (NRC) and accepted industry standards. The participants in the Yankee Decommissioning Quality Assurance Program (YDQAP) ensure that the design, procurement, construction, testing, operation, maintenance, repair, and modifications are performed in a safe and effective manner.

The YDQAP satisfies the criteria set forth in Appendix B of 10CFR50 and is approved by the NRC as sufficient to meet the requirements of 10CFR50, 10CFR71, and 10CFR72 for QA Programs. References to specific industry standards for quality assurance and quality control measures governing FSS activities will be reflected in supporting procedures, plans and instructions.

The quality assurance (QA) and quality control (QC) measures of the YDQAP have been integrated into the decommissioning activities, including the development of the LTP and eventual implementation of the FSS. FSS activities essential to data quality will be implemented and performed under approved procedures. Effective implementation of administrative controls will be verified through surveillance and audit activities. Corrective action will be prescribed, implemented and verified in the event any deficiencies are identified. These measures apply to the related services provided by off-site vendors, in addition to any on-site vendors and contractors or sub-contractors.

With respect to the FSS, QA/QC activities will serve to ensure that trained individuals perform the surveys. These surveys be performed using approved written procedures and properly calibrated instruments that are sensitive to the suspected contaminant. In addition, QC measures will be taken to obtain quantitative information to demonstrate that results have the required precision and are sufficiently free of errors to accurately represent the site being investigated. QC checks will be performed as prescribed by the implementing procedures for both field measurements and laboratory analysis (both on-site and third party). The YAEC Nuclear Safety organization will assess the performance of FSS activities.

5.9.2 Organization

The organization described herein is defined in the YDQAP. The Chief Executive Officer (CEO) reports to the YAEC Board of Directors. The CEO is the final management authority responsible for assuring that the YDQAP is effectively implemented by the YAEC organization. The President reports to the CEO and has the necessary authority and assigned responsibility for developing, maintaining, and implementing the YDQAP. The President has delegated this responsibility to the Vice President. The Site Manager reports to the Vice-President and is responsible for implementing the YDQAP during decommissioning of the facility. The Nuclear Safety Manager reports to the Vice President and is responsible for the Quality Assurance function, which will provide independent audits and surveillances for the FSS. An organizational chart of the FSS is provided in Figure 5-1

5.9.3 Program Controls

Program Controls shall be established for performing specific FSS activities. Activities will be accomplished using suitable instructions, procedures and drawings that incorporate appropriate regulatory and industry guidance.

Personnel conducting activities shall be appropriately trained and qualified. Training, qualification, and any appropriate maintenance of proficiency requirements shall be defined in administrative procedures or instructions. Professional resumes, other verifiable credentials, and/or discrete certification packages, as applicable, shall be used to document personnel qualifications.

5.9.4 Design Controls

Design control requirements are established to ensure that the applicable regulatory bases, codes, technical standards, and quality standards are identified in the FSS. Design controls also include independent verification and design interface control. These design controls will be implemented to determine the DCGLs, MDCs, area factors, and other DQO and FSS elements.

5.9.5 Procurement Document Control

Procurement documents related to the FSS shall be prepared in accordance with approved procedures and instructions. These procedures and instructions shall contain provisions to assure that procurement documents include or reference applicable regulatory requirements and any other requirement necessary to assure adequate quality for the purchased service, equipment, or material.

5.9.6 Instructions, Procedures, and Drawings

The performance of the FSS will require procedures for personnel training, survey implementation, data collection, chain of custody, instrument calibration and maintenance, verification, and record storage. These procedures will ensure compliance with the License Termination Plan and will meet applicable quality requirements. These include the development and approval in accordance with the site controls.

5.9.7 Document Control

Instructions, Procedures and Drawings shall be controlled as described in approved procedures or instructions. Controlled copies shall be available for use by personnel performing activities affecting the FSS Program. These controls shall assure that only current information is issued and used. The results of the FSS will be retained at least for the duration of the possession only license.

5.9.8 Control of Purchased Material, Items, and Services

Vendors may be used for the performance of the FSS and laboratory activities. Quality related services, such as instrument calibration and laboratory analysis, are procured from qualified vendors whose internal QA program is subject to approval in accordance with the YAEC Quality Assurance Program. Additionally, audits and surveillances of these contractors will be performed to provide an adequate level of assurance that the quality activities are being effectively performed and conform to the requirements of the procurement document.

5.9.9 Control of Special Processes

Procedures will be developed to implement any special processes that may be utilized in support of FSS implementation. The special processes utilized will be validated and will be implemented by trained, qualified individuals using approved procedures.

5.9.10 Inspections

Inspections and verification activities will be delineated in implementing procedures. These programs and procedures will be used to verify that sampling and surveying protocols are appropriately performed. Appropriate members of the line organization that are qualified, or an independent organization, as described in administrative procedures may perform these inspections.

5.9.11 Control of Measuring and Test Equipment

Approved procedures will be developed for the control, use, calibration, and testing of the equipment utilized for the FSS, including both laboratory and field use equipment. These procedures will ensure confidence in the data obtained. Instrument calibrations will be performed periodically in accordance with appropriate industry standards.

5.9.12 Handling, Storage and Shipping

Some of the material samples will be transported to off-site laboratories for analysis. The process for controlling this material will be sufficient to ensure that a chain-of-custody is maintained. Measures shall be established to ensure that samples are received, handled, stored, packaged, and shipped in accordance with approved procedures or instructions. These procedures or instructions shall be responsive to applicable industry or manufacturer's requirements and include controls for "shelf life" of sensitive products. Additionally, protocols must be established to ensure that there is no cross-contamination between samples and sample packaging. Appropriate controls will be defined in administrative procedures to ensure that sample integrity is maintained.

5.9.13 Control of Nonconformances

During the performance of the FSS, non-conforming conditions may be identified with equipment or services. The data associated with the non-conforming condition will be controlled until such time that it is accepted, rejected, or reworked in accordance with an appropriate procedure. Non-conforming equipment will not be utilized until conformance with applicable requirements has been established.

5.9.14 Corrective Action Program

The existing Corrective Action Program established under the YDQAP will be utilized for the FSS Program to ensure conditions adverse to quality are promptly identified and corrected.

5.9.15 Records

Measures have been established that ensure that FSS records are maintained as quality records. These measures also include procedures by which the records are reviewed and approved and procedures which ensure the records can be retrieved within a reasonable period of time. The controls shall also provide for the protection of the records to ensure that they are not lost or subject to degradation over time.

5.9.16 Audits

Audits of FSS activities will be performed periodically, in accordance with approved procedures or instructions, to verify the implementation of quality activities.

5.10 References

5-1 Title 10 to the Code of Federal Regulations, Part 20.1402, "Radiological criteria for unrestricted use."

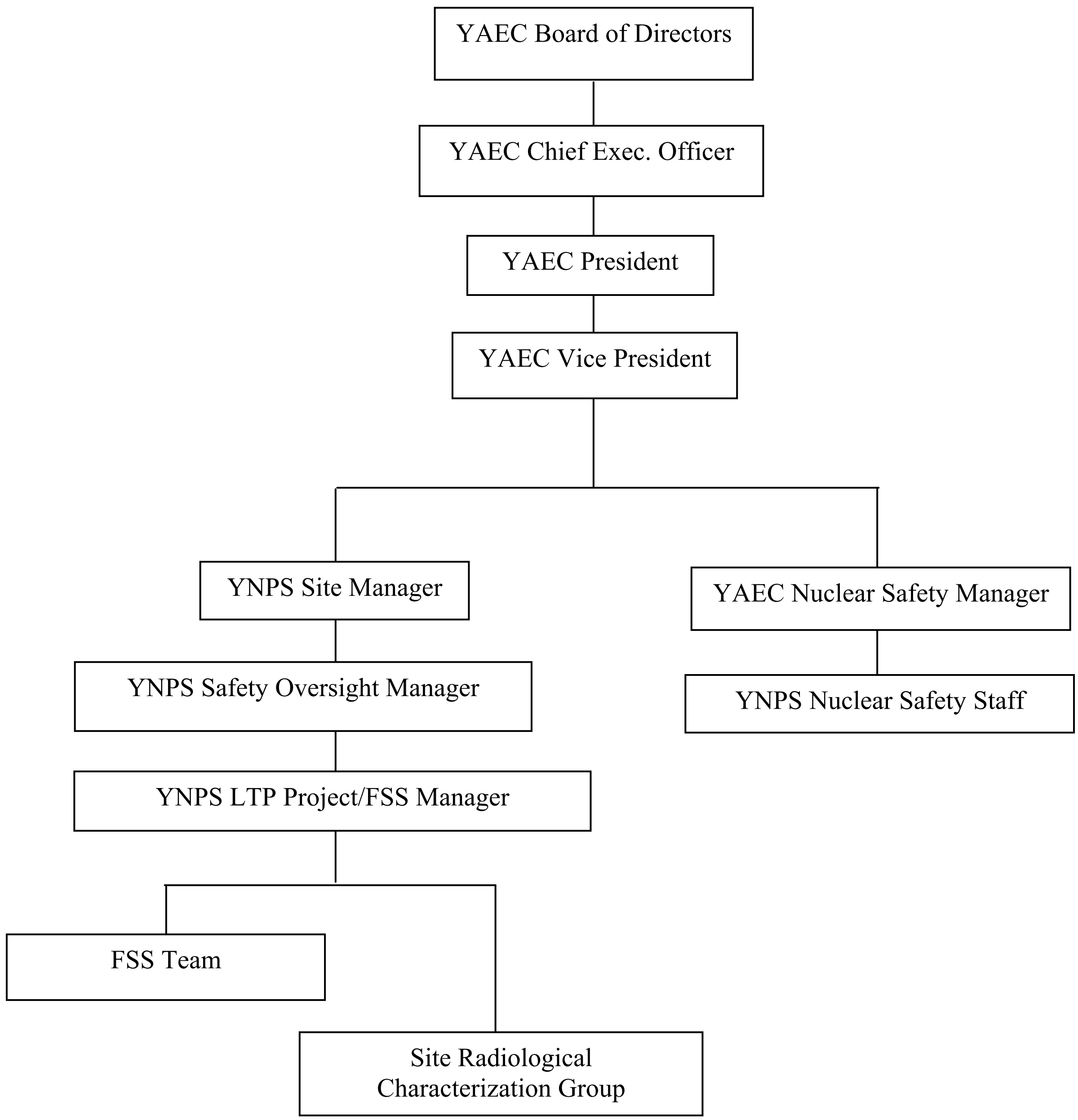
5-2 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual," Revision 1, dated August 2000.

5-3 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Power Reactors," dated January 1999.

5-4 NUREG-1727, "NMSS Decommissioning Standard Review Plan," dated September 2000.

5-5 NUREG-1757, Volume 2, "Consolidated NMSS Decommissioning Guidance," dated September 2003.

Figure 5-1
FSS Organization



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6 COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 Site Release Criteria

6.1.1 Radiological Criteria for Unrestricted Use

The site release criteria for the Yankee Nuclear Power Station (YNPS) site are the NRC's radiological criteria for unrestricted use given in 10 CFR 20.1402 (Reference 6-1):

- Dose Criterion: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/year, including that from groundwater sources; and

- ALARA Criterion: The residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).

6.1.2 Conditions Satisfying the Site Release Criteria

Levels of residual radioactivity that correspond to the allowable radiation dose and ALARA levels described above are calculated by analysis of various scenarios and pathways (e.g., direct radiation, inhalation, ingestion) through which exposures could be reasonably expected to occur. LTP Section 2.3.2 discusses the radionuclides for which derived concentration guideline levels (DCGLs) must be calculated. These DCGLs form the basis for the following conditions which, when met, satisfy the site release criteria as prescribed in 10 CFR 20.1402:

- The average residual radioactivity above background is less than or equal to the DCGL.

- Individual measurements representing small areas of residual radioactivity that exceed the DCGL, do not exceed the elevated measurement comparison DCGL. The elevated measurement comparison DCGL ($DCGL_{EMC}$) is described in Section 5.4.6.3.

- Where one or more individual measurements exceed the DCGL, the average residual radioactivity passes the Wilcoxon Rank Sum (WRS) or Sign statistical tes. (see Section 5.7.1 and 5.7.2 for a detailed discussion on application of statistical tests).

- Remediation is performed where ALARA considerations require that levels of residual radioactivity be below DCGLs (see Section 4 and Appendix 4.A for detailed discussions of ALARA considerations).

The methods in MARSSIM (Reference 6-2) and the DCGLs may not be appropriate for non-structural components (such as conduit and piping). For those non-structural components and systems to which MARSSIM does not apply, the current "no detectable" criteria (consistent with IEC 81-07) will be applied to free release these items.

6.2 Dose Modeling Approach

6.2.1 Overview

Dose models were developed, which translate levels of residual radioactivity into potential radiation doses to the public. Dose models, appropriate to the YNPS site, are based on the guidance found in NUREG-1549 (Reference 6-3) and NUREG/CR-5512, Volumes 1, 2, and 3 (Reference 6-4). A conceptual model was based on the site conditions expected at the time of unrestricted release. Conditions at the YNPS site, such as potential soil contamination at depths greater than 15 cm below the soil surface, require that site-specific dose modeling be performed. The dose modeling approach taken for the YNPS site is consistent with the information provided in Section 5 and Appendix I of NUREG-1757 (Reference 6-5) for site-specific modeling, including the information regarding source term abstraction and scenarios, pathways, and critical groups.

The dose model is defined by three factors: 1) the scenario, 2) the critical group and 3) the exposure pathways. The scenarios described in NUREG/CR-5512, Volume 1, address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. The scenarios also identify the critical group, which is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumptions of the particular scenario. The scenarios and their modeling are specifically designed to be reasonably conservative by generally overestimating rather than underestimating potential dose.

The approach outlined above was used to develop dose models to calculate DCGLs for soil and concrete. It should be noted that the scenarios described in NUREG/CR-5512, Volume 1, were developed to estimate potential radiation doses from radioactive material in buildings (building occupancy scenario) and soil.

6.2.2 Resident Farmer Scenario

6.2.2.1 Scenario Definition

The resident farmer scenario, described as the "Residential Scenario" in NUREG/CR-5512, Volume 1, was selected to conservatively estimate human radiation exposures resulting from residual radioactivity in soil to determine corresponding DCGLs.

6.2.2.2 Critical Group

The average member of the critical group was determined to be the resident farmer who lives on the plant site following decommissioning, grows all or a portion of his/her diet on site, and uses the water from a groundwater source on the site for drinking water and irrigation. The dose from residual radioactivity in soil is evaluated for the average member of the critical group as required by 10 CFR Part 20, Subpart E, and described in NUREG -1757, Appendix I, and NUREG-1549.

It is unlikely that any other set of plausible human activities could occur onsite that would result in a dose exceeding that calculated for the hypothetical resident farmer. It is more likely that the behavior of future occupants would result in a lower dose. For example, it is more likely that the YNPS site will be reused for land conservation. The hypothetical dose from the soil to an individual in these settings would be less than for a resident farmer, since the such an individual would not ingest food grown onsite. Therefore, the use of the resident farmer as the average member of the critical group is both conservative and bounding for the calculation of soil DCGLs.

6.2.2.3 Exposure Pathways

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.

6.2.3 Building Occupancy Scenario

6.2.3.1 Scenario Definition

The building occupancy scenario, based upon NUREG/CR-5512, Volume 1, was selected to estimate human radiation exposure resulting from residual radioactivity in concrete from standing buildings and to determine corresponding DCGLs.

6.2.3.2 Critical Group

Given the fact that the buildings associated with the YNPS site are commercial, the average member of the critical group is an adult engaging in light industrial work within the buildings following decommissioning of the site. The person occupies a commercial facility performing standard activities that do not deliberately disturb sources of residual radioactivity. The dose from residual radioactivity in the concrete from the standing building is evaluated for the average member of the critical group as required by 10 CFR Part 20, Subpart E, and described in NUREG -1757, Appendix I.

6.2.3.3 Exposure Pathways

The potential exposure pathways, described in NUREG/CR-5512, Volume 1, are listed below:

- Direct exposure to external radiation from
  - Material deposited on the walls
  - Material deposited on the floor
  - Submersion in airborne dust
- Internal dose from inhalation of airborne radionuclides
- Internal dose from inadvertent ingestion of radionuclides

6.2.4 Code Selection

The RESRAD Family of Codes has been selected for use at YNPS. The RESRAD computer codes are pathway-analysis models developed at Argonne National Laboratory (ANL). This family of computer codes includes RESRAD, used to analyze pathways associated with soil, and RESRAD-BUILD, used to analyze pathways associated with buildings.

The RESRAD computer code (Version 6.21) was used in this analysis to consider three major exposure pathways to a resident farmer:

- Direct exposure to external radiation from soil containing residual radioactivity;
- Internal exposure from inhalation of airborne radionuclides; and
- Internal exposure from ingestion of radionuclides.

The RESRAD-BUILD computer code (Version 3.21) is used in this analysis to consider five exposure pathways to occupants of a building:

- External exposure directly from the sources;
- External exposure to material deposited on the floor;
- External exposure due to air submersion;
- Inhalation of airborne radioactive particulates; and
- Inadvertent ingestion of radioactive material directly from the sources.

Information on the use of these codes and their applications are outlined in NUREG/CRs-6676, -6692, -6697, -6755 (References 6-6, 6-7, 6-8 and 6-9) and the "Users Manual for RESRAD, Version 6-0" (Reference 6-10).

6.2.5 Input Parameter Selection Process

The dose and conceptual models are quantified by a set of input parameters. Incorporated within RESRAD Version 6.21 and RESRAD-BUILD Version 3.21 are probabilistic modules that allow the evaluation of dose as a function of parameter distributions. A schematic flow diagram of the parameter selection process is provided in Figure 6-1. Each step of the selection process is discussed below.

6.2.5.1 Classification (Type):

The input parameters were classified as behavioral, metabolic or physical, consistent with NUREG/CR-6697. Behavioral parameters depend on the behavior of the receptor and the scenario definition. Metabolic parameters represent the metabolic characteristics of the receptor and are independent of the scenario definition. Physical parameters are those 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 values for parameters not already assigned a value (see Section 6.2.5.3), a sensitivity analysis was performed to determine which of the stochastic parameters influence 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 to 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 without site data, 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 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

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

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

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

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 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 and 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, were determined using Equation 6-1 and 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 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 and distributions assigned to all parameters for the sensitivity analyses and the basis for assigning such values and 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 Section 6.2.5.5 and the sensitivity analysis in Appendix 6H. The DCGL determination was performed using RESRAD-BUILD Version 3.21 and 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, were calculated using Equations 6-2 through 6-4 and are provided in Appendix 6J.

6.4.2 Structure Volumetric DCGL

Two methodologies were 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 who 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 anticipates that five structures will remain 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 known 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 six radionuclides in the analyses.

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 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.

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

6.4.3.3 Parameter Value Assignment

The total release from 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 that were 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:

- 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 wrre applied to obtain separate H-3 DCGL values for each case.

The first model, described in Section 6.4.4.2.1, uses modified RESRAD parameters to reflect a contaminated zone within the saturated zone (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 such a 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:

- 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 and distributions assigned to all parameters for the sensitivity analyses and the basis for assigning such values and 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 Section 6.2.5.5 and 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 using the input parameters from 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 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.

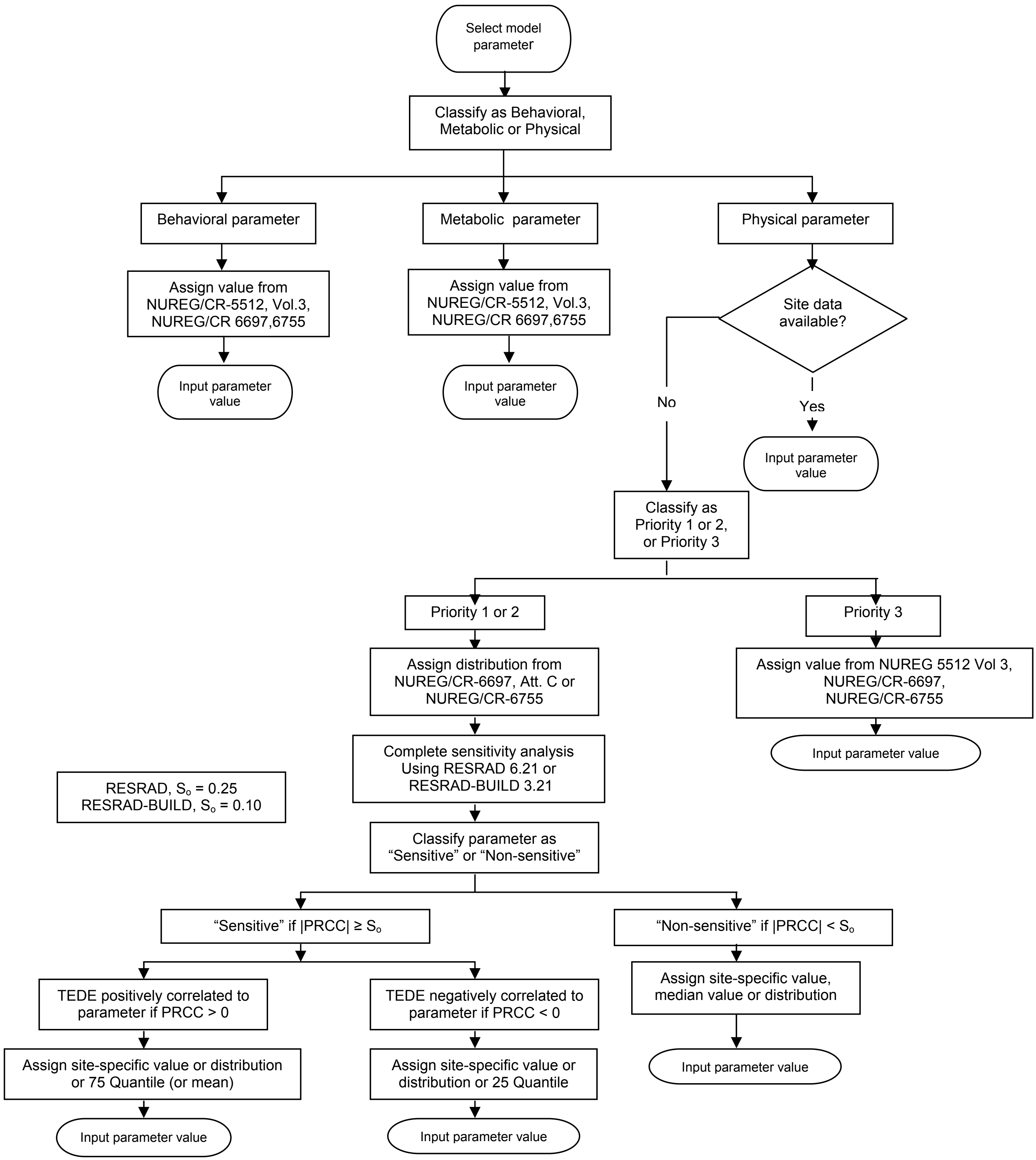
6-13 YA-CALC-00-003-04, “Assessment of radionuclide release from contaminated concrete at the Yankee Rowe Nuclear Power Plant,” dated August 2004.

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.

Figure 6-1
Parameter Selection Process



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Appendix 6A

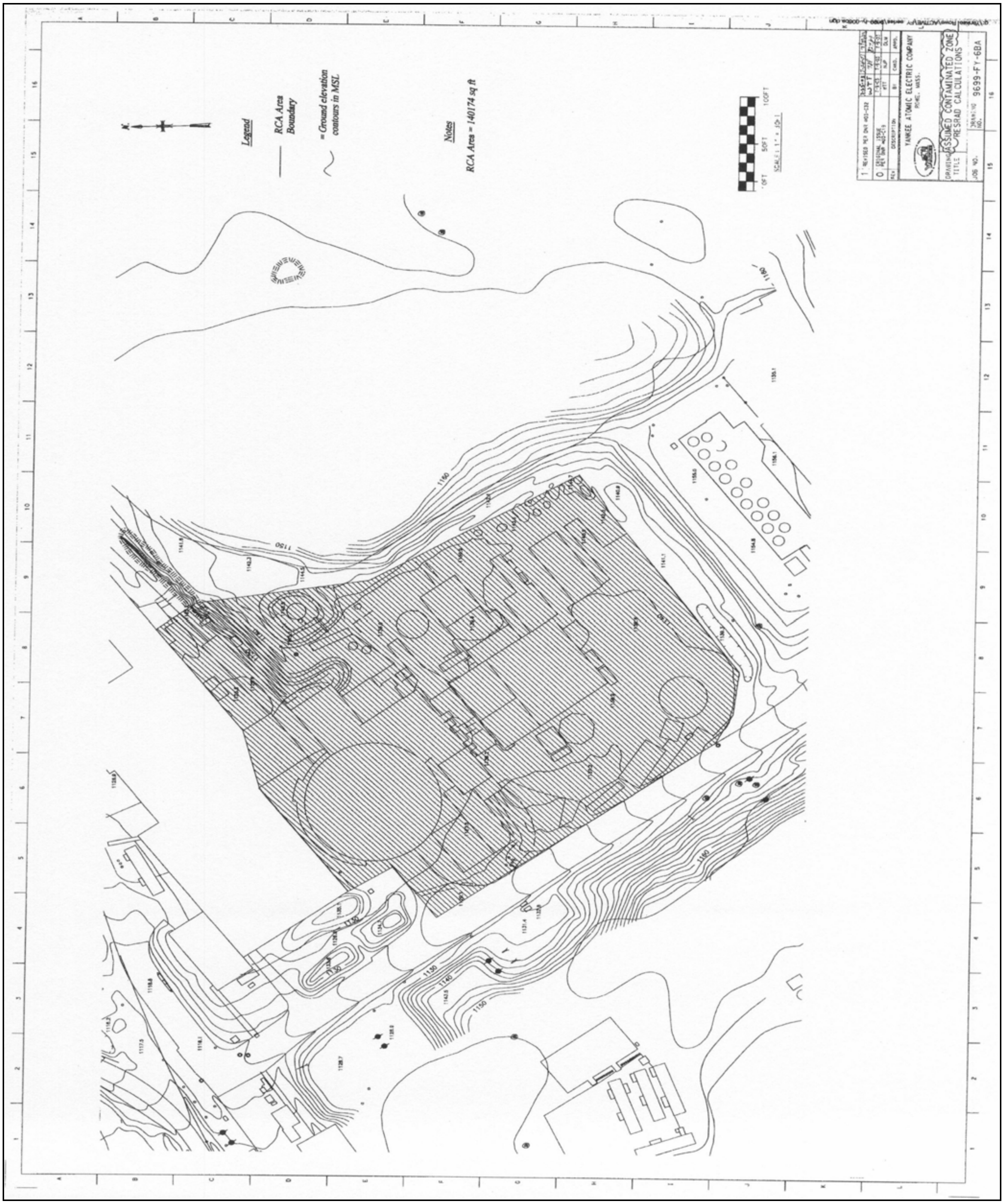
Basis Document for Site-Specific Parameter Value Assignment, Soil

Table of Contents

1. Area of the Contaminated Zone

Figure 1-1, (YR Drawing: 9699-FY-6BA Revision 1) was generated with AutoCAD Version 6. The area of the contaminated zone was drawn and calculated by the AutoCAD software and was found to be 140,174 ft^2. Converting this value into m^2: 140174 ft^2 x 9.29E-02 m^2/ft^2 = 13022m^2. Thus, the area of the contaminated zone was assigned a value of 13022m^2.

Figure 1-1


Legend
RCA Area Boundary
= Ground elevation contours in MSL
Notes
RCA Area = 140174 sq ft
YANKEE ATOMIC ELECTRIC COMPANY
ASSUMED CONTAMINATED ZONE RESRAD CALCULATIONS
9699-FY-6BA

2. Contaminated Zone Erosion Rate

The slope of the contaminated zone was determined from the Rowe Site Closure Base Map (YR Drawing: 9699-FY-6BB, Revision 0) generated by AutoCAD Map Version 6. A line was extended from the contour line near Monitoring Well CB-2 to the contour line near Monitoring Well CB-3. The distance between the two wells is 700 feet, with a decreasing change in elevation from 1140 to 1120 feet. Thus, the slope at the Rowe site is 20’ per 700’, which corresponds to a 2.86% slope.

The following YR site drawing provides a transferable scale to the Vapor Containment
Reference: VC Site Drawing Number 9699-FV-1a
Scale: Outer Diameter of VC sphere = 125'

Data from NUREG/CR-6697, Attachment C, Section 3.8, 2nd paragraph of the section labeled “Discussion” were used to select the appropriate Erosion Rate that corresponds to the Rowe Site slope of 2.86%.

Table 2-1 provides values for Erosion Rate (m/y) corresponding to different percent slopes. The value for Erosion Rate representative of row-crop agriculture and a 2% slope (from NUREG/CR-6697, Att. C) is 6.0 E-4 m/y. Erosion Rates were then calculated for 5%, 10% and 15% slopes using the rate increase factors specified in Section 3.8 of NUREG/CR-6697.

Table 2-1 Erosion Rate

Percent Slope	Erosion Rate (m/y)
2	6.0E-4
5	1.8E-3
10	4.2E-3
15	9.0E-3

Using this Erosion Rate/percent slope data, the value for Erosion Rate corresponding to 2.86% is 8.5E-04 m/yr.

3. Humidity in Air

"Regional and Site-Specific Absolute Humidity Data for Use in Tritium Dose Calculations", Health Physics, Vol.39, pp. 318-320, 1980, provides a table of absolute humidity for selected locations in the United States. These values were calculated from data from the National Oceanic and Atmospheric Administration, 1977, Climatological Data, Annual Summary, Volume 28(13), United States Department of Commerce.

The value of 6.1 g/m^3 was chosen for the RESRAD humidity parameter corresponding to the Northeast region in the vicinity of Albany, NY, approximately 70 miles west of the YNPS.

4. Average Annual Wind Speed

The wind speed and direction, joint frequency distributions from Table 3.3-2 of the YNPS Environmental Decommissioning Report, dated December 1993, (Table 4-2) were used to calculate the average annual wind speed. The mid-range value was calculated for each of the ranges for which data were available. An average wind speed was calculated by summing the product of the mid-range value for each range and the percentage of time the wind was recorded to be within the range. A value of 2.03 m/s was assigned to this parameter.

Table 4- 1 Wind Speed

Min. Wind Speed (mph)	Max. Wind Speed (mph)	Mid-Range Wind Speed (mph)	Percent of Time in Range	Mid-Range (weighted by percentage of time)
0.00	0.95	0.48	0.00	0.00
0.95	3.00	1.98	47.11	0.93
4.00	7.00	5.50	38.98	2.14
8.00	12.00	10.00	12.72	1.27
13.00	18.00	15.50	1.16	0.18
19.00	24.00	21.50	0.02	0.00
			Average:	4.53 mph

Converting to m/s: 4.53 mi/h x 1 m/6.214-04 mi x 1h/3600 s = 2.03 m/s

Table 4- 2
Joint Frequency Distribution Table Produced by YAEC METROSE Computer Code Using Meteorological Data Collected at Yankee Nuclear Power Station Met Tower

YNPS 35-Foot
Wind Speed and Direction Joint Frequency Distributions
1988-1992

WIND DIRECTION FROM

SPEED(MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	VRBL	TOTAL
CALM	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
(1)	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00
(2)	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00
C-3	570	1010	1351	1672	2941	3952	2556	1741	1284	942	763	385	283	221	169	297	0	20137
(1)	1.33	2.36	3.16	3.91	6.88	9.25	5.98	4.07	3.00	2.20	1.78	.90	.66	.52	.40	.69	.00	47.11
(2)	1.33	2.36	3.16	3.91	6.88	9.25	5.98	4.07	3.00	2.20	1.78	.90	.66	.52	.40	.69	.00	47.11
4-7	1468	1363	975	727	761	374	455	773	1230	2046	2570	1176	773	565	625	783	0	16664
(1)	3.43	3.19	2.28	1.70	1.78	.87	1.06	1.81	2.88	4.79	6.01	2.75	1.81	1.32	1.46	1.83	.00	38.98
(2)	3.43	3.19	2.28	1.70	1.78	.87	1.06	1.81	2.88	4.79	6.01	2.75	1.81	1.32	1.46	1.83	.00	38.98
8-12	1137	952	263	35	4	2	2	13	96	483	1159	481	179	137	188	308	0	5439
(1)	2.66	2.23	.62	.08	.01	.00	.00	.03	.22	1.13	2.71	1.13	.42	.32	.44	.72	.00	12.72
(2)	2.66	2.23	.62	.08	.01	.00	.00	.03	.22	1.13	2.71	1.13	.42	.32	.44	.72	.00	12.72
13-18	141	195	12	1	0	0	0	0	1	12	116	9	1	1	0	7	0	496
(1)	.33	.46	.03	.00	.00	.00	.00	.00	.00	.03	.27	.02	.00	.00	.00	.02	.00	1.16
(2)	.33	.46	.03	.00	.00	.00	.00	.00	.00	.03	.27	.02	.00	.00	.00	.02	.00	1.16
19-24	2	5	1	0	0	0	0	0	0	0	2	0	0	0	0	0	0	10
(1)	.00	.01	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.02
(2)	.00	.01	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.02
GT 24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
(1)	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00
(2)	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00	.00
ALL SPEEDS	3318	3525	2602	2435	3706	4328	3013	2527	2611	3483	4610	2051	1236	924	982	1395	0	42746
(1)	7.76	8.25	6.09	5.70	8.67	10.12	7.05	5.91	6.11	8.15	10.78	4.80	2.89	2.16	2.30	3.26	.00	100.00
(2)	7.76	8.25	6.09	5.70	8.67	10.12	7.05	5.91	6.11	8.15	10.78	4.80	2.89	2.16	2.30	3.26	.00	100.00

(1)=PERCENT OF ALL GOOD OBSERVATIONS FOR THIS PAGE
(2)=PERCENT OF ALL GOOD OBSERVATIONS FOR THIS PERIOD

C= CALM (WIND SPEED LESS THAN OR EQUAL TO .95 MPH)

5. Precipitation

Table 3.3-4 of the YNPS Environmental Decommissioning Report (December 1993) provides monthly mean precipitation totals for Readsboro, Vt., located approximately 5 miles from the YNPS. This information is based on data from the National Oceanic and Atmospheric Administration.

Table 5- 1 Mean Precipitation Rate

Readsboro Monthly Mean Precipitation Totals (inches of water)	
Period:1961-1990	
Month	Precipitation (inches)
Jan	3.49
Feb	3.43
Mar	3.86
April	4.32
May	4.59
Jun	4.54
Jul	4.08
Aug	4.29
Sept	3.79
Oct	3.8
Nov	4.61
Dec	4.28
Year Total	49.08

Converting to meters/year: 49.08 in/y x 2.54 cm/in x m/100 cm = 1.2 m/y. The precipitation rate was assigned a value of 1.2 m/y.

6. Irrigation Rate (Evapotranspiration and Runoff Coefficients)

NUREG/CR-6697 Attachment C, Section 4.3 discusses the Irrigation Rate in terms of the Evapotranspiration Coefficient. Equation 4.3-1 of NUREG/CR-6697 expresses the Evapotranspiration Coefficient as:

Where: ETr = the Evapotranspiration Rate (m/y)
Pr = the Precipitation Rate (m/y)
IRr = the Irrigation Rate (m/y) and
Cr = the Runoff Coefficient.

Based upon this equation, the Irrigation Rate can be expressed as:

$$IRr = \frac{ETr}{Ce} - (1-Cr)(Pr)$$

The input values for the variables in this equation:

- YA-REPT-00-002-03 (Ref. 1) cites a value for the average annual Evapotranspiration Rate, ETr, in the upper Housatonic River basin of 21.6 in/y or 0.549 m/y from 1931 to 1960.
- The Precipitation Rate, Pr, has been assigned a site-specific value of 1.2 m/y as discussed in Section 5 of this Attachment.
- Appendix E, Table E.1 of Ref. 2 provides the equation below to calculate the Runoff Coefficient, C_r, for an agricultural environment. Table E.1, Runoff Coefficient Values, also lists values for c_1, c_2 and c_3 for various environments:

$$C_r = 1 - c_1 - c_2 - c_3$$

c_1 = 0.1 for hilly land with an average slopes of 46 m/mi (Refer to section 2 of this Attachment for the site slope determination- 20' drop per 700' run or 46 m/mi).
c_2 = 0.2 for intermediate combinations of clay and loam as identified at the site in Ref. 3.
c_3 = 0.1 for cultivated lands which also fits the scenario for the site.

$$Cr = 1 - 0.1 - 0.2 - 0.1 = 0.6$$

- NUREG/CR-6697, Attachment C, Section 4.3-Evapotranspiration Coefficient, Ce, defines this parameter as the ratio of the total volume of water (a combination of evaporation from soil surfaces and transpiration from vegetation) transferred to the atmosphere to the total volume of water available within the root zone of the soil. The NUREG/CR recommends the use of a uniform distribution with minimum and maximum values of 0.5 and 0.75, respectively and with 0.625 as median. Any selected value for the irrigation rate should satisfy the Ce minimum to maximum range.

Substituting the minimum and maximum values of Ce into Equation 4.3-1 results in the following range for the Irrigation Rate, IRr:

Table 6- 1 Irrigation Rate

Variable	"Min" Value	"Max" Value	Units
ETr	0.549	0.549	m/y
Pr	1.2	1.2	m/y
Cr	0.6	0.6	--
Ce	0.5	0.75	--
IRr	0.252	0.618	m/y

Based on the calculated minimum and maximum IRr values, the median value is 0.435 m/y. A uniform distribution was assigned to this parameter and a positive input correlation to the Well Pumping Rate was assigned based upon guidance in NUREG/CR-6697 and NUREG/CR-6676.

7. Field Capacity: Contaminated Zone, Unsaturated Zone and Saturated Zone

The "Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil," (Ref. 3) defines the relationship of field capacity (residual water content) to effective porosity. The field capacity is the ratio of the volume of water retained in the soil sample, after all drainage has ceased, to the total volume of the soil sample. Equation 4.4 of Ref. 3 relates Total and Effective Porosity to Field Capacity as follows:

Effective Porosity = Total Porosity - Field Capacity

Thus,

Field Capacity = Total Porosity - Effective Porosity

The total and effective porosity values for the various zones are the mean values of the NUREG/CR-6697 distributions for sand.

Table 7- 1 Field Capacity

Zone / Soil Type	Total Porosity	Effective Porosity	Field Capacity
Contaminated/sand	0.43	0.383	0.047
Unsaturated/sand	0.43	0.383	0.047
Saturated/sand	0.43	0.383	0.047

8. Saturated Zone Hydraulic Gradient

NUREG/CR-6697, Attachment C, Section 3.6, discusses this parameter's use in the determination of the groundwater flow rate, which effects the rise time and the dilution of radionuclides in the well water. The method for calculating the hydraulic gradient is given in NUREG/CR-6697, Attachment C, Equation 3.6-1:

$$J_x = \frac{h_1 - h_2}{\Delta_x}$$

Where h_1 and h_2 represent the hydraulic heads or the water level elevations at location 1 and 2, and Δ_x is the distance between the two locations. The water level elevations are referenced to mean sea level, msl. This methodology was followed in YA -REPT-00-002-03, "Hydrogeological Parameter Estimates for Radiation Dose Modeling" (Ref. 1) to determine the average hydraulic gradient across the site. An average value for the site was calculated from three separate hydraulic gradient determinations as follows in Table 8.1. LTP Figure 2-8 shows the well locations.

Table 8- 1 Hydraulic Gradient

Well / Location Designation	Water Level Elevation (msl), ft	Distance Between Wells, ft	Hydraulic Gradient, ft/ft
CB-3	1135		
Deerfield River below Sherman Dam	1020	1000	(1135-1020)/1000 = 0.115
CB-3	1135		
CB-2	1105	533	(1135-1105)/533 = 0.056
CW-3	1132		
CB-1	1114	118	(1132-1114)/118 = 0.152
			Average = 0.1

The hydraulic gradient was assigned a value of 0.1 feet/foot.

9. Well Pumping Rate

NUREG/CR-6697, Attachment C Section 3.10 states that "a site-specific input distribution for well pumping rate can be determined as the sum of individual water needs." The household use component is calculated from the Domestic Water Use discussed in YA-REPT-00-002-03, Ref. 1.

Based upon the most recent data available, (1997 Census of Agriculture Volume 1: Part 21, Chapter 1, Massachusetts State-Level Data), irrigation of pastureland is not a common practice in Massachusetts. This data indicates that in Massachusetts, while 24,269 total acres of crop land were irrigated, only 295 total acres of pastureland were irrigated. Furthermore, only one farm, in Franklin County, claimed irrigation of pasture.

Table 9-1 Water Use Components Used to Determine Well Pumping Rate

Water Use Components for a Family of Four	Median	Minimum	Maximum	Units
Household*	**374**	**374**	**374**	m^3/y
Livestock	**76.7**	**76.7**	**76.7**	m^3/y
Irrigation of vegetable plot				
Contaminated fraction f_p = min (Area/2000, 0.5)	1	1	1	
Irrigation rate I_r (m/y)	0.435	0.252	0.618	
Irrigation water $f_p \times l_r \times 2000$	**870**	**504**	**1236**	m^3/y
Irrigation of pasture (Not a New England practice.)				
Contaminated fraction f_m = Area/20,000 <= 1.	1	1	1	
Irrigation rate I_r (m/y)	0	0	0	m/y
Irrigation water $f_m \times l_r \times 20{,}000$	**0**	**0**	**0**	m^3/y
Drinking water **	**1.91**	**1.91**	**1.91**	m^3/y
TOTAL FOR A FAMILY OF FOUR				
(sum of water components in **Bold** type)	**1323**	**957**	**1689**	m^3/y

A uniform distribution was assigned to this parameter with a positive correlation to the Irrigation Rate.

* Household Use: Domestic Water Use for family of four of 272 gallons per day (Ref. 1) minus the drinking water component of $1.91 m^3/y$.

** 478 l/y per individual adjusted to family of four and converted to m^3/y.

conversion: 478 l/y-Ind x 4 Ind x $1m^3/1000l = 1.91 m^3/y$

272 gal/day x 3.79E-3 gal/m^3 x 365.25day/y = 376 m^3/y

10. Watershed for Nearby Stream or Pond

The following figure is taken from a letter to the USNRC from Yankee Atomic Electric Co., FYR 82-59, June 16, 1982, that delineates the watersheds to Wheeler Brook and to the site. An evaluation of this topographic map and the drainage areas is also included in Ref. 1.

The watershed area to the site is 0.3 square miles. Converting to square meters yields a total watershed area of 0.3 mi^2 x $(1609.3\ m/mi)^2$ = 7.77E+05 m^2. This parameter was assigned a value of 7.77E+05 m^2.

Figure 10- 1 Local Site Drainage Area

Attachment J, Page 2 of 4

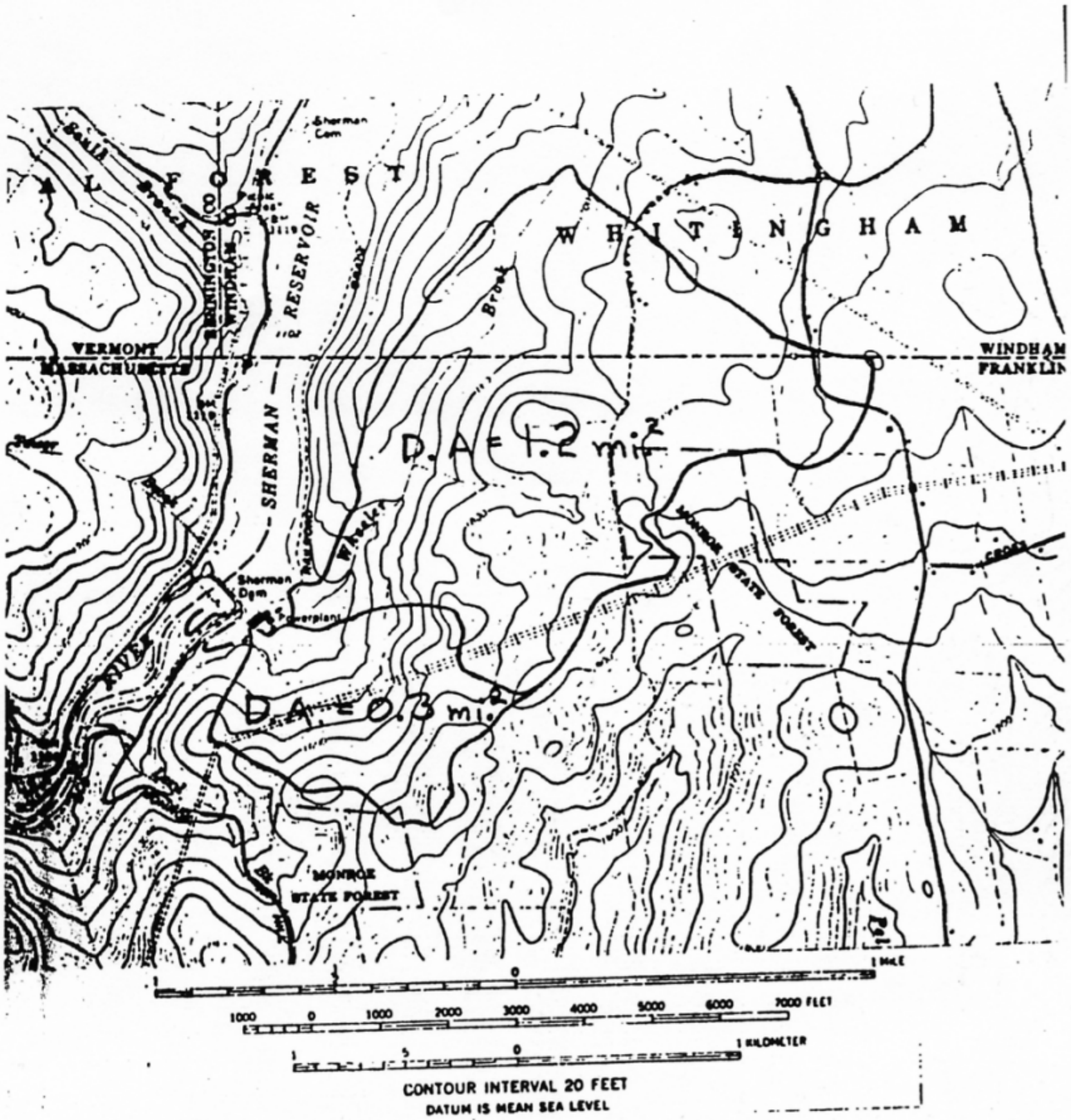


FIGURE 1

LOCAL SITE DRAINAGE AREAS

References:

1. YA-REPT-00-002-03, "Hydrogeological Parameter Estimates for Radiation Dose Modeling," May 2003.

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

3. Yu, C., et al., Argonne National Laboratory, “Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil,” April 1993.

Appendix 6B

Input Parameter Values for Sensitivity Analysis, Soil

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Soil Concentrations										
Basic radiation dose limit (mrem/yr)	P	3	D	25	10 CFR 20.1402 (Ref. 1)	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Assumed unit concentration	NR	NR	NR	NR	
Distribution coefficients (contam., unsat. and sat. zones) (cm^3/g)										
Ac-227+progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Ag-108m	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.38	2.1	0.001	0.999	216
Am-241	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.28	3.15	0.001	0.999	1445
Am-243+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.28	3.15	0.001	0.999	1445
C-14	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.4	3.22	0.001	0.999	11
Cm-243	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.82	1.82	0.001	0.999	6761
Co-60	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.46	2.53	0.001	0.999	235
Cs-134	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.1	2.33	0.001	0.999	446
Cs-137+progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.1	2.33	0.001	0.999	446
Eu-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Eu-154	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Eu-155	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Fe-55	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.34	2.67	0.001	0.999	209
Gd-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
H-3	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-2.81	0.5	0.001	0.999	0.06
Nb-94	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.94	3.22	0.001	0.999	380
Ni-63	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.05	1.46	0.001	0.999	424
Np-237+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.84	2.25	0.001	0.999	17
Pa-231	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.94	3.22	0.001	0.999	380
Pb-210+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.78	2.76	0.001	0.999	2392
Pu-238	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.86	1.89	0.001	0.999	953
Pu-239	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.86	1.89	0.001	0.999	953
Pu-241+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.86	1.89	0.001	0.999	953
Ra-226+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.17	1.7	0.001	0.999	3533
Sb-125	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.94	3.22	0.001	0.999	380
Sr-90+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.45	2.12	0.001	0.999	32
Tc-99	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-0.67	3.16	0.001	0.999	0.51
Th-229+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.68	3.62	0.001	0.999	5884
Th-230	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.68	3.62	0.001	0.999	5884
U-233	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.84	3.13	0.001	0.999	126
U-234	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.84	3.13	0.001	0.999	126

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
U-235+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.84	3.13	0.001	0.999	126
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	Ground water assumed 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	
Contaminated Zone										
Area of contaminated zone (m^2)	P	2	D	13022	Site-specific- radiation control area (LTP App. 6A, Section 1)	NR	NR	NR	NR	
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) (Ref. 3)	0.15	3.8	NR	NR	1.975
Length parallel to aquifer flow (m)	P	2	D	129	Site-specific - diameter of circle with an area of 13022 m^2 (LTP App. 6A, Section 1)	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	0	No cover assumed	NR	NR	NR	NR	
Density of contaminated zone (g/cm^3)	P	1	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	1.5105	0.159	1.019	2.002	1.5105
Contaminated zone erosion rate (m/yr)	P	2	D	8.5E-04	Calculated value based on site-specific slope of 2.9% (LTP App. 6A, Section 2)	NR	NR	NR	NR	
Contaminated zone total porosity	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	0.43	0.06	0.2446	0.6154	0.43
Contaminated zone field capacity	P	3	D	0.05	Site-specific value (LTP App. 6A, Section 7) calculated using Equation 4.4 from Ref. 4	NR	NR	NR	NR	0.05

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Contaminated zone hydraulic conductivity (m/yr)	P	2	S	Beta	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	110	5870	1.398	1.842	2506
Contaminated zone b parameter	P	2	S	Bounded Lognormal-N	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	- 0.0253	0.216	0.501	1.90	0.975
Humidity in air (g/m^3)	P	3	D	6.1	Regional value. (LTP App. 6A, Section 3).	NR	NR	NR	NR	
Evapotranspiration coefficient	P	2	S	Uniform	NUREG/CR-6697 Att. C, (Ref. 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 (LTP App. 6A, Section 4)	NR	NR	NR	NR	
Precipitation (m/yr)	P	2	D	1.2	Site-specific value calculated from site geographical area precip. (LTP App. 6A, Section 5)	NR	NR	NR	NR	
Irrigation (m/yr)	B	3	S	Uniform	NUREG/CR-6697, Att. C methodology(Ref. 2, Att. L)	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 (Ref. 2, App. 6A, Section 6)	NR	NR	NR	NR	
Watershed area for nearby stream or pond (m^2)	P	3	D	7.77E+05	Site-specific- drainage area (LTP App. 6A, Section 10)	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	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	1.5105	0.159	1.019	2.002	1.5105
Saturated zone total porosity	P	1	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	0.43	0.06	0.2446	0.6154	0.43
Saturated zone effective porosity	P	1	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	0.383	0.0610	0.195	0.572	0.383

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Saturated zone field capacity	P	3	D	0.05	Site-specific value calculated using Equation 4.4 from Ref.4 (LTP App. 6A, Section 7)	NR	NR	NR	NR	0.05
Saturated zone hydraulic conductivity (m/yr)	P	1	S	Beta	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	110	5870	1.398	1.842	2506
Saturated zone hydraulic gradient	P	2	D	0.1	Site gradient (LTP App. 6A, Section 8)	NR	NR	NR	NR	
Saturated zone b parameter	P	2	S	Bounded Lognormal-N	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	- 0.0253	0.216	0.501	1.90	0.975
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	S	Triangular	NUREG/CR-6697, Att. C (Ref. 2)	6	10	30		14.51
Model: Nondispersion (ND) or Mass-Balance (MB)	P	3	D	ND	ND model recommended for contaminant areas > 1,000 m^2 (Ref. 4)	NR	NR	NR	NR	
Well pumping rate (m^3/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. (Ref. 2 and LTP App. 6A, Section 9)	957	1689	NR	NR	1323
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	3	D	1	Site-specific value	NR	NR	NR	NR	
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 (Ref. 3)	0.01	3.65			1.82
Unsat. zone 1, soil density (g/cm^3)	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	1.5105	0.159	1.019	2.002	1.5105
Unsat. zone 1, total porosity	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	0.43	0.06	0.2446	0.6154	0.43
Unsat. zone 1, effective porosity	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	0.383	0.0610	0.195	0.572	0.383

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Unsat. zone 1, field capacity	P	3	D	0.05	Site-specific value calculated using Equation 4.4 from Ref. 4 (LTP App. 6A, Section 7)	NR	NR	NR	NR	0.05
Unsat. zone 1, hydraulic conductivity (m/yr)	P	2	S	Beta	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	110	5870	1.398	1.842	2506
Unsat. zone 1, soil-specific b parameter	P	2	S	Bounded Lognormal-N	NUREG/CR-6697 dist. for site soil type: sand (Ref. 3)	- 0.0253	0.216	0.501	1.90	0.975
Occupancy										
Inhalation rate (m^3/yr)	B	3	D	8400	NUREG/CR-6697, Att. C (Ref. 2)	NR	NR	NR	NR	
Mass loading for inhalation (g/m^3)	P	2	S	Continuous Linear	NUREG/CR-6697, Att. C (Ref. 2)	NR	NR	NR	NR	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 (Ref. 2)	0.15	0.95	NR	NR	0.55
Shielding factor, external gamma	P	2	S	Bounded Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 5)	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) (Ref. 5)	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) (Ref. 5)	NR	NR	NR	NR	
Leafy vegetable consumption (kg/yr)	B	3	D	21.4	NUREG/CR-5512, Vol. 3 (Ref. 5)	NR	NR	NR	NR	
Milk consumption (L/yr)	B	2	D	233	NUREG/CR-5512, Vol. 3 (Ref. 5)	NR	NR	NR	NR	
Meat and poultry consumption (kg/yr)	B	3	D	65.1	NUREG/CR5512, Vol. 3 (beef + poultry) (Ref. 5)	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Fish consumption (kg/yr)	B	3	D	20.6	NUREG/CR-5512, Vol. 3 (Ref. 5)	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 (Ref. 5)	NR	NR	NR	NR	
Drinking water intake (L/yr)	B	2	D	478.5	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	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 (Ref. 5)	NR	NR	NR	NR	
Contamination fraction of plant food	P	3	D	1	NUREG/CR-5512, Vol. 3 (Ref. 5)	NR	NR	NR	NR	
Contamination fraction of meat	P	3	D	1	NUREG/CR-5512, Vol. 3 (Ref. 5)	NR	NR	NR	NR	
Contamination fraction of milk	P	3	D	1	NUREG/CR-5512, Vol. 3 (Ref. 5)	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 (Ref. 5)	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 (Ref. 5)	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Livestock water intake for meat (L/day)	M	3	D	50.6	NUREG/CR5512, Vol. 3 Table 6.87, beef cattle + poultry + layer hen (Ref. 5)	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87 (Ref. 5)	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 (Ref. 5)	NR	NR	NR	NR	
Depth of soil mixing layer (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 2)	0	0.15	0.6	NR	0.23
Depth of roots (m)	P	1	S	Uniform	Min. from NUREG/CR-6697, Att. C (Ref. 2) Max. is site specific depth to water table (Ref. 3)	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	
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 (Ref. 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 (Ref. 5)	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 (Ref. 5)	NR	NR	NR	NR	
Growing Season for Non-Leafy (years)	P	3	D	0.246	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Growing Season for Leafy (years)	P	3	D	0.123	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Growing Season for Fodder (years)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/yr)	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 5)	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 2)	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 (Ref. 5)	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 (Ref. 5)	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 (Ref. 5)	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 (Ref. 5)	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 5)	NR	NR	NR	NR	
Meat and poultry	B	3	D	20	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef) (Ref. 5)	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	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
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/cm3)	P	3	D	2.00E-05	RESRAD Default	NR	NR	NR	NR	
C-12 concentration in contaminated soil (g of C-12/g of soil)	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 (Ref. 2)	0.2	0.3	0.6	NR	0.3
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 (Ref. 2)	NR	NR	NR	NR	
Fraction of grain in milk cow feed	B	3	D	0.1000	NUREG/CR-6697, Att. B (Ref. 2)	NR	NR	NR	NR	
Dose Conversion Factors (Inhalation mrem/pCi)										
Ac-227+ progeny	M	3	D	6.72E+00	FGR-11 (Ref. 6) (RESRAD Dose Conversion Library)	NR	NR	NR	NR	
Ag-108m	M	3	D	2.83E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Am-241	M	3	D	4.44E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Am-243+ progeny	M	3	D	4.40E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
C-14	M	3	D	2.09E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Cm-243	M	3	D	3.07E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Co-60	M	3	D	2.19E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Cs-134	M	3	D	4.63E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Cs-137+ progeny	M	3	D	3.19E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Eu-152	M	3	D	2.21E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Eu-154	M	3	D	2.86E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Eu-155	M	3	D	4.14E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Fe-55	M	3	D	2.69E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Gd-152	M	3	D	2.43E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Nb-94	M	3	D	4.14E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Ni-63	M	3	D	6.29E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Np-237+ progeny	M	3	D	5.40E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pa-231	M	3	D	1.28E+00	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pb-210+ progeny	M	3	D	1.38E-02	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pu-238	M	3	D	3.92E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pu-239	M	3	D	4.29E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pu-241+ progeny	M	3	D	8.25E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Ra-226+ progeny	M	3	D	8.60E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Sb-125	M	3	D	1.22E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Sr-90+ progeny	M	3	D	1.31E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Tc-99	M	3	D	8.33E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Th-229+ progeny	M	3	D	2.16E+00	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Th-230	M	3	D	3.26E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
U-233	M	3	D	1.35E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
U-234	M	3	D	1.32E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
U-235+ progeny	M	3	D	1.23E-01	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Dose Conversion Factors (Ingestion mrem/pCi)										
Ac-227+ progeny	M	3	D	1.48E-02	FGR-11 (Ref. 6) (RESRAD Dose Conversion Library)	NR	NR	NR	NR	
Ag-108m	M	3	D	7.62E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Am-241	M	3	D	3.64E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Am-243+ progeny	M	3	D	3.63E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
C-14	M	3	D	2.09E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Cm-243	M	3	D	2.51E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil

Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Co-60	M	3	D	2.69E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Cs-137+ progeny	M	3	D	5.00E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Eu-152	M	3	D	6.48E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Eu-154	M	3	D	9.55E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Eu-155	M	3	D	1.53E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Fe-55	M	3	D	6.07E-07	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Gd-152	M	3	D	1.61E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Nb-94	M	3	D	7.14E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Np-237+ progeny	M	3	D	4.44E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pa-231	M	3	D	1.06E-02	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pb-210+ progeny	M	3	D	5.37E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pu-238	M	3	D	3.20E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pu-239	M	3	D	3.54E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Pu-241+ progeny	M	3	D	6.85E-05	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Ra-226+ progeny	M	3	D	1.33E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Sb-125	M	3	D	2.81E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Sr-90+ progeny	M	3	D	1.53E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Tc-99	M	3	D	1.46E-06	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Th-229+ progeny	M	3	D	4.03E-03	FGR-11 (Ref. 6)	NR	NR	NR	NR	
Th-230	M	3	D	5.48E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
U-233	M	3	D	2.89E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
U-234	M	3	D	2.83E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	
U-235+ progeny	M	3	D	2.67E-04	FGR-11 (Ref. 6)	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Plant Transfer Factors (pCi/g plant)/(pCi/g soil)										
Ac-227+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	1.1	0.001	0.999	1.0E-03
Ag-108m	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-5.52	0.9	0.001	0.999	4.0E-03
Am-241	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Am-243+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
C-14	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-0.36	0.9	0.001	0.999	7.0E-01
Cm-243	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Co-60	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-2.53	0.9	0.001	0.999	8.0E-02
Cs-134	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.22	1.0	0.001	0.999	4.0E-02
Cs-137+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.22	1.0	0.001	0.999	4.0E-02
Eu-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	2.0E-03
Eu-154	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	2.0E-03
Eu-155	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	2.0E-03
Fe-55	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Gd-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	2.0E-03
H-3	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	1.57	1.1	0.001	0.999	4.8E+00
Nb-94	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-4.61	1.1	0.001	0.999	1.0E-02
Ni-63	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.0	0.9	0.001	0.999	5.0E-02

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Np-237+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.91	0.9	0.001	0.999	2.0E-02
Pa-231	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-4.61	1.1	0.001	0.999	1.0E-02
Pb-210+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-5.52	0.9	0.001	0.999	4.0E-03
Pu-238	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Pu-239	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Pu-241+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Ra-226+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.22	0.9	0.001	0.999	4.0E-02
Sb-125	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-4.61	1.0	0.001	0.999	1.0E-02
Sr-90+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-1.20	1.0	0.001	0.999	3.0E-01
Tc-99	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	1.61	0.9	0.001	0.999	5.0E+00
Th-229+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Th-230	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
U-233	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.9	0.001	0.999	2.0E-03
U-234	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.9	0.001	0.999	2.0E-03
U-235+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.9	0.001	0.999	2.0E-03
Meat Transfer Factors (pCi/kg)/(pCi/d)										
Ac-227+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-6.21	0.7	0.001	0.999	2.0E-03

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Am-241	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-9.90	0.2	0.001	0.999	5.0E-05
Am-243+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-3.00	0.4	0.001	0.999	5.0E-02
Cs-137+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-6.21	1.0	0.001	0.999	2.0E-03
Eu-155	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-5.30	0.9	0.001	0.999	5.0E-03
Np-237+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-12.21	1.0	0.001	0.999	5.0E-06
Pb-210+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-7.13	0.7	0.001	0.999	8.0E-04

Input Parameters for Sensitivity Analysis, Soil

Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Pu-238	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-9.21	0.2	0.001	0.999	1.0E-04
Pu-241+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-9.21	0.2	0.001	0.999	1.0E-04
Ra-226+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-6.91	0.9	0.001	0.999	1.0E-03
Sr-90+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-9.21	0.7	0.001	0.999	1.0E-04
Th-229+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-7.13	0.7	0.001	0.999	8.0E-04
U-235+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-7.13	0.7	0.001	0.999	8.0E-04
Milk Transfer Factors (pCi/l)/(pCi/d)										
Ac-227+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-13.12	0.7	0.001	0.999	2.0E-06
Am-243+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-4.4	0.9	0.001	0.999	1.2E-02

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Cm-243	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-4.61	0.5	0.001	0.999	1.0E-02
Cs-137+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-9.72	0.9	0.001	0.999	6.0E-05
Eu-154	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-3.91	0.7	0.001	0.999	2.0E-02
Np-237+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-12.21	0.9	0.001	0.999	5.0E-06
Pb-210+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-13.82	0.5	0.001	0.999	1.0E-06
Pu-241+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-13.82	0.5	0.001	0.999	1.0E-06

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Ra-226+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-9.72	0.9	0.001	0.999	6.0E-05
Sr-90+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.5	0.001	0.999	2.0E-03
Tc-99	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.7	0.001	0.999	1.0E-03
Th-229+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-7.82	0.6	0.001	0.999	4.0E-04
U-235+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-7.82	0.6	0.001	0.999	4.0E-04
Bioaccumulation Factors for Fish ((pCi/kg)/(pCi/l))										
Ac-227+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.7	1.1	NR	NR	1.5E+01
Ag-108m	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	1.6	1.1	NR	NR	5.0E+00
Am-241	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Am-243+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
C-14	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	10.8	1.1	NR	NR	4.9E+04
Cm-243	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Co-60	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.7	1.1	NR	NR	3.0E+02
Cs-134	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.6	0.7	NR	NR	2.0E+03

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Cs-137+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.6	0.7	NR	NR	2.0E+03
Eu-152	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Eu-154	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Eu-155	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Fe-55	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.3	1.1	NR	NR	2.0E+02
Gd-152	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.2	1.1	NR	NR	2.5E+01
H-3	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	0	0.1	NR	NR	1.0E+00
Nb-94	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.7	1.1	NR	NR	3.0E+02
Ni-63	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
Np-237+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Pa-231	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
Pb-210+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.7	1.1	NR	NR	3.0E+02
Pu-238	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Pu-239	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Pu-241+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Ra-226+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Sb-125	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
Sr-90+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.1	1.1	NR	NR	6.0E+01

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Tc-99	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3	1.1	NR	NR	2.0E+01
Th-229+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
Th-230	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
U-233	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
U-234	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
U-235+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks ((pCi/kg)/(pCi/l))										
Ac-227+ progeny	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+ progeny	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+ progeny	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
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+ progeny	P	3	D	4.00E+02	RESRAD Default	NR	NR	NR	NR	

Input Parameters for Sensitivity Analysis, Soil
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Median
						1	2	3	4	
Pa-231	P	3	D	1.10E+02	RESRAD Default	NR	NR	NR	NR	
Pb-210+ progeny	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+ progeny	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Ra-226+ progeny	P	3	D	2.50E+02	RESRAD Default	NR	NR	NR	NR	
Sr-90+ progeny	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+ progeny	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+ progeny	P	3	D	6.00E+01	RESRAD Default	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	
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
NR = Not required

Additional Sensitivity Analysis Data:
Sampling Technique = Latin Hypercube
Number of observations =2000
Number of repetitions = 1
Input Rank Correlation Coefficients:

Thickness of contaminated zone and unsaturated zone = - 0.99
Total porosity and bulk density = - 0.99 (contaminated zone, unsaturated and saturated zones)
Total porosity and effective porosity = 0.96 (unsaturated and saturated zones)
Effective porosity and bulk density = -0.99 (unsaturated and saturated zones)
Well Pumping Rate and Irrigation Rate = 0.96

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-00-008-03, “Evaluation of GeoTesting Express Soil Testing and Determination of Depth to Groundwater,” December 2003

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. NUREG/CR-5512, Volume 3, “Residual Radioactive Contamination From Decommissioning: Parameter Analysis, Draft Report for Comment,” October 1999.

6. 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 (FGR-11), U.S. EPA, 1988.

Appendix 6C

Results of Sensitivity Analysis, Soil

Results of Sensitivity Analysis, Soil (Based on the Partial Rank Correlation Coefficient [PRCC])			
Radionuclide	**Sensitive Parameter**	**Units**	**PRCC**
H-3	Depth of roots	m	-0.59
	Kd of H-3 in contaminated zone	cm^3/g	-0.54
	Thickness of contaminated zone	m	0.45
C-14	Depth of roots	m	-0.59
	Thickness of contaminated zone	m	0.48
	Thickness of evasion layer of C	m	0.35
Fe-55	Meat transfer factor for Fe	pCi/kg per pCi/day	0.92
	Plant transfer factor for Fe	pCi/g plant per pCi/g soil	0.68
Ni-63	Plant transfer factor for Ni	pCi/g plant per pCi/g soil	0.90
	Milk transfer factor for Ni	pCi/l per pCi/day	0.80
	Depth of roots	m	-0.49
Co-60	External gamma shielding factor	Unit-less	0.95
	Plant transfer factor for Co	pCi/g plant per pCi/g soil	0.67
	Meat transfer factor for Co	pCi/g plant per pCi/day	0.36
Sr-90	Plant transfer factor for Sr	pCi/g plant per pCi/g soil	0.93
	Depth of roots	m	-0.53
Nb-94	External gamma shielding factor	Unit-less	0.99
	Kd of Nb in contaminated zone	cm^3/g	0.30
Tc-99	Plant transfer factor for Tc	pCi/g plant per pCi/g soil	0.88
	Depth of roots	m	-0.44
	Kd of Tc in contaminated zone	cm^3/g	0.49
Ag-108m	External gamma shielding factor	Unit-less	1.00
Sb-125	External gamma shielding factor	Unit-less	0.99
	Kd of Sb in contaminated zone	cm^3 /g	0.29
Cs-134	External gamma shielding factor	Unit-less	0.84
	Plant transfer factor for Cs	pCi/g plant per pCi/g soil	0.84
	Depth of roots	m	-0.33
	Milk transfer factor for Cs	pCi/l per pCi/day	0.32
	Meat transfer factor for Cs	pCi/kg per pCi/day	0.25

Results of Sensitivity Analysis, Soil (Based on the Partial Rank Correlation Coefficient [PRCC])			
Radionuclide	**Sensitive Parameter**	**Units**	**PRCC**
Cs-137	Plant transfer factor for Cs	pCi/g plant per pCi/g soil	0.88
	External gamma shielding factor	Unit-less	0.75
	Depth of roots	m	-0.39
	Milk transfer factor for Cs	pCi/l per pCi/day	0.39
	Meat transfer factor for Cs	pCi/kg per pCi/day	0.31
Eu-152	External gamma shielding factor	Unit-less	0.99
Eu-154	External gamma shielding factor	Unit-less	0.99
Eu-155	External gamma shielding factor	Unit-less	0.99
	Plant transfer factor for Eu	pCi/g plant per pCi/g soil	0.29
Pu-238	Plant transfer factor for Pu	pCi/g plant per pCi/g soil	0.92
	Depth of roots	m	-0.54
Pu-239	Plant transfer factor for Pu	pCi/g plant per pCi/g soil	0.92
	Depth of roots	m	-0.53
Pu-241	Plant transfer factor for Am	pCi/g plant per pCi/g soil	0.85
	Depth of roots	m	-0.44
	Kd of Am-241 (parent radionuclide) in contaminated zone	cm^3/g	0.28
Am-241	Plant transfer factor for Am	pCi/g plant per pCi/g soil	0.92
	Depth of roots	m	-0.54
Cm-243	Plant transfer factor for Cm	pCi/g plant per pCi/g soil	0.91
	External gamma shielding factor	Unit-less	0.58
	Depth of roots	m	-0.50

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Appendix 6D

Input Parameter Values for Soil DCGL Determination

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Soil Concentrations										
Basic radiation dose limit (mrem/y)		3	D	25	10 CFR 20.1402 (Ref. 1)	NR	NR	NR	NR	
Initial principal radionuclide (pCi/g)	P	2	D	1	Unit Value	NR	NR	NR	NR	
Distribution coefficients (cm^3/g) applied to contaminated, unsaturated, and saturated zone, unless otherwise noted										
Ac-227+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Ag-108m	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.38	2.1	0.001	0.999	216
Am-241	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2) – applied to saturated & unsaturated zones	7.28	3.15	0.001	0.999	
			D	1.20E+04	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) – applied only to contaminated zone in Pu-241 RESRAD run	NR	NR	NR	NR	1.45E+03
Am-243+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.28	3.15	0.001	0.999	1445
C-14	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.4	3.22	0.001	0.999	11
Cm-243	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.82	1.82	0.001	0.999	6761
Co-60	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.46	2.53	0.001	0.999	235
Cs-134	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.1	2.33	0.001	0.999	446
Cs-137+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.1	2.33	0.001	0.999	446
Eu-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Eu-154	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Eu-155	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825
Fe-55	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.34	2.67	0.001	0.999	209
Gd-152	P	1	S	Truncafted Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.72	3.22	0.001	0.999	825

Input Parameter Values for Soil DCGL Determination

Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
H-3	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2) – applied to saturated and unsaturated zones.	-2.81	0.5	0.001	0.999	6.00E-02
			D	4.30E-02	25th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) – applied to contaminated zone only	NR	NR	NR	NR	
Nb-94	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2) – applied to saturated and unsaturated zones.	5.94	3.22	0.001	0.999	
			D	3.31E+03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) – applied to contaminated zone only	NR	NR	NR	NR	3.80E+02
Ni-63	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.05	1.46	0.001	0.999	424
Np-237+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.84	2.25	0.001	0.999	17
Pa-231	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.94	3.22	0.001	0.999	380
Pb-210+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.78	2.76	0.001	0.999	2392
Pu-238	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.86	1.89	0.001	0.999	953
Pu-239	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.86	1.89	0.001	0.999	953
Pu-241+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	6.86	1.89	0.001	0.999	953
Ra-226+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.17	1.7	0.001	0.999	3533
Sb-125	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2) – applied to saturated and unsaturated zones.	5.94	3.22	0.001	0.999	
			D	3.31E+03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) – applied to contaminated zone only	NR	NR	NR	NR	3.80E+02
Sr-90+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.45	2.12	0.001	0.999	32

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Tc-99	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2) – applied to saturated and unsaturated zones.	-0.67	3.16	0.001	0.999	
			D	4.28E+00	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) – applied to contaminated zone only	NR	NR	NR	NR	5.10E-01
Th-229+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.68	3.62	0.001	0.999	5884
Th-230	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	8.68	3.62	0.001	0.999	5884
U-233	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.84	3.13	0.001	0.999	126
U-234	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.84	3.13	0.001	0.999	126
U-235+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.84	3.13	0.001	0.999	126
Initial concentration of radionuclides present in groundwater (pCi/l)	P	3	D	0	Assumed ground water uncontaminated	NR	NR	NR	NR	
Calculation Times										
Time since placement of material (y)	P	3	D	0		NR	NR	NR	NR	
Time for calculations (y)	P	3	D	0, 1, 3, 10, 30, 100, 300, 1000	RESRAD Default	NR	NR	NR	NR	
Contaminated Zone										
Area of contaminated zone (m^2)	P	2	D	13022	Site-specific- radiation control area (LTP App. 6A, Section 1)	NR	NR	NR	NR	
Thickness of contaminated zone (m)	P	2	S	Uniform	Minimum equal depth of soil mixing layer (0.15m) ; maximum equal to depth to water table (3.8m) (Ref. 4) – applied to all nuclides except C-14 and H-3	0.15	3.8	NR	NR	
			D	2.89E+00	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) – applied to C-14 and H-3 only	NR	NR	NR	NR	1.98E+00

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Length parallel to aquifer flow (m)	P	2	D	129	Site-specific - diameter of circle with an area of 13022 m^2 (LTP App. 6A, Section 1)	NR	NR	NR	NR	
Cover and Contaminated Zone Hydrological Data										
Cover depth (m)	P	2	D	0	Site-specific - no cover assumed	NR	NR	NR	NR	
Density of contaminated zone (g/cm^3)	P	1	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	1.5105	0.159	1.019	2.002	1.5105
Contaminated zone erosion rate (m/y)	P	2	D	8.5E-04	Calculated value based on site-specific slope of 2.9% (LTP App. 6A, Section 2)	NR	NR	NR	NR	
Contaminated zone total porosity	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	0.43	0.06	0.2446	0.6154	0.43
Contaminated zone field capacity	P	3	D	0.05	Site-specific value calculated (LTP App. 6A, Section 7) using Equation 4.4 in Ref. 5	NR	NR	NR	NR	0.05
Contaminated zone hydraulic conductivity (m/y)	P	2	S	Beta	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	110	5870	1.398	1.842	2506
Contaminated zone b parameter	P	2	S	Bounded Lognormal-N	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	- 0.0253	0.216	0.501	1.90	0.975
Humidity in air (g/m^3)	P	3	D	6.1	Regional value. (LTP App. 6A, Section 3)	NR	NR	NR	NR	
Evapotranspiration coefficient	P	2	S	Uniform	NUREG/CR-6697, Att. C (Ref. 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 (LTP App. 6A, Section 4)	NR	NR	NR	NR	
Precipitation (m/y)	P	2	D	1.2	Site-specific value calculated from site geographical area precip. (LTP App. 6A, Section 5)	NR	NR	NR	NR	
Irrigation (m/y)	B	3	S	Uniform	NUREG/CR-6697, Att. C methodology (Ref. 2)	0.252	0.618	NR	NR	0.435

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
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 (Ref. 2, App. 6A, Section 6)	NR	NR	NR	NR	
Watershed area for nearby stream or pond (m^2)	P	3	D	7.77E+05	Site-specific- drainage area (LTP App. 6A, Section 10)	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	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	1.5105	0.159	1.019	2.002	1.5105
Saturated zone total porosity	P	1	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	0.43	0.06	0.2446	0.6154	0.43
Saturated zone effective porosity	P	1	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	0.383	0.0610	0.195	0.572	0.383
Saturated zone field capacity	P	3	D	0.05	Calculated site-specific value LTP App. 6A, Section 7) using Equation 4.4 from Ref. 5	NR	NR	NR	NR	0.05
Saturated zone hydraulic conductivity (m/y)	P	1	S	Beta	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	110	5870	1.398	1.842	2506
Saturated zone hydraulic gradient	P	2	D	0.1	Site gradient (LTP App. 6A, Section 8)	NR	NR	NR	NR	
Saturated zone b parameter	P	2	S	Bounded Lognormal-N	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	- 0.0253	0.216	0.501	1.90	0.975
Water table drop rate (m/y)	P	3	D	1.00E-03	RESRAD Default	NR	NR	NR	NR	
Well pump intake depth (m below water table)	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 2)	6	10	30	NR	14.51
Model: Nondispersion (ND) or Mass-Balance (MB)	P	3	D	ND	ND model recommended for contaminant areas > 1,000 m^2 (Ref. 5)	NR	NR	NR	NR	

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Well pumping rate (m^3/y)	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 (Ref.. 2 and LTP App. 6A, Section 9)	957	1689	NR	NR	1323
Unsaturated Zone Hydrological Data										
Number of unsaturated zone strata	P	3	D	1	Site-specific value	NR	NR	NR	NR	
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 (Ref. 4)	0.01	3.65	NR	NR	1.82
Unsat. zone 1, soil density (g/cm^3)	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for6site soil type: sand (Ref. 4)	1.5105	0.159	1.019	2.002	1.5105
Unsat. zone 1, total porosity	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	0.43	0.06	0.2446	0.6154	0.43
Unsat. zone 1, effective porosity	P	2	S	Bounded Normal	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	0.383	0.0610	0.195	0.572	0.383
Unsat. zone 1, field capacity	P	3	D	0.05	Calculated site-specific value LTP App. 6A, Section 7, using Equation 4.4 from Ref. 5	NR	NR	NR	NR	0.05
Unsat. zone 1, hydraulic conductivity (m/y)	P	2	S	Beta	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	110	5870	1.398	1.842	2506
Unsat. zone 1, soil-specific b parameter	P	2	S	Bounded Lognormal-N	NUREG/CR-6697 dist. for site soil type: sand (Ref. 4)	- 0.0253	0.216	0.501	1.90	0.975
Occupancy										
Inhalation rate (m^3/y)	B	3	D	8400	NUREG/CR-6697, Att. C (Ref. 2)	NR	NR	NR	NR	
Mass loading for inhalation (g/m^3)	P	2	S	Continuous Linear	NUREG/CR-6697, Att. C (Ref. 2)	NR	NR	NR	NR	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 (Ref. 2)	0.15	0.95	NR	NR	0.55

Input Parameter Values for Soil DCGL Determination Resident Farmer Scenario										
Parameter (unit)	**Type[a]**	**Priority[b]**	**Treatment[c]**	**Value/Distribution**	**Basis**	**Distribution's Statistical Parameters[d]**				**Mean {Median}**
						1	**2**	**3**	**4**	
Shielding factor, external gamma	P	2	S	Bounded Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-1.3	0.59	0.044	1	{0.2725}
			D	3.98E-01	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) Applied to Ag-108m, Cm-243, Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Eu-155, Nb-94, and Sb-125.	NR	NR	NR	NR	3.12E-01
Fraction of time spent indoors	B	3	D	0.6571	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	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) (Ref. 6)	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/y)	B	2	D	112	NUREG/CR-5512, Vol. 3 (other vegetables + fruits + grain) (Ref. 6)	NR	NR	NR	NR	
Leafy vegetable consumption (kg/y)	B	3	D	21.4	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Milk consumption (L/y)	B	2	D	233	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Meat and poultry consumption (kg/y)	B	3	D	65.1	NUREG/CR5512, Vol. 3 (beef + poultry) (Ref. 6)	NR	NR	NR	NR	
Fish consumption (kg/y)	B	3	D	20.6	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Other seafood consumption (kg/y)	B	3	D	0.9	RESRAD Default	NR	NR	NR	NR	
Soil ingestion rate (g/y)	B	2	D	18.26	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Drinking water intake (L/y)	B	2	D	478.5	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	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	

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Contamination fraction of irrigation water	P	3	D	1	RESRAD Default - all water assumed contaminated	NR	NR	NR	NR	
Contamination fraction of aquatic food	P	2	D	1	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Contamination fraction of plant food	P	3	D	1	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Contamination fraction of meat	P	3	D	1	NUREG/CR-5512, Vol. 3 (Ref. 6)	NR	NR	NR	NR	
Contamination fraction of milk	P	3	D	1	NUREG/CR-5512, Vol. 3 (Ref. 6)	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 (Ref. 6)	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 (Ref. 6)	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 (Ref. 6)	NR	NR	NR	NR	
Livestock water intake for milk (L/day)	M	3	D	60	NUREG/CR5512, Vol. 3 Table 6.87 (Ref. 6)	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 (Ref. 6)	NR	NR	NR	NR	
Depth of soil mixing layer (m)	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 2)	0	0.15	0.6	NR	0.23
Depth of roots (m)	P	1	S	uniform	Min from NUREG/CR-6697,Att. C (Ref. 2). Max is site specific depth to water table	0.3	3.8	NR	NR	2.05E+00
			D	1.17E+00	25th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) Applied to Am-241, C-14, Cm-243, Cs-134, Cs137, H-3, Ni-63, Pu-238, Pu239, Pu-241, Sr-90, and Tc-99	NR	NR	NR	NR	
Drinking water fraction from ground water	P	3	D	1	RESRAD Default - all water assumed to be supplied from groundwater	NR	NR	NR	NR	

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {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 (Ref. 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 (Ref. 6)	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 (Ref. 6)	NR	NR	NR	NR	
Growing Season for Non-Leafy (years)	P	3	D	0.246	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Growing Season for Leafy (years)	P	3	D	0.123	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Growing Season for Fodder (years)	P	3	D	0.082	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Translocation Factor for Non-Leafy	P	3	D	0.1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Translocation Factor for Leafy	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Translocation Factor for Fodder	P	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Weathering Removal Constant for Vegetation (1/y)	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 6)	NR	NR	NR	NR	
Wet Foliar Interception Fraction for Leafy	P	2	S	Triangular	NUREG/CR-6697, Att. C (Ref. 2)	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 (Ref. 6)	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Non-Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	

Input Parameter Values for Soil DCGL Determination

Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Mean {Median}
						1	2	3	4	
Dry Foliar Interception Fraction for Leafy	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Dry Foliar Interception Fraction for Fodder	P	3	D	0.35	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	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 (Ref. 6)	NR	NR	NR	NR	
Leafy vegetables	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Milk	B	3	D	1	NUREG/CR-5512, Vol. 3 Table 6.87 (Ref. 6)	NR	NR	NR	NR	
Meat and poultry	B	3	D	20	NUREG/CR-5512, Vol. 3 Table 6.87 (holdup period for beef) (Ref. 6)	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	D	4.27E-01	75th percentile value (from RESRAD (.mco) file created in SA (Ref. 3)) Applied only to C-14	NR	NR	NR	NR	{3.00E-01}
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 (Ref. 2)	NR	NR	NR	NR	

Input Parameter Values for Soil DCGL Determination Resident Farmer Scenario										
Parameter (unit)	**Type[a]**	**Priority[b]**	**Treatment[c]**	**Value/Distribution**	**Basis**	**Distribution's Statistical Parameters[d]**				**Mean {Median}**
						1	**2**	**3**	**4**	
Fraction of grain in milk cow feed	B	3	D	0.1000	NUREG/CR-6697, Att. B (Ref. 2)	NR	NR	NR	NR	
Inhalation Exposure Dose Conversion Factors (mrem/pCi)										
Ac-227+ progeny	M	3	D	6.72E+00	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ag-108m	M	3	D	2.83E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Am-241	M	3	D	4.44E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Am-243+ progeny	M	3	D	4.40E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
C-14	M	3	D	2.09E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Cm-243	M	3	D	3.07E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Co-60	M	3	D	2.19E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Cs-134	M	3	D	4.63E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Cs-137+ progeny	M	3	D	3.19E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Eu-152	M	3	D	2.21E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Eu-154	M	3	D	2.86E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Eu-155	M	3	D	4.14E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Fe-55	M	3	D	2.69E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Gd-152	M	3	D	2.43E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Nb-94	M	3	D	4.14E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ni-63	M	3	D	6.29E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Np-237+ progeny	M	3	D	5.40E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pa-231	M	3	D	1.28E+00	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pb-210+ progeny	M	3	D	1.38E-02	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pu-238	M	3	D	3.92E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pu-239	M	3	D	4.29E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pu-241+ progeny	M	3	D	8.25E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ra-226+ progeny	M	3	D	8.60E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Sb-125	M	3	D	1.22E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	

Input Parameter Values for Soil DCGL Determination Resident Farmer Scenario										
Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Mean {Median}
						1	2	3	4	
Sr-90+ progeny	M	3	D	1.31E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Tc-99	M	3	D	8.33E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Th-229+ progeny	M	3	D	2.16E+00	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Th-230	M	3	D	3.26E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
U-233	M	3	D	1.35E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
U-234	M	3	D	1.32E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
U-235+ progeny	M	3	D	1.23E-01	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ingestion Exposure Dose Conversion Factors (mrem/pCi)										
Ac-227+ progeny	M	3	D	1.48E-02	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ag-108m	M	3	D	7.62E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Am-241	M	3	D	3.64E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Am-243+ progeny	M	3	D	3.63E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
C-14	M	3	D	2.09E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Cm-243	M	3	D	2.51E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Co-60	M	3	D	2.69E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Cs-134	M	3	D	7.33E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Cs-137+ progeny	M	3	D	5.00E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Eu-152	M	3	D	6.48E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Eu-154	M	3	D	9.55E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Eu-155	M	3	D	1.53E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Fe-55	M	3	D	6.07E-07	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Gd-152	M	3	D	1.61E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
H-3	M	3	D	6.40E-08	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Nb-94	M	3	D	7.14E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ni-63	M	3	D	5.77E-07	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Np-237+ progeny	M	3	D	4.44E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pa-231	M	3	D	1.06E-02	FGR-11 (Ref. 7)	NR	NR	NR	NR	

Input Parameter Values for Soil DCGL Determination Resident Farmer Scenario										
Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Mean {Median}
						1	2	3	4	
Pb-210+ progeny	M	3	D	5.37E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pu-238	M	3	D	3.20E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pu-239	M	3	D	3.54E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Pu-241+ progeny	M	3	D	6.85E-05	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Ra-226+ progeny	M	3	D	1.33E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Sb-125	M	3	D	2.81E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Sr-90+ progeny	M	3	D	1.53E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Tc-99	M	3	D	1.46E-06	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Th-229+ progeny	M	3	D	4.03E-03	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Th-230	M	3	D	5.48E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
U-233	M	3	D	2.89E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
U-234	M	3	D	2.83E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
U-235+ progeny	M	3	D	2.67E-04	FGR-11 (Ref. 7)	NR	NR	NR	NR	
Plant Transfer Factors (pCi/g plant per pCi/g soil)										
Ac-227+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	1.1	0.001	0.999	{1.0E-03}
Ag-108m	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-5.52	0.9	0.001	0.999	{4.0E-03}
Am-241	P	1	D	1.83E-03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3)) Applied also in the Pu-241 RESRAD run	NR	NR	NR	NR	1.48E-03
Am-243+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	{1.0E-03}
C-14	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-0.36	0.9	0.001	0.999	{7.0E-01}
Cm-243	P	1	D	1.83E-03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.48E-03
Co-60	P	1	D	1.46E-01	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.18E-01

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Cs-134	P	1	D	7.82E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	6.48E-02
Cs-137+ progeny	P	1	D	7.82E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	6.48E-02
Eu-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	{2.0E-03}
Eu-154	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	{2.0E-03}
Eu-155	P	1	D	4.21E-03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	3.60E-3
Fe-55	P	1	D	1.83E-03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.48E-03
Gd-152	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.1	0.001	0.999	{2.0E-03}
H-3	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	1.57	1.1	0.001	0.999	{4.8E+00}
Nb-94	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-4.61	1.1	0.001	0.999	{1.0E-02}
Ni-63	P	1	D	9.11E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	7.37E-02
Np-237+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.91	0.9	0.001	0.999	{2.0E-02}
Pa-231	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-4.61	1.1	0.001	0.999	{1.0E-02}
Pb-210+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-5.52	0.9	0.001	0.999	{4.0E-03}
Pu-238	P	1	D	1.83E-03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.48E-03
Pu-239	P	1	D	1.83E-03	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.48E-03
Pu-241+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	{1.0E-03}
Ra-226+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-3.22	0.9	0.001	0.999	{4.0E-02}
Sb-125	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-4.61	1.0	0.001	0.999	{1.0E-02}

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Sr-90+ progeny	P	1	D	5.90E-01	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	4.88E-01
Tc-99	P	1	D	9.16E+00	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	7.41E+00
Th-229+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	{1.0E-03}
Th-230	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.9	0.001	0.999	{1.0E-03}
U-233	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.9	0.001	0.999	{2.0E-03}
U-234	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.9	0.001	0.999	{2.0E-03}
U-235+ progeny	P	1	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.9	0.001	0.999	{2.0E-03}
Meat Transfer Factors (pCi/kg per pCi/d)										
Ac-227+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-9.90	0.2	0.001	0.999	{5.0E-05}
Am-243+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-10.82	1.0	0.001	0.999	{2.0E-05}
Co-60	P	2	D	5.86E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	4.85E-02
Cs-134	P	2	D	6.51E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	5.38E-02
Cs-137+ progeny	P	2	D	6.51E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	5.38E-02
Eu-152	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-6.21	1.0	0.001	0.999	{2.0E-03}
Eu-155	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	1.0	0.001	0.999	{2.0E-03}

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Fe-55	P	2	D	3.91E-02	75[th] percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	3.23E-02
Gd-152	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-5.30	0.9	0.001	0.999	{5.0E-03}
Np-237+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-12.21	1.0	0.001	0.999	{5.0E-06}
Pb-210+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-9.21	0.2	0.001	0.999	{1.0E-04}
Pu-241+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-9.21	0.2	0.001	0.999	{1.0E-04}
Ra-226+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-6.91	0.9	0.001	0.999	{1.0E-03}
Sr-90+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-9.21	0.7	0.001	0.999	{1.0E-04}
Th-229+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-7.13	0.7	0.001	0.999	{8.0E-04}
U-235+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-7.13	0.7	0.001	0.999	{8.0E-04}
Milk Transfer Factors (pCi/l per pCi/d)										
Ac-227+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-13.12	0.7	0.001	0.999	{2.0E-06}
Am-243+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-13.12	0.7	0.001	0.999	{2.0E-06}

Input Parameter Values for Soil DCGL Determination Resident Farmer Scenario										
Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Mean {Median}
						1	2	3	4	
C-14	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-6.21	0.7	0.001	0.999	{2.0E-03}
Cs-134	P	2	D	1.39E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.12E-02
Cs-137+ progeny	P	2	D	1.39E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	1.12E-02
Eu-152	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-9.72	0.9	0.001	0.999	{6.0E-05}
Eu-154	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-13.12	0.7	0.001	0.999	{2.0E-06}
Ni-63	P	2	D	3.21E-02	75th percentile value (from RESRAD (.mco) file created in the sensitivity analysis (Ref. 3))	NR	NR	NR	NR	2.54E-02
Np-237+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-12.21	0.9	0.001	0.999	{5.0E-06}
Pb-210+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 2)	-13.82	0.5	0.001	0.999	{1.0E-06}
Pu-241+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-13.82	0.5	0.001	0.999	{1.0E-06}
Ra-226+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 2)	-9.72	0.9	0.001	0.999	{6.0E-05}
Sr-90+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.21	0.5	0.001	0.999	{2.0E-03}
Tc-99	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-6.91	0.7	0.001	0.999	{1.0E-03}

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
Th-229+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 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 (Ref. 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 (Ref. 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 (Ref. 2)	-7.82	0.6	0.001	0.999	{4.0E-04}
U-235+ progeny	P	2	S	Truncated Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	-7.82	0.6	0.001	0.999	{4.0E-04}
Bioaccumulation Factors for Fish (pCi/kg per pCi/l)										
Ac-227+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.7	1.1	NR	NR	1.5E+01
Ag-108m	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	1.6	1.1	NR	NR	5.0E+00
Am-241	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Am-243+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
C-14	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	10.8	1.1	NR	NR	4.9E+04
Cm-243	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Co-60	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.7	1.1	NR	NR	3.0E+02
Cs-134	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.6	0.7	NR	NR	2.0E+03
Cs-137+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	7.6	0.7	NR	NR	2.0E+03
Eu-152	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Eu-154	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Eu-155	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Fe-55	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.3	1.1	NR	NR	2.0E+02
Gd-152	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.2	1.1	NR	NR	2.5E+01
H-3	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	0	0.1	NR	NR	1.0E+00
Nb-94	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.7	1.1	NR	NR	3.0E+02
Ni-63	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
Np-237+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Pa-231	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
Pb-210+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	5.7	1.1	NR	NR	3.0E+02
Pu-238	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d]				Mean {Median}
						1	2	3	4	
Pu-239	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Pu-241+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.4	1.1	NR	NR	3.0E+01
Ra-226+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3.9	1.1	NR	NR	4.9E+01
Sb-125	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
Sr-90+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.1	1.1	NR	NR	6.0E+01
Tc-99	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	3	1.1	NR	NR	2.0E+01
Th-229+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
Th-230	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	4.6	1.1	NR	NR	9.9E+01
U-233	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
U-234	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
U-235+ progeny	P	2	S	Lognormal-N	NUREG/CR-6697, Att. C (Ref. 2)	2.3	1.1	NR	NR	1.0E+01
Bioaccumulation Factors for Crustacea/ Mollusks (pCi/kg per pCi/l)										
Ac-227+ progeny	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+ progeny	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+ progeny	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
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	

Input Parameter Values for Soil DCGL Determination
Resident Farmer Scenario

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Parameters[d] 1	2	3	4	Mean {Median}
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+ progeny	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+ progeny	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+ progeny	P	3	D	1.00E+02	RESRAD Default	NR	NR	NR	NR	
Ra-226+ progeny	P	3	D	2.50E+02	RESRAD Default	NR	NR	NR	NR	
Sr-90+ progeny	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+ progeny	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+ progeny	P	3	D	6.00E+01	RESRAD Default	NR	NR	NR	NR	
Graphics Parameters										
Number of points				32	RESRAD Default	NR	NR	NR	NR	
Spacing				log	RESRAD Default	NR	NR	NR	NR	
Time Integration Parameter										
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
NR = Not required

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

Input Rank Correlation Coefficients for situation where distributions remain for both parameters:
Thickness of contaminated zone and unsaturated zone = - 0.99
Total porosity and bulk density = - 0.99 (contaminated zone, unsaturated and saturated zones)
Total porosity and effective porosity = 0.96 (unsaturated and saturated zones)
Effective porosity and bulk density = -0.99 (unsaturated and saturated zones)
Well Pumping Rate and Irrigation Rate = 0.96

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", Yu, C. et al., US Department of Energy-Argonne National Laboratory, November 2000.

3. YA-CALC-01-001-03, "RESRAD 6.21 Sensitivity Analysis for Resident Farmer Scenario - Soil," December 2003

4. YA-REPT-00-008-03, "Evaluation of GeoTesting Express Soil Testing and Determination of Depth to Groundwater," December 2003.

5. Yu, C., et al, Argonne National Laboratory, "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil," April 1993.

6. NUREG/CR-5512, "Residual Radioactive Contamination From Decommissioning," Volume 3: "Parameter Analysis, Draft Report for Comment," October 1999.

7. 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 (FGR-11), U.S. EPA, 1988.

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Appendix 6E

Soil DCGL Results

Soil DCGL Results

Nuclide	Dose Conversion Factor (DCF) (mrem/y per pCi/g)	DCGL (pCi/g)
H-3	6.79E-02	3.7E+02
C-14	4.52E+00	5.5E+00
Fe-55	8.57E-04	2.9E+04
Co-60	6.21E+00	4.0E+00
Ni-63	3.07E-02	8.1E+02
Sr-90	1.45E+01	1.7E+00
Nb-94	3.46E+00	7.2E+00
Tc-99	1.76E+00	1.4E+01
Ag-108m	3.44E+00	7.3E+00
Sb-125	7.82E-01	3.2E+01
Cs-134	5.02E+00	5.0E+00
Cs-137	2.92E+00	8.6E+00
Eu-152	2.43E+00	1.0E+01
Eu-154	2.63E+00	9.5E+00
Eu-155	6.29E-02	4.0E+02
Pu-238	7.48E-01	3.3E+01
Pu-239	8.30E-01	3.0E+01
Pu-241	2.54E-02	9.8E+02
Am-241	8.59E-01	2.9E+01
Cm-243	7.85E-01	3.2E+01

Appendix 6F

Basis Document for Site-Specific Parameter Value Assignment, Building Occupancy

Table of Contents

1. Room Dimension

An inventory of the rooms and partial rooms that would remain on site following Phase I of the DEMCO demolition project (Ref. 1) was used to determine room dimensions. Wall dimensions were determined from site drawings showing the building locations, building elevations and dimensions. Ceilings were not included in the model, as partial rooms remaining at the time of license termination will either have no ceiling or will be covered with a ceiling constructed of new, uncontaminated building material.

The average wall dimensions (wall length and height) were calculated using data describing the walls expected to remain at the time of final status survey. One wall was excluded from the data set: the 40.5 m long Primary Auxiliary Building (PAB) south wall. This wall extends the entire length of the PAB and was excluded from the data set because it is atypical of a standard room. The data on the walls expected to remain at the time of final status survey and the calculated average wall dimensions are shown in Table 1-1. The portion of the room expected to remain at the time of final status survey is highlighted. The resulting average wall length is 4.44 m, and the average wall height is 3.51 m.

These average wall dimensions were used to calculate the floor surface area (meters^2) for the modeled room:

Floor surface area = (4.44 m) x (4.44 m) = 19.71 m^2

The wall surface area in square meters was calculated from the average wall length and height in meters for the modeled room:

Wall area = (4.44 m) x (3.51 m) = 15.58 m^2

The floor and wall surface areas and the average wall length and height were used as inputs to the RESRAD-BUILD v 3.21 code to define the room model and to locate the receptor.

Tab;e 1- 1 Remaining Room/Walls Dimensions							
Building	**Area**	**Width**		**Length**		**Height**	
		Ft/in	**Meters**	**Ft/in**	**Meters**	**Ft/in**	**Meters**
PAB	TK-30 in (PAB Basement) **Room**	12’-6”	3.81E+00	15’-6”	4.72E+00	18’-6”	5.64E+00
PAB	TK-27 (PAB Basement) **Room**	10’-2”	3.10E+00	15’-6”	4.72E+00	18’-6”	5.64E+00
PAB	South **Wall** (G-Line)			133’-0”*	4.05E+01	13’-0”	3.96E+00
PAB	East **Wall** (2-Line to Fa)			17’-0”	5.18E+00	13’-0”	3.96E+00
I-X PIT	Southernmost **Wall**			33’-0”	1.01E+01	14’-8”	4.47E+00
I-X PIT	Easternmost **Wall** (Total Length)			31’-10”	9.70E+00	14’-8”	4.47E+00
SFP	Spent Fuel **Pool**	16’-6”	5.03E+00	33’-8”	1.03E+01	14’-8”	4.47E+00
New Fuel Vault	New Fuel Storage (South **Wall**)			15’-0”	4.57E+00	13’-6”	4.11E+00
Safe Shutdown	Pipe Chase **Cubicle**	4’-0”	1.22E+00	4’-0”	1.22E+00	8’-0”	2.44E+00
Waste Vault	Waste Transfer Pit **Cubicle**	9’-0”	2.74E+00	14’-0”	4.27E+00	9’-10”	3.00E+00
Elevator Pit	Elevator Pit **Cubicle**	7’-10”	2.39E+00	9’-0”	2.74E+00	6’-6”	1.98E+00
Waste Disposal	Pipe Chase **Cubicle**	5’-0”	1.52E+00	11’-10”	3.61E+00	10’-1”	3.07E+00
Waste Disposal	Distillate Heat Exchanger **Cubicle**	9’-0”	2.74E+00	16’-0”	4.88E+00	7’-0”	2.13E+00
Waste Disposal	Evaporator **Cubicle**	10’-6”	3.20E+00	16’-0”	4.88E+00	7’-0”	2.13E+00
Waste Disposal	Drumming Pit **Cubicle**	10’-4”	3.15E+00	27’-0”	8.23E+00	7’-0”	2.13E+00
PAB	PAB Back Stairwell Pit **Cubicle**	11’-4”	3.45E+00	13’-0”	3.96E+00	8’-2”	2.49E+00
Average Wall Length (meters) = 4.44E+00							
Average Wall Height (meters) = 3.51E+00							

* As previously noted, the south (G-Line) wall of the PAB is excluded from the calculation of average wall length.

Table 1- 2

Remaining Structures and Drawing Reference

Building	Room/Wall/Pit	Room/Wall Width	Drawing Reference	Wall Length	Drawing Reference	Wall Height (Note 1)	Drawing Reference
PAB	Drain Collecting Tank Room (TK-30)	12' 6"	PAB 9699-FC-40D	15' 6"	PAB 9699-RC-40A	1022' 8"-1004' 2"=18' 6"	PAB 9699-FM-57A
PAB	Gravity Drain Tank Room (TK-27)	10' 2"	PAB 9699-FC-40D	15' 6"	PAB 9699-RC-40A	1022' 8"-1004' 2"=18' 6"	PAB 9699-FM-57A
PAB	South Wall (G-Line)			133' 0"	PAB 9699-FR-16A	1035' 8" - 1022' 8' =13' 0"	PAB 9699-FM-57A
PAB	East Wall (2-Line to Fa)			17' 0"	PAB 9699-FR-16A	1035' 8" - 1022' 8' =13' 0"	PAB 9699-FM-57A
I-X PIT	Southernmost Wall			33' 0"	I-X Pit 9699-FM-35B	1035' 8" - 1021' 0" =14' 8"	I-X Pit 9699-FM-35B
I-X PIT	Easternmost Wall F to E			25' 6"	PAB 9699-FM-57A	1035' 8" - 1021' 0" =14' 8"	I-X Pit 9699-FM-35B
I-X PIT	Easternmost Wall E to Wall End			6'4"	I-X Pit 9699-FM-35B		
I-X PIT	Easternmost Wall (Total Length)			31' 10"			
SFP	Spent Fuel Pool	16' 6"	Fuel Pit 9699-FM-21A	33' 8"	Fuel Pit 9699-FM-21A	1022' 8" - 1008' 0" =14' 8"	Fuel Pit 9699-FC-45B
New Fuel Vault	New Fuel Storage (South Wall)			15' 0"	PAB 9699-FM-57A	1035' 0" - 1021' 6" =13' 6"	Fuel Pit 9699-FM-21A
Safe Shutdown	Pipe Chase (555)	4' 0"	CES Rev.1 85005-F-1001	4' 0"	CES Rev.1 85005-F-1001	1034' 0" - 1026' 0" = 8' 0"	CES Rev.1 85005-F-1001
Waste Vault	Waste Transfer Pump Pit (underground)	9' 0"	9699-FC-50C	14' 0"	9699-FC-50C	1020' 6" - 1010' 8" =9' 10"	9699-FC-50C
Elevator Pit	Elevator Pit	7' 10"	PAB 9699-FC-43A	9' 0"	PAB 9699-FC-43A	1022' 8' - 1016' 2" =6' 6"	PAB 9699-FC-43A
Waste Disposal	Pipe Chase Cubicle	5' 0"	Waste Disp.9699-FA-17A	11' 10"	Waste Disp.9699-FA-17A	1035' 8" - 1025' 7" = 10' 1"	Waste Disp.9699-FA-17A
Waste Disposal	Distillate Heat Exchanger Cubicle	9' 0"	Waste Disp.9699-FA-17A	16' 0"	Waste Disp.9699-FA-17A	1035' 8" - 1028' 8" = 7' 0"	Waste Disp.9699-FA-17A
Waste Disposal	Evaporator Cubicle	10' 6"	Waste Disp.9699-FA-17A	16' 0'	Waste Disp.9699-FA-17A	1035' 8" - 1028' 8" = 7' 0"	Waste Disp.9699-FA-17A
Waste Disposal	Drumming Pit Cubicle	10' 4"	Waste Disp.9699-FA-17A	27' 0"	Waste Disp.9699-FA-17A	1035' 8" - 1028' 8" = 7' 0"	Waste Disp.9699-FA-17A
PAB	Back of PAB Stairwell Pit Cubicle	11' 4"	PAB 9699 RC-40B	13' 0"	PAB 9699 RC-40B	1035' 8" - 1027' 6' = 8' 2"	PAB 9699-FM-57B

Note 1: Top/ceiling height elevation is from DEMCO work scope Ref. 1

2. Source Configuration

NUREG/CR-6755 (Ref. 2), Section 4.1, describes three principal assumptions inherent in the Building Occupancy scenario: a fixed room area, uniform surface contamination, and the receptor location at the center of the floor at a height of 1 m. The configuration of the receptor and sources is illustrated in Figure 2-1. The RESRAD- BUILD input parameters, receptor location and center of source coordinates, are provided in Table 2-1.

Figure 2-1
Configuration of Source and Receptor Locations
for RESRAD-BUILD Model

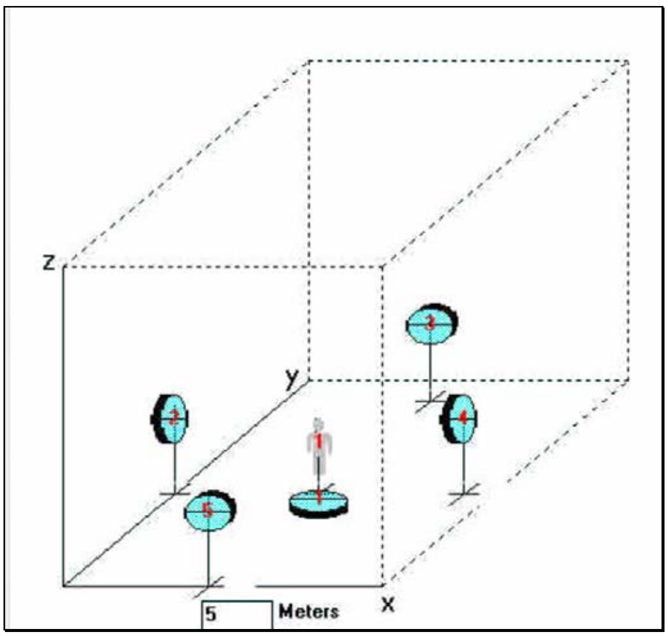

Table 2-1
Receptor and Center of Source Locations, meters

Source #	Source Description	Axis		
		X	Y	Z
1	Floor	2.22	2.22	0
2	West Wall	0	2.22	1.76
3	North Wall	2.22	4.44	1.76
4	East Wall	4.44	2.22	1.76
5	South Wall	2.22	0	1.76
	Receptor Location	2.22	2.22	1

3. Direct Ingestion Rate

The source specific input parameter, Direct Ingestion Rate, is described in RESRAD-BUILD as the direct ingestion rate of the source by any receptor in the room. Direct ingestion is possible only if the receptor and the source are in the same room and represents the fraction of the source ingested per hour.

NUREG/CR-5512, Volume 3, (Ref. 3) defines the average ingestion rate of 1.1E-4 m^2/hr as representative for the average individual in an industrial setting. The Direct Ingestion Rate for use in the Building Occupancy Scenario is calculated based upon the total room surface area (source area). The surface area is equal to sum of the surface area of four walls (15.58 m^2 per wall, as discussed in Section 1) plus the surface area of the floor (19.71 m^2, as discussed in Section 1).

$$\begin{aligned}\text{Direct Ingestion Rate} &= \text{Average Ingestion Rate / Source Area}\\ &= (1.1\text{E-04 m}^2\text{/hr}) / ((4 \times 15.58\text{ m}^2) + 19.71\text{ m}^2)\\ &= (1.1\text{E-04 m}^2\text{/hr}) / (82.03\text{ m}^2)\\ &= 1.34\text{E-06 hr}^{-1}\end{aligned}$$

The direct ingestion defined in this manner used in conjunction with an indirect ingestion rate set to zero, adequately models the Building Occupancy Ingestion pathway.

References:

1 Attachment E to the "Contract for the Performance of Demolition and Disposal and Related Services, By and Between DEMCO, Inc. and Yankee Atomic Electric Company," dated February 28, 2003.

2. NUREG/CR-6755, “Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code,” February. 2002 (ANL/EAD/TM/02-1).

3. NUREG/CR-5512, “Residual Radioactive Contamination from Decommissioning,” Volume 3: “Parameter Analysis, Draft Report for Comment,” October 1999 (SAND99-2148).

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Appendix 6G

Input Parameter Values for Sensitivity Analysis, Building Occupancy

Input Parameter Values for Sensitivity Analysis, Building Occupancy

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Paramters[d]				Median
						1	2	3	4	
General										
Exposure Duration (days)	B	3	D	365.25	NUREG/CR-5512 (Ref. 1), Vol.3, Section 5.2.1	NR	NR	NR	NR	NR
Indoor Fraction	B	2	D	0.267	NUREG/CR-5512 (Ref. 1), Vol. 3, Section 5.2.2	NR	NR	NR	NR	NR
Evaluation Time (year)	P	3	D	0	t=0 corresponds maximum dose over the first year	NR	NR	NR	NR	NR
Number of Rooms	P	3	D	1	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Deposition Velocity (m/sec)	P	2	S	Loguniform	NUREG/CR-6755 (Ref. 2), Section 3.3	2.70E-06	2.70E-03	-	-	-
Resuspension Rate (sec^{-1})	P	1	S	Loguniform	NUREG/CR-6755 (Ref. 2), Section 3.1	2.5E-11	1.35E-5	-	-	-
Air exchange rate for room (1/ h)	B	2	D	1.52	NUREG/CR-6697 (Ref. 3), Att. C, 7.4 and NUREG/CR-6755 (Ref. 2), Section 3.2	NR	NR	NR	NR	NR
Room area (m^2)	P	2	D	19.71	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Room height (m)	P	2	D	3.51	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Time fraction	B	3	D	1	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Breathing rate (m^3/day)	B	2	D	33.6	NUREG/CR-5512 (Ref. 1) Vol. 3 Section 5.3	NR	NR	NR	NR	NR
Indirect ingestion rate (m^2/hr)	B	2	D	0	NUREG/CR-5512 (Ref. 1) Vol. 3 Section 5.2.3 Indirect ingestion is not modeled	NR	NR	NR	NR	NR
Receptor location: x,y,z (m)	B	3	D	2.22, 2.22, 1	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Shielding thickness (cm)	P	2	D	0	No shielding assumed	NR	NR	NR	NR	NR
Shielding density (g/cc)	P	1	D	0	No shielding assumed	NR	NR	NR	NR	NR
Shielding material	P	3	D	None	No shielding assumed	NR	NR	NR	NR	NR
Number of sources	P	3	D	5	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
External dose conversion factor ((mrem/yr)/(dpm/m^2))	M	3	D	RESRAD-BUILD default	FGR-12 (Ref. 4)	NR	NR	NR	NR	NR
Air submersion dose conversion factor ((mrem/yr)/(pCi/m^3))	M	3	D	RESRAD-BUILD default	FGR-12 (Ref. 4)	NR	NR	NR	NR	NR
Inhalation dose conversion factor (mrem/pCi/g)	M	3	D	RESRAD-BUILD default	FGR-11 (Ref. 5)	NR	NR	NR	NR	NR

Input Parameter Values for Sensitivity Analysis, Building Occupancy

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Paramters[d]				Median
						1	2	3	4	
Ingestion dose conversion factor (mrem/pCi/g)	M	3	D	RESRAD-BUILD default	FGR-11 (Ref. 5)	NR	NR	NR	NR	NR
Source 1. Floor										
Type	P	3	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	3	D	Z	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	3	D	2.22, 2.22, 0	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m2)	P	2	D	19.71	Site-specific model, LTP App.6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	2	D	1	NUREG/CR-6697 (Ref. 3), Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	2	D	0.07	NUREG/CR-6697 (Ref. 3), Att. C, Section 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr^{-1})	B	2	D	1.34E-6	NUREG/CR-5512 (Ref. 1), Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m^2/h / 82.03 m^2)	NR	NR	NR	NR	NR
Removable fraction	P	1	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), Section 3.5	NR	NR	NR	NR	NR
Time for source removal (days)	P	2	S	Triangular	NUREG/CR-6755 (Ref. 2), Section 3.6	1000	100000	10000	-	-
Radionuclide concentration (pCi/m^2)	P	2	D	1	Assumes unit concentration	NR	NR	NR	NR	NR
Source 2. West Wall										
Type	P	3	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	3	D	X	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	3	D	0, 2.22, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m2)	P	2	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	2	D	1	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	2	D	0.07	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR

Input Parameter Values for Sensitivity Analysis, Building Occupancy

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Paramters[d]				Median
						1	2	3	4	
Direct ingestion (hr-1)	B	2	D	1.34E-6	NUREG/CR-5512 (Ref. 1) Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m^2/h / 82.03 m^2)	NR	NR	NR	NR	NR
Removable fraction	P	1	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), 3.5	NR	NR	NR	NR	NR
Time for source removal (days)	P	2	S	Triangular	NUREG/CR-6755 (Ref. 2), 3.6 and NUREG/CR- 6697 (Ref. 3)	1000	100000	10000	-	-
Radionuclide concentration (pCi/m^2)	P	2	D	1	Allows for proportional DCGL calculation	NR	NR	NR	NR	NR
Source 3. North Wall										
Type	P	3	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	3	D	Y	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	3	D	2.22, 4.44, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m2)	P	2	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	2	D	1	NUREG/CR-6697 (Ref. 3) Att. C, 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	2	D	0.07	NUREG/CR-6697 (Ref. 3) Att. C, 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr-1)	B	2	D	1.34E-6	NUREG/CR-5512 (Ref. 1) Vol. 3,5.2.3 1.1E-04m^2/h / 82.03m^2	NR	NR	NR	NR	NR
Removable fraction	P	1	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), 3.5	NR	NR	NR	NR	NR
Time for source removal (days)	P	2	S	Triangular	NUREG/CR-6755 (Ref. 2), 3.6 and NUREG/CR- 6697 (Ref. 3)	1000	100000	10000	-	-
Radionuclide concentration (pCi/m^2)	P	2	D	1	Allows for proportional DCGL calculation	NR	NR	NR	NR	NR
Source 4. East Wall										
Type	P	3	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	3	D	X	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	3	D	4.44, 2.22, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR

Input Parameter Values for Sensitivity Analysis, Building Occupancy

Parameter (unit)	Type[a]	Priority[b]	Treatment[c]	Value/Distribution	Basis	Distribution's Statistical Paramters[d]				Median
						1	2	3	4	
Area (m2)	P	2	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	2	D	1	NUREG/CR-6697 (Ref. 3), Att. C, 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	2	D	0.07	NUREG/CR-6697 (Ref. 3), Att. C, 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr-1)	B	2	D	1.34E-6	NUREG/CR-5512 (Ref. 1),Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 ($1.1E\text{-}04 m^2/h$ / $82.03 m^2$)	NR	NR	NR	NR	NR
Removable fraction	P	1	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), 3.5	NR	NR	NR	NR	NR
Time for source removal (days)	P	2	S	Triangular	NUREG/CR-6755 (Ref. 2), 3.6 and NUREG/CR- 6697	1000	100000	10000	-	-
Radionuclide concentration (pCi/m^2)	P	2	D	1	Allows for proportional DCGL calculation	NR	NR	NR	NR	NR
Source 5. South Wall										
Type	P	3	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	3	D	Y	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	3	D	2.22, 0, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m2)	P	2	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	2	D	1	NUREG/CR-6697 (Ref. 3) Att. C, 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	2	D	0.07	NUREG/CR-6697 (Ref. 3) Att. C, 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr-1)	B	2	D	1.34E-6	NUREG/CR-5512 (Ref. 1), Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 ($1.1E\text{-}04 m^2/h$ / $82.03 m^2$)	NR	NR	NR	NR	NR
Removable fraction	P	1	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), 3.5	NR	NR	NR	NR	NR
Time for source removal (days)	P	2	S	Triangular	NUREG/CR-6755 (Ref. 2). 3.6 and NUREG/CR- 6697 (Ref. 3)	1000	100000	10000	-	-

Input Parameter Values for Sensitivity Analysis, Building Occupancy										
Parameter (unit)	**Type[a]**	**Priority[b]**	**Treatment[c]**	**Value/Distribution**	**Basis**	**Distribution's Statistical Paramters[d]**				**Median**
						1	**2**	**3**	**4**	
Radionuclide concentration (pCi/m^2)	P	2	D	1	Allows for proportional DCGL calculation	NR	NR	NR	NR	NR

Notes:

[a] P = physical, B = behavioral, M = metabolic (NUREG/CR-6697, Att. B, Table 4.3)
[b] 1 = high priority parameter, 2 = medium priority parameter, 3 = low priority parameter (NUREG/CR-6697, Att. B, Table 4.3)
[c] D = deterministic, S = stochastic
[d] Statistical Parameters
Loguniform 1 = minimum, 2 = maximum
Triangular 1 = minimum, 2 = maximum, 3 = most likely
NR = Not required

Input Correlations: Resuspension Rate and Deposition Velocity = 0.9
Time to Source Removal = 0.9 (correlation set between sources)

Run Specifications: Random seed = 1000
Number of observations = 300
Number of repetitions = 1
Dose integrations = 5

References:

1. NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning," Volume 3: "Parameter Analysis, Draft Report for Comment," October 1999 (SAND99-2148).

2. NUREG/CR-6755, "Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code," February. 2002 (ANL/EAD/TM/02-1).

3. NUREG/CR-6697,"Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes," November 2000, (ANL/EAD/TM-98).

4. Eckerman, K.F., et al, "External Exposure to Radionuclides In Air, Water, And Soil," EPA 402-R-93-081, Federal Guidance Report No. 12 (FGR-12), U.S. EPA, 1993.

5. 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 (FGR-11), U.S. EPA, 1988.

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

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Appendix 6H

Results of Sensitivity Analysis, Building Occupancy

Results of Sensitivity Analysis, Building Occupancy
(Based on the Partial Rank Correlation Coefficient)

Radionuclide	Rank 1 parameter	Rank 2 parameter	Rank 3 parameter	Rank 4 parameter	Rank 5 parameter	Rank 6 parameter	Rank 7 parameter
H-3	RFO(1) -0.85	RFO(2) -0.78	RFO(4) -0.78	RFO(5) -0.76	RFO(3) -0.76	DKSUS 0.35	UD -0.31
C-14	RFO(1) 0.89	RFO(4) 0.85	RFO(3) 0.84	RFO(5) 0.82	RFO(2) 0.82	--	--
Fe-55	RFO(3) 0.28	RFO(1) 0.28	RFO(4) 0.27	RFO(5) 0.23	DKSUS 0.23	UD -0.21	RFO(2) 0.15
Co-60	RFO(1) 0.45	RFO(5) 0.16	DKSUS -0.14	RFO(2) 0.13	UD 0.13	RFO(4) 0.10	--
Ni-63	RFO(1) - 0.64	RFO(3) -0.58	RFO(5) -0.58	RFO(2) -0.58	RFO(4) -0.58	DKSUS 0.12	--
Sr-90	RFO(1) - 0.71	RFO(4) -0.59	RFO(2) -0.56	RFO(5) -0.56	RFO(3) -0.52	DKSUS 0.37	UD -0.30
Nb-94	DKSUS -0.38	UD 0.25	RFO(1) 0.24	--	--	--	--
Tc-99	RFO(1) 0.40	RFO(4) 0.34	RFO(2) 0.29	RFO(3) 0.27	RFO(5) 0.26	--	--
Ag-108m	RFO(1) 0.50	DKSUS -0.49	UD 0.41	RFO(4) 0.13	RFO(5) -0.10	--	--
Sb-125	RFO(1) 0.65	RFO(4) 0.38	RFO(3) 0.32	UD 0.26	DKSUS -0.25	RFO(2) 0.20	RFO(5) 0.16
Cs-134	RFO(1) 0.37	RFO(4) 0.20	RFO(2) 0.16	RFO(5) 0.15	RFO(3) 0.14	UD 0.12	DKSUS -0.11
Cs-137	RFO(1) 0.81	RFO(5) 0.59	RFO(3) 0.59	RFO(4) 0.57	RFO(2) 0.54	DKSUS -0.48	UD 0.43
Eu-152	RFO(1) 0.24	RFO(5) 0.11	--	--	--	--	--
Eu-154	RFO(1) 0.42	DKSUS -0.20	UD 0.17	RFO(2) 0.13	--	--	--
Eu-155	RFO(4) -0.48	RFO(5) -0.46	RFO(3) -0.44	RFO(2) -0.43	RFO(1) 0.19	--	--
Pu-238	RFO(1) -0.91	RFO(4) -0.87	RFO(3) -0.86	RFO(5) -0.85	RFO(2) -0.85	DKSUS 0.11	--
Pu-239	RFO(1) -0.91	RFO(4) -0.87	RFO(3) -0.86	RFO(5) -0.85	RFO(2) -0.85	--	--
Pu-241	DKSUS 0.60	UD -0.54	RFO(2) -0.14	RFO(1) -0.13	RFO(4) -0.13	--	--
Am-241	RFO(1) -0.91	RFO(4) -0.87	RFO(3) -0.86	RFO(5) -0.85	RFO(2) -0.85	--	--
Cm-243	RFO(1) -0.91	RFO(4) -0.86	RFO(3) -0.85	RFO(5) -0.85	RFO(2) -0.84	DKSUS 0.22	UD -0.17

Parameter Definition:

DKSUS = Resuspension Rate

UD = Deposition Velocity

RFO(#) = Time for Source Removal, where # represents the source number delineated as follow: 1=floor, 2=-west wall, 3=north wall, 4=east wall, 5=south wall

Appendix 6I

Input Parameter Values for Building Occupancy DCGL Determination

Input Parameters for Building Occupancy DCGL Determination										
Parameter (unit)	**Type[a]**	**Radionuclide**	**Treatment[b]**	**Value or Distribution**	**Basis**	**Distribution's Statistical Parameters[c]**				**Median**
						1	**2**	**3**	**4**	
General										
Exposure Duration (days)	B	All	D	365.25	NUREG/CR-5512 (Ref. 1), Vol.3, 5.2.1	NR	NR	NR	NR	NR
Indoor Fraction	B	All	D	0.267	NUREG/CR-5512 (Ref. 1), Vol. 3, 5.2.2	NR	NR	NR	NR	NR
Evaluation Time (year)	P	All	D	0	t=0 corresponds maximum dose over the first year (year 9 for Pu-241)	NR	NR	NR	NR	NR
Number of Rooms	P	All	D	1	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Deposition Velocity (m/sec)	P	H-3	D	1.51E-05	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Fe-55	D	1.51E-05	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Co-60	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Sr-90	D	1.51E-05	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Nb-94	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Ag-108m	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Sb-125	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Cs-134	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Cs-137	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Eu-154	D	4.79E-04	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-

Input Parameters for Building Occupancy DCGL Determination										
Parameter (unit)	**Type[a]**	**Radionuclide**	**Treatment[b]**	**Value or Distribution**	**Basis**	**Distribution's Statistical Parameters[c]**				**Median**
						1	**2**	**3**	**4**	
	P	Pu-241	D	1.51E-05	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Cm-243	D	1.51E-05	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	All others	S	Loguniform	NUREG/CR-6755 (Ref. 2), Section 3.3	2.70E-06	2.70E-03	-	-	8.53E-05
Resuspension Rate (sec^{-1})	P	H-3	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Fe-55	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Co-60	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Ni-63	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Sr-90	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Nb-94	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Ag-108	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Sb-125	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Cs-134	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Cs-137	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Eu-154	D	6.75E-10	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Pu-238	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	Pu-241	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c]				Median
						1	2	3	4	
	P	Cm-243	D	1.02E-06	Mean value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	-	-	-
	P	All others	S	Loguniform	NUREG/CR-6755 (Ref. 2), 3.1	2.5E-11	1.3E-5	-	-	1.83E-08
Air exchange rate for room (hr^{-1})	B	All	D	1.52	NUREG/CR-6697 (Ref. 3), Att.C, Section 7.4 and NUREG/CR-6755 (Ref. 2), Section 3.2	NR	NR	NR	NR	NR
Room area (m^2)	P	All	D	19.71	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Room height (m)	P	All	D	3.51	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Time fraction	B	All	D	1	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Breathing rate (m^3/day)	B	All	D	33.6	NUREG/CR-5512 (Ref. 1) Vol. 3 Section 5.3	NR	NR	NR	NR	NR
Indirect ingestion rate (m^2/hr)	B	All	D	0	NUREG/CR-5512 (Ref. 1) Vol. 3 Section 5.2.3 Indirect ingestion is not modeled	NR	NR	NR	NR	NR
Receptor location: x,y,z (m)	B	All	D	2.22, 2.22, 1	NUREG/CR-5512 (Ref. 1) and LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Shielding thickness (cm)	P	All	D	0	No shielding assumed	NR	NR	NR	NR	NR
Shielding density (g/cc)	P	All	D	0	No shielding assumed	NR	NR	NR	NR	NR
Shielding material	P	All	D	None	No shielding assumed	NR	NR	NR	NR	NR
Number of sources	P	All	D	5	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
External dose conversion factor ((mrem/yr)/(dpm/m^2))	M	All	D	RESRAD-BUILD default	FGR-12 (Ref. 4)	NR	NR	NR	NR	NR
Air submersion dose conversion factor ((mrem/yr)/(pCi/m^3))	M	All	D	RESRAD-BUILD default	FGR-12 (Ref. 4)	NR	NR	NR	NR	NR
Inhalation dose conversion factor (mrem/pCi/g)	M	All	D	RESRAD-BUILD default	FGR-11 (Ref. 5)	NR	NR	NR	NR	NR
Ingestion dose conversion factor (mrem/pCi/g)	M	All	D	RESRAD-BUILD default	FGR-11 (Ref. 5)	NR	NR	NR	NR	NR
Source 1. Floor										
Type	P	All	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	All	D	Z	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	All	D	2.22, 2.22, 0	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c]				Median
						1	2	3	4	
Area (m^2)	P	All	D	19.71	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	All	D	1	NUREG/CR-6697 (Ref. 3), Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	All	D	0.07	NUREG/CR-6697 (Ref. 3), Att. C, Section 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr^{-1})	B	All	D	1.34E-6	NUREG/CR-5512 (Ref. 1), Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m^2/h / 82.03 m^2)	NR	NR	NR	NR	NR
Removable fraction	P	All	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), Section 3.5	NR	NR	NR	NR	NR
Time for source removal (days)	P	H-3	D	18240	25^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	C-14	D	52777	75^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Fe-55	D	52777	75^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Co-60	D	52777	75^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ni-63	D	18240	25^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sr-90	D	18240	25^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Nb-94	D	52777	75^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Tc-99	D	52777	75^{th} percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination										
Parameter (unit)	**Type[a]**	**Radionuclide**	**Treatment[b]**	**Value or Distribution**	**Basis**	**Distribution's Statistical Parameters[c]**				**Median**
						1	**2**	**3**	**4**	
	P	Ag-108m	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sb-125	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-134	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-137	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-152	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-154	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-155	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-238	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-239	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Am-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cm-243	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
Radionuclide concentration (pCi/m^2)	P	All	D	1	Assumed unit concentration	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c] 1	2	3	4	Median
Source 2. West Wall										
Type	P	All	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	All	D	X	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	All	D	0, 2.22, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m^2)	P	All	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	All	D	1	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	All	D	0.07	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr^{-1})	B	All	D	1.34E-6	NUREG/CR-5512 (Ref. 1) Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m^2/h / 82.03m^2)	NR	NR	NR	NR	NR
Removable fraction	P	All	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), Section 3.5	NR	NR	NR	NR	NR
Time for source removal (d)	P	H-3	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	C-14	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Fe-55	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Co-60	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ni-63	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sr-90	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c]				Median
						1	2	3	4	
	P	Nb-94	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Tc-99	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ag-108m	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sb-125	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-134	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-137	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-152	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-154	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-155	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-238	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-239	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Am-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c] 1	2	3	4	Median
	P	Cm-243	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
Radionuclide concentration (pCi/m²)	P	All	D	1	Assumed unit concentration	NR	NR	NR	NR	NR
Source 3. North Wall										
Type	P	All	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	All	D	Y	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	All	D	2.22, 4.44, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m2)	P	All	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	All	D	1	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	All	D	0.07	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr^{-1})	B	All	D	1.34E-6	NUREG/CR-5512 (Ref. 1) Vol. 3, Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m²/h / 82.03m²)	NR	NR	NR	NR	NR
Removable fraction	P	All	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), Section 3.5	NR	NR	NR	NR	NR
Time for source removal (d)	P	H-3	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	C-14	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Fe-55	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Co-60	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ni-63	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination										
Parameter (unit)	**Type[a]**	**Radionuclide**	**Treatment[b]**	**Value or Distribution**	**Basis**	**Distribution's Statistical Parameters[c]**				**Median**
						1	**2**	**3**	**4**	
	P	Sr-90	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Nb-94	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Tc-99	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ag-108m	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sb-125	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-134	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-137	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-152	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-154	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-155	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-238	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-239	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c] 1	2	3	4	Median
	P	Am-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cm-243	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
Radionuclide concentration (pCi/m^2)	P	All	D	1	Assumed unit concentration	NR	NR	NR	NR	NR
Source 4. East Wall										
Type	P	All	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	All	D	X	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	All	D	4.44, 2.22, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m^2)	P	All	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	All	D	1	NUREG/CR-6697 (Ref. 3), Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	All	D	0.07	NUREG/CR-6697 (Ref. 3), Att. C, Section 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr^{-1})	B	All	D	1.34E-6	NUREG/CR-5512 (Ref. 1), Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m^2/h / 82.03m^2)	NR	NR	NR	NR	NR
Removable fraction	P	All	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), Section 3.5	NR	NR	NR	NR	NR
Time for source removal (d)	P	H-3	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	C-14	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Fe-55	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Co-60	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination										
Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c]				Median
						1	2	3	4	
	P	Ni-63	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sr-90	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Nb-94	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Tc-99	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ag-108m	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sb-125	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-134	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-137	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-152	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-154	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-155	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-238	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-239	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c] 1	2	3	4	Median
	P	Pu-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Am-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cm-243	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
Radionuclide concentration (pCi/m^2)	P	All	D	1	Assumed unit concentration	NR	NR	NR	NR	NR
Source 5. South Wall										
Type	P	All	D	Area	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Direction	P	All	D	Y	NUREG/CR-5512 (Ref. 1)	NR	NR	NR	NR	NR
Location of center of source: x,y,z (m)	P	All	D	2.22, 0, 1.76	Site-specific model, LTP App. 6F, Section 2	NR	NR	NR	NR	NR
Area (m^2)	P	All	D	15.58	Site-specific model, LTP App. 6F, Section 1	NR	NR	NR	NR	NR
Air fraction for H-3	B	All	D	1	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Air fraction (for all nuclides except H-3)	B	All	D	0.07	NUREG/CR-6697 (Ref. 3) Att. C, Section 8.6	NR	NR	NR	NR	NR
Direct ingestion (hr^{-1})	B	All	D	1.34E-6	NUREG/CR-5512 (Ref. 1), Section 5.2.3 and LTP App. 6F, Section 3 (1.1E-04m^2/h / 82.03m^2)	NR	NR	NR	NR	NR
Removable fraction	P	All	D	0.1	NUREG-1727 (Ref. 6) Table C7.1 and NUREG/CR-6755 (Ref. 2), Section 3.5	NR	NR	NR	NR	NR
Time for source removal (d)	P	H-3	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	C-14	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Fe-55	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination

Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c]				Median
						1	2	3	4	
	P	Co-60	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ni-63	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sr-90	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Nb-94	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Tc-99	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Ag-108m	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Sb-125	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-134	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cs-137	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-152	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-154	D	52777	75th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Eu-155	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-238	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR

Input Parameters for Building Occupancy DCGL Determination										
Parameter (unit)	Type[a]	Radionuclide	Treatment[b]	Value or Distribution	Basis	Distribution's Statistical Parameters[c]				Median
						1	2	3	4	
	P	Pu-239	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Pu-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Am-241	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
	P	Cm-243	D	18240	25th percentile value of distribution in NUREG/CR-6755 (Ref. 2)	NR	NR	NR	NR	NR
Radionuclide concentration (pCi/m^2)	P	All	D	1	Assumed unit concentration	NR	NR	NR	NR	NR

Notes:

[a] P = physical, B = behavioral, M = metabolic (NUREG/CR-6697 (Ref. 3) Att. B, Table 4.3)
[b] D = deterministic, S = stochastic
[c] Statistical Parameters
Loguniform 1 = minimum, 2 = maximum
NR = Not Required
Input Correlations (used only if both input parameters use distributions): resuspension rate and deposition velocity = 0.9
Run Specifications:
Random seed = 1000
Number of Observations = 300
Number of Repetitions = 1
Dose Integrations = 5

References:

1. NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning," Volume 3: "Parameter Analysis, Draft Report for Comment," October 1999 (SAND99-2148).

2. NUREG/CR-6755, "Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code," February. 2002 (ANL/EAD/TM/02-1).

3. NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes," November 2000, (ANL/EAD/TM-98).

4. Eckerman, K.F., et al, "External Exposure to Radionuclides In Air, Water, And Soil," EPA 402-R-93-081, Federal Guidance Report No. 12 (FGR-12), U.S. EPA, 1993.

5. 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 (FGR-11), U.S. EPA, 1988.

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

Appendix 6J

Building Surface Area DCGL Results

Building Surface DCGL Results

Nuclide	Dose Conversion Factor (DCF) (mrem/yr per pCi/m^2)	DCGL (pCi/m^2)	DCGL (dpm/100cm^2)
H-3	1.6E-09	1.5E+10	3.4E+08
C-14	5.4E-08	4.6E+08	1.0E+07
Fe-55	1.4E-08	1.8E+09	4.0E+07
Co-60	3.1E-05	8.1E+05	1.8E+04
Ni-63	1.5E-08	1.7E+09	3.7E+07
Sr-90	4.0E-06	6.3E+06	1.4E+05
Nb-94	2.1E-05	1.2E+06	2.6E+04
Tc-99	3.9E-08	6.5E+08	1.4E+07
Ag-108m	2.2E-05	1.1E+06	2.5E+04
Sb-125	5.5E-06	4.5E+06	1.0E+05
Cs-134	1.9E-05	1.3E+06	2.9E+04
Cs-137	8.8E-06	2.8E+06	6.3E+04
Eu-152	1.5E-05	1.7E+06	3.7E+04
Eu-154	1.6E-05	1.6E+06	3.4E+04
Eu-155	8.5E-07	2.9E+07	6.5E+05
Pu-238	9.7E-05	2.6E+05	5.7E+03
Pu-239	1.1E-04	2.3E+05	5.1E+03
Pu-241	2.3E-06	1.1E+07	2.5E+05
Am-241	1.1E-04	2.2E+05	5.0E+03
Cm-243	7.7E-05	3.2E+05	7.2E+03

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

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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 (cubic yards)	Subtotal (cubic yards)	Total (cubic yards)	Source	Comments
Borrow	Concrete	Structures to Grade	13,105	17,705	69,805	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 SCFA	10,000	50,900		3	
		ISFSI soils in mid-parking lot	11,000			4	
		SCFA Removal	29,000			5	
		Detention Basin Excavation	900			5	
Fill	ABC	Building Voids	4,800	18,300	58,850	1	Based on information provided by Yankee.
		Shaping material (< 3 feet) for H&A Plan	10,800	Slopes		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.
		SCFA Below 3 feet	6,500			5	Does YAEC want to place ABC in SCFA?
	Soil	Service Building Foundation	350	24,050		2	
		Cap for H&A Plan	18,700			5	
		SCFA Upper 3 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.

ABC - asphalt, Brick, Concrete 6" or less

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

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

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

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:

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

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

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

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.

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

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.

$$C = \frac{1000\ S_o \rho_b}{[1 + (k_d \rho_b / n)]\ n} \qquad \text{Equation 4}$$

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 $5020 m^2$ 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]l	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 6N
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	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
R^2 = 1.0	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	Kd of Am in contaminated zone	-0.96	Loguniform	200	5000				4.47E02		
R^2 = 1.0	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	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
R^2 = 0.84	Fish transfer factor for C	0.67	Lognormal-n	10.8	1.1			8.98E04		1.03E05	1.03E05
Cm-243	Weathering removal constant of all vegetation	-0.81	Triangular	5.1	18	84			2.15E01		
R^2 = 0.99	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	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
R^2 = 1.0	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	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
R^2 = 0.91	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 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
Eu-154 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
Eu-155 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
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	Density of contaminated zone	0.62	Uniform	1.41	1.67			1.54E00		1.60E00	1.60E00
cellar hole	Irrigation	-0.58	Uniform	0.252	0.618			4.35E-01	3.43E-01		3.43E-01
R^2 = 0.98											
H-3	Depth of roots	-0.73	Uniform	0.3	3.8			2.05E00	1.17E00		1.17E00
graded	Thickness of contaminated zone	0.66	Uniform	0.15	3.8			1.98E00		2.89E00	2.89E00
R^2 = 0.97											
Nb-94 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
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 1	2	3	4	Mean	Percentile Values 25%	75%	Assigned Value
Pu-238	Kd of Pu in contaminated zone	-0.83	Loguniform	500	5000				8.88E02		
R^2 = 0.99	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	Kd of Pu in contaminated zone	-0.91	Loguniform	500	5000				8.88E02		
R^2 = 1.0	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	Kd of Am241 in contaminated zone	-0.95	Loguniform	200	5000				4.47E02		
R^2 = 1.0	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	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											
Sr-90	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
R^2 = 0.91	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	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
R^2 = 0.99	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

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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

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Appendix 6P

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):

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

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

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:

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

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 6Q

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

7 Update of Site-Specific Decommissioning Costs

7.1 Introduction

In accordance with 10CFR50.82(a)(9)(ii)(F) and Regulatory Guide 1.179, the site specific cost estimates and funding plans are provided. Regulatory Guide 1.179 discusses the details of the information to be presented.

The License Termination Plan (LTP) must:

Provide an estimate of the remaining decommissioning costs, and compare the estimated costs with the present funds set aside for decommissioning. The financial assurance instrument required by 10CFR50.75 (Reference 7-1) must be funded to the amount of the cost estimate. If there is a deficit in the present funding, the LTP must indicate the means for ensuring adequate funds to complete the decommissioning.

The decommissioning cost estimate should include an evaluation of the following cost elements:

- Cost assumptions used, including contingency
- Major decommissioning activities and tasks
- Unit cost factors
- Estimated decontamination and equipment and structure removal
- Estimated cost of radioactive waste disposal including disposal surcharges
- Estimated final survey costs
- Estimated total costs

The cost estimate should focus on the remaining work, detailed activity by activity, including costs of labor, materials, equipment, energy, and services.

During plant operations, YAEC sold the entire electrical output of the Yankee Nuclear Power Station (YNPS) to wholesale power purchase contracts (i.e., Power Contracts) with the ten New England utilities that collectively own 100% of the common equity of YAEC (the "Customers"). Over YNPS's operating life, YAEC recovered, and since the shutdown continues to recover, its costs of providing service (including the estimated costs of decommissioning YNPS) through a formula rate set forth in its Power Contracts. Collections for decommissioning have been placed in a trust established under Massachusetts law, with three separate funds—the Qualified Fund, a Non-Qualified A Fund, and a Non-Qualified B Fund.

The most recent cost estimate, submitted to the Federal Energy Regulatory Commission (FERC) on April 4, 2003, was prepared by Yankee Atomic Electric Company in accordance with 10 CFR 50.82(a)(8)(iii). [Reference 7-2] The assumed method of decommissioning anticipated a prompt decommissioning technique commonly referred to as DECON. FERC accepted the filing by its June 3, 2003 order, certified settlement on September 16, 2003 and granted approval on October 2, 2003.

In February 28, 2003, YAEC entered into a contract with a demolition contractor for completion of certain decommissioning activities including the completion of removing contaminated equipment and the demolition of structures above grade.

7.2 Decommissioning Cost Estimate

7.2.1 Cost Estimate Previously Docketed in Accordance with 10 CFR 50.82 and 10 CFR 50.75 Post Shutdown

The YNPS Decommissioning Plan was submitted to the NRC in December 1993 and was approved on February 14, 1995, and later became part of the FSAR. It described all activities associated with decommissioning the facility and included a cost study for those activities. The cost study was subsequently revised in October 1994 and updated in August 1995, December 1999, and most recently in April 2003.

In June 2001, YAEC elected to relocate pertinent information from the FSAR, including the cost estimate summary, to a PSDAR which conforms to the guidance in RG 1.185. This estimate was the basis for which YAEC collected its rates and its anticipated decommissioning expenses. This cost estimate was prepared by YAEC in accordance with 10CFR50.82(a)(8)(iii). It was prepared in sufficient detail to identify an activity by activity work breakdown complete with costs for radioactive waste, utility labor, contractor labor, energy, materials and equipment.

7.2.2 Summary of the Site Specific Decommissioning Cost Estimate

The current 2003 decommissioning cost estimate is based on the April 4, 2003, submittal to FERC. This estimate completely replaces and supercedes the 1999 “to-go” cost estimate. The filing was necessary due to cost increases due to continued DOE default in taking title of the spent fuel and GTCC waste, increased security measures, increased costs to implement transportable dry storage, additional waste disposal, the environmental site closure process, and investment market volatility. This most recent filing permits resumption of decommissioning and includes a fixed priced contract for the majority of remaining D&D activities as well as a market tracking mechanism to address investment market volatility. The filing also provides for a future filing requirement triggered by the outcome of the DOE litigation.

The D&D contractor work scope assumes YAEC continues to operate and maintain the spent fuel storage, hold the NRC license and provide oversight to the D&D contractor. Site-specific costs, such as insurance costs, property taxes, and decommissioning oversight costs, are the responsibility of YAEC. The D&D contractor is responsible for all other costs associated with decommissioning activities within the scope of the work with includes the removal of components and structures to grade elevation. An Integrated Project Schedule has been developed to mirror the costs expected to be expended during the completion of decommissioning. The schedules were also used to determine all other period dependent costs such as small tool allowances, health physics supplies, energy requirements and security. The activity dependent and period dependent costs were added together and a contingency applied to arrive at the total decommissioning cost estimate.

Specific factors assessed were:

1. Staffing
2. Labor - Dismantlement of Contaminated Areas
3. Labor - Dismantlement of Non-contaminated Areas
4. Preparation, Packaging and Transportation of LLW
5. Disposal of Class A LLW (excluding soil and asphalt)
6. Disposal of Class B and C LLW (excluding soil and asphalt)
7. Preparation, Packaging and Transportation of Non-Radioactive Waste
8. Disposal of Non-Radioactive Waste*
9. Building and Structure Demolition*
10. Soil and Asphalt Remediation and Disposal
11. Final Status Survey and License Termination Plan
12. Site Restoration*
13. Administration, General and Overhead
14. Materials & Services
15. Fees, Licenses and Permits
16. Salvage of Equipment and Components
17. Transition to Dry Storage Implementation*
18. Other (Health Physics Supplies, Small Tools, Decon and Removal Materials and Environmental Surveys)
19. Security (administration & enhanced security)
20. Long Term ISFSI Operation*
21. D&D of ISFSI*
22. Environmental Site Closure

*Included but not part of NRC required decommissioning activities.

The decommissioning cost estimate uses the approach identified in the LTP.

Table 7-1 identifies, as of 1/1/2003, that the remaining cost to complete NRC required decommissioning activities is $121.1 million (excluding contingency). The $121.1 million is comprised of $97.1 million for dismantlement and decontamination, $20.0 million for radioactive waste disposal, and $4.0 million for final status survey. This estimate is a subset of the FERC approved estimate (References 7-3 and 7-4) and excludes significant other expenditures approved in the FERC settlement such as decommissioning costs of $347.9 million incurred prior to 2003, a contingency amount, and long-term spent fuel storage.

In addition to the $121.1 million of NRC required decommissioning costs identified in Table 7-1, the FERC approved cost estimate includes: 1) contingency ($37.9 million); 2) long term spent fuel storage costs through 2022 ($129.2 million), and 3) site restoration ($0.3 million). While approved by the FERC rate settlement, Table 7-1 presents the long-term spent fuel storage, site restoration, and final status survey costs, separately from the D&D cost.

The decommissioning cost estimate figures are in 2003 dollars. Collections to the trust fund are based on the following economic assumption: "average annual inflation adjustment of 2.2% for costs that are not subject to escalation (i.e., excluding the demolition contractor contract, already incurred costs, and other costs that are not subject to inflation)." As set forth in Table 7-2, the comparison of estimated forward costs to trust funds on hand has been done on an explicit year-by-year model. As such, each year's statement of expenditures, income, and trust fund balances is in that year's current dollars.

Table 7-1

Actual and Projected Decommissioning Expenditures

($ Millions)

Cost Categories	Total
1992 – 2002 Dismantlement and Decontamination [A]	**$ 347.9**
2003 FERC Approved "To-go" Decommissioning Cost Estimate: 2003 – 2022	
Cost Elements [B]:	
(1) Dismantlement and Decontamination	$ 97.1
(2) Radioactive Waste Costs	$ 20.0
Subtotal Dismantlement and Decontamination and Waste (1+2)	$ 117.1
(3) Long Term SNF Storage*	$ 129.2
(4) Site Restoration*	$ 0.3
(5) Final Status Survey (FSS)	$ 4.0
Subtotal SNF Storage, Site Restoration and FSS (3+4+5)	$ 133.5
(6) Contingency	$ 37.9
Total 2003 FERC Approved Decommissioning "To-go" Cost Estimate 2003 – 2022	**$ 288.5**
Total Decommissioning Cost Including Incurred and Estimated Cost	**$ 636.4**

[A] 1992 – 2002 are stated in actual/nominal year dollars.
[B] "To-go" 2003 cost estimate is stated in year 2003 dollars.
* Included but not part of NRC required decommissioning activities.

Table 7-2

Decommissioning Trust Analysis

Total Funds (in $1000)

					Cash		
Period Ending	Contributions	Decom[1] Expenses	Earnings	Income Taxes	TOTAL	Non-Qualified	Qualified
31-Dec-02					64,042	8,062	55,979
31-Dec-03	32,453	(70,772)	2,959	(477)	28,205	3,407	24,798
31-Dec-04	55,634	(71,021)	1,286	(887)	13,217	5,300	7,917
31-Dec-05	55,634	(41,797)	1,205	(3,132)	25,127	8,317	16,810
31-Dec-06	14,005	(10,260)	1,872	(297)	30,447	1	30,446
31-Dec-07	14,005	(7,847)	2,555	1,774	40,935	1	40,934
31-Dec-08	14,005	(7,489)	3,328	178	50,958	1	50,957
31-Dec-09	14,005	(7,344)	4,076	(449)	61,247	1	61,246
31-Dec-10	14,005	(7,676)	4,832	(749)	71,660	0	71,660
31-Dec-11		(7,372)	5,098	(882)	68,505	0	68,505
31-Dec-12		(7,975)	4,839	(837)	64,532	0	64,532
31-Dec-13		(7,771)	4,548	(787)	60,523	0	60,523
31-Dec-14		(8,030)	4,238	(733)	55,998	0	55,998
31-Dec-15		(8,289)	3,889	(672)	50,926	0	50,926
31-Dec-16		(8,428)	3,503	(606)	45,396	0	45,396
31-Dec-17		(9,106)	3,063	(530)	38,824	0	38,824
31-Dec-18		(8,968)	2,576	(445)	31,986	0	31,986
31-Dec-19		(9,151)	1,850	(288)	24,397	0	24,397
31-Dec-20		(5,941)	1,286	(173)	19,569	0	19,569
31-Dec-21		(12,376)	703	(76)	7,820	0	7,820
31-Dec-22		(7,877)	73	(16)	0	0	0
31-Dec-23		0	0	0	0	0	0
	$213,748	($325,490)	$57,779	($10,082)			

TOTAL FUNDS AVAILABLE **($0)**

[1] Decommissioning expenses include contingency and escalation.

7.2.3 Dismantlement and Decontamination

The costs for the remaining dismantlement and decontamination activities include: utility oversight and project management, labor to remove contaminated and non-contaminated systems, structures and components, disposal of non-radiological waste, soil and asphalt disposal, materials, services, equipment, fees and permits, salvage credits, and spent fuel island transition. Also included are the costs associated with non-radiological remediation required by Federal and State agencies for such items as RCRA and TSCA closure, asbestos disposal, etc. These specific decommissioning activities which remain to be performed are described in Section 3 of this LTP. As provided in Table 7-1, the NRC-related decommissioning activities and funding amounts for those activities are separately identified.

7.2.4 Radiological Waste Disposal

Radiological waste disposal includes: preparation, packaging, transportation and disposal of all forms of low-level radioactive wastes. The quantity of radioactive waste remains bounded by the estimate of radioactive waste volume for a PWR provided in the FGEIS.

The large majority of the waste is Class A waste which is either sent to an approved waste processing facility or the Envirocare facility in Clive, Utah. The rates for these two facilities are comparable to or lower than the published rates for the Barnwell facility. The portion of the waste going to Barnwell consists mainly of Class B and C waste (e.g., resin liners).

7.2.5 Long-Term Spent Fuel Storage

In parallel with the final phase of decommissioning, an Independent Spent Fuel Storage Installation (ISFSI) was constructed on-site for the long-term storage of the spent nuclear fuel. Long-term fuel storage costs encompasses the completion of fuel transfer from the existing Spent Fuel Pit to the ISFSI, and ISFSI operational and maintenance expenditures from mid-year 2003 through 2022 when the Department of Energy (DOE) is assumed to honor its contract obligations for the spent fuel. ISFSI operations and maintenance include: YAEC oversight labor and benefits, insurance, regulatory fees, legal fees, maintenance materials, and other administrative and general expenditures.

7.2.6 Final Status Survey (FSS) and Site Restoration

The current estimate for site restoration includes the cost to perform the final site survey, perform final site grading and terminate the Part 50 license.

7.3 Decommissioning Funding

On July 25, 1990, YAEC submitted to the NRC a report as required by 10CFR50.75, indicating how reasonable assurance will be provided for funds to decommission the facility (Reference 7-5). The report described how YAEC has established an external sinking fund in 1981 to accumulate decommissioning funds. YAEC certified that each owner agreed to be financially responsible for its share of the decommissioning costs pursuant to the terms of the Power Contracts and Amendatory Agreements in accordance with the FERC regulations. These contracts have been filed with and approved by FERC. The Power Contracts and Amendatory Agreements were attached to the report.

On March 31, 2003, YAEC provided the most recent status report on the decommissioning fund to the NRC in accordance with 10CFR50.75 (Reference 7-6). This report restated the obligation that each wholesale power purchaser is responsible for its share of the facility decommissioning costs pursuant to the Power Contracts regardless of when the costs occur or the operational status of the facility.

Collections began in June 2003 at $32.5 million, increase to $55.6 million in years 2004 and 2005, and decrease to $14.0 million annually from 2006 through 2010.

All decommissioning activities are scheduled for completion by January 2006. Assuming a SAFSTOR condition, decommissioning trust fund expenditures would be minimized to dry fuel storage activities only. Therefore, YAEC forecasts sufficient funding will exist should a SAFSTOR condition occur during the period 2003 through 2005.

All spent fuel has been transferred to the ISFSI and the existing Spent Fuel Pit Building is expected to be decommissioned by the end of 2004. The long term spent fuel storage costs after 2004 consist of the operation, and maintenance of the ISFSI, as well as decommissioning of the ISFSI. As shown in Table 7-2, sufficient funding will exist, based on YAEC's assumption that the DOE will assume responsibility to complete spent fuel storage and removal by 2022.

Finally, as demonstrated in Table 7-2, YAEC shows that sufficient funds will be available from current assets and future contributions to complete decommissioning. The availability of ongoing contributions to the trust funds provides reasonable assurance that decommissioning costs will be paid when incurred. This assurance is further founded on the power contract obligations of the owners of YAEC. Pursuant to 10CFR50.75 and 10CFR50.82 regulations, YAEC has demonstrated a financial plan which includes adequate reserves for the entire decommissioning and ISFSI-related costs, which therefore meet the requirements for decommissioning costs associated with decommissioning and dismantlement as defined by these regulations.

7.4 References

7-1 Code of Federal Regulations, Title 10, part 50.75, “Reporting and Recordkeeping for Decommissioning Planning.”

7-2 Letter, YAEC to FERC, Filing Revisions to Yankee’s Wholesale Power Contract, dated April 4, 2003, Docket No.ER03-704-000.

7-3 Letter, FERC to YAEC, “Certification of Uncontested Offer of Settlement,” dated September 16, 2003, Docket No. ER03-704-000.

7-4 Letter, FERC to YAEC, “Order Approving Uncontested Settlement,” dated October 2, 2003, Docket No. ER03-704-000.

7-5 Letter, YAEC to USNRC, “Decommissioning Financial Assurance Certification Report,” dated July 25, 1990, BYR 90-102.

7-6 Letter, YAEC to USNRC, “Decommissioning Funding Assurance,” dated March 31, 2003, BYR 2003-016.

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8 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1 Introduction

8.1.1 Overview

A decommissioning environmental report (Reference 8-1), dated December 1993, was prepared for the YNPS site, in conjunction with the plant's Decommissioning Plan. This report concluded that the environmental impacts of decommissioning activities are small and bounded by the previously issued Final Generic Environmental Impact Statement (FGEIS) issued by the Nuclear Regulatory Commission as NUREG-0586 (Reference 8-2) and by the YNPS environmental assessment, associated with construction period recapture. In 1997, a License Termination Plan (LTP) was prepared and submitted to the NRC but was later withdrawn, following the release of MARSSIM guidance (Reference 8-3). In 2002, activities associated with the LTP restarted using MARSSIM and other updated guidance.

The purpose of this section of the LTP is to describe any new information on significant environmental impacts associated with site-specific license termination activities and to determine if these impacts are within the scope of the environmental impacts previously evaluated either generically or on a site-specific basis by:

1. the environmental impact statement developed in support of the original facility,
2. the environmental impacts described in conjunction with the Decommissioning Plan (and PSDAR) related to decommissioning activities, or
3. the Final Generic Environmental Impact Statement addressing decommissioning (NUREG-0586).

The NRC has issued guidance associated with the impacts of decommissioning, including Supplement 1 to NUREG-0586 (Reference 8-4). Supplement 1 to NUREG-0586 focuses on the impacts of decommissioning nuclear power reactors licensed by the NRC, unlike the 1988 FGEIS, which took a broad look at decommissioning of a variety of sites and activities.

Supplement 1 to NUREG-0586 is intended to consider, in a comprehensive manner, all aspects related to the radiological decommissioning of nuclear reactor facilities. Supplement 1 uses an approach that defines a measure of significance and severity of potential environmental impacts and an applicability of these impacts to a variety of facilities. The significance of an impact is described as being SMALL, MODERATE, or LARGE. The applicability of impacts is described as being generic or site-specific. These terms are clearly defined in Section 4 of Supplement 1 to NUREG-0586.

Table H-1, located in Appendix H to Supplement 1 of NUREG-0586, provides a listing of activities for which the NRC has generically determined that no environmental impacts exist. Because these activities have already been determined not to result in environmental impacts, no further review is required in connection with the LTP.

Table H-2 provides a summary of the decommissioning activities and associated environmental issues that have been determined to have *potential* impacts. As stated in Section 4.3 of Supplement 1 to the FGEIS, if these plant-specific impacts fall within the scope of the environmental impacts previously identified and evaluated by the NRC staff, these activities can be performed without further evaluation. The issues identified in Table H-2 to be evaluated for plant-specific impacts are:

- Onsite/offsite land use
- Water use
- Water quality
- Air quality
- Aquatic ecology
- Terrestrial ecology
- Threatened and endangered species
- Radiological
- Radiological accidents
- Occupational
- Socioeconomics
- Environmental justice
- Cultural impacts
- Aesthetics
- Noise
- Transportation
- Irretrievable resources.

According to Supplement 1 to NUREG-0586, the NRC assessed the impacts of each of these issues using data from previous studies and environmental reviews in addition to information obtained during site visits and provided by plants undergoing decommissioning. The NRC then examined the cumulative impacts of decommissioning activities and other past, present, and reasonably foreseeable future activities at the sites. After analyzing the issues, the NRC determined the impact of each and assigned a significance level (SMALL, MODERATE, or LARGE).

The NRC also determined whether the analysis of the environmental issues could be applied to all plants. Each environmental issue identified was assigned one of the following two categories: generic or site-specific.

Generic issues met the following three criteria:

1. The environmental impacts associated with the issue have been determined to apply to all plants, or, for some issues, to a group of plants of a specific size, specific locations, or having a specific type of cooling system or site characteristic.

2. A single significance criterion (SMALL, MODERATE, or LARGE) has been assigned to describe the impacts.

3. Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are likely not to be sufficiently beneficial to warrant implementation.

If one or more of the above criteria cannot be met, the issue is considered to be "site-specific" and a site-specific evaluation of the issue is required. Table 8-1 summarizes the NRC's findings with respect to applicability and impact of the identified environmental issues pertinent to decommissioning.

Decommissioning and license termination activities at YNPS fall within the range of activities evaluated for the FGEIS and NUREG-0586, Supplement 1. For those issues identified as "generic" in Table 8-1, the NRC's prior conclusions bound environmental impacts at YNPS from decommissioning and license termination.

The LTP addresses the issues identified in Table 8-1 as "site-specific." In addition, consistent with RG 1.179, the review focuses on any new information or significant environmental change associated with site-specific termination issues. Impacts associated with site-specific termination activities have been compared to previously analyzed decommissioning and termination activities, in this LTP and its references. The proposed termination activities related to the end use of the site do not result in significant environmental changes that are not bounded by the site-specific decommissioning activities described in the Decommissioning Plan, PSDAR, the FGEIS, or NUREG-0586.

Note that the review and conclusion in this Section relate only to activities and impacts associated with termination of the NRC license. YNPS is conducting other site characterization for non-radiological remediation and site restoration, which are not part of the license termination activities and are outside of the scope of NRC regulation. The non-radiological activities are addressed in an environmental closure plan that was submitted to the Massachusetts Department of Environmental Protection acting as the lead agency. Other agencies, such as the EPA, are also routinely involved in aspects of non-radiological site remediation.

8.1.2 Proposed Site Conditions at the Time of License Termination

The YNPS site is intended to be released for unrestricted use, under the radiological release criteria of 10CFR20.1402 (Reference 8-5) upon termination of its NRC license. Sections 3 and 4 of this LTP discuss in greater detail the activities that have been completed, those ongoing and remaining, and the proposed final state of the site.

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 that 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

8.1.3.2 Other Decommissioning Considerations

The PSDAR discusses other decommissioning considerations, including decontamination and dismantlement methods, storage and removal of spent fuel and GTCC waste, and site restoration.

8.1.3.3 General Decommissioning Activities Related to Removal of Radiological Components and Structures

Site structures and components are being removed using techniques and methods appropriate for the particular circumstances and are consistent with Decommissioning Work Packages. Openings in structures will typically be covered or sealed to minimize the spread of contamination. Components may be moved to an area for processing or volume reduction and/or packaging into containers, so that they can then be shipped to a processing facility for decontamination or to a low-level radioactive waste disposal facility. Buried contaminated components are being decontaminated to meet the free release criteria or are being excavated and removed for disposal.

8.1.3.3.1 Decontamination Methods

Contaminated systems and components are being removed and sent to an offsite processing facility or to a low-level radioactive waste disposal facility. Onsite decontamination of systems and components is generally limited to those activities needed to maintain personnel exposure ALARA, to expedite equipment removal, and to minimize the spread of contamination.

Application of coating and hand wiping are the preferred methods for stabilizing or removing loose surface contamination. If other methods are employed (e.g., grit blasting, high-pressure washing), airborne contamination control and waste processing systems are used, as necessary, to control and monitor any release of contamination.

Contaminated and activated concrete, as well as other contaminated materials, are being removed and sent to a low-level radioactive waste disposal facility. Concrete removal methods, such as scabbling and scarifying, will control concrete removal depth in order to minimize the waste volume produced. HEPA filtration is being used on dust and debris effluents in order to minimize the need for additional respiratory protection control measures. YAEC will consider new decommissioning techniques and technologies, as appropriate.

8.1.3.3.2 Dismantlement Methods

YAEC uses two basic dismantlement methods:

- Mechanical methods: Mechanical methods machine the surface of the material that is being cut. Typically, these methods are capable of cutting remotely without generating significant amounts of airborne contamination. This attribute makes mechanical methods attractive for removing most of the contaminated piping, components, and equipment.

- Thermal methods: Thermal methods melt or vaporize the surface of the material being cut. The cutting debris is transported from the cut region with a gas jet or water spray. Although thermal methods are more expedient than mechanical methods, they have large power requirements and generate airborne contamination when applied to contaminated systems in an air environment. However, thermal methods can be used with a cutting station and air filtration. For these reasons, application of thermal cutting methods on contaminated systems, structures or equipment is being restricted to areas that can be easily sealed, filtered, or maintained under water. Appropriate lead paint removal controls must also be implemented when using thermal cutting methods.

While these methods represent the most practicable and widely used decontamination methods available at this time, YAEC will consider new decontamination technologies if appropriate.

8.1.3.3.3 Special Programs

There are no special or unusual programs related to the decommissioning of YNPS. All procedures and processes used at YNPS are consistent with those considered in the FGEIS and its supplement.

8.1.3.3.4 Removal of LLW and Compaction or Incineration

LLW is being processed in accordance with plant procedures and sent to LLW disposal facilities. While no incineration will be performed onsite, YAEC may use an offsite licensed facility.

8.1.3.3.5 Soil Remediation

Soils and pavement are being surveyed and characterized in accordance with the site radiological characterization program. As necessary, soils, and pavement will be remediated (i.e., removed, processed and disposed of at a licensed facility) if determined to contain contamination levels above the site release criteria.

8.1.3.3.6 Processing and Disposal Site Locations

Currently, there are several facilities available for (1) processing of waste materials to achieve volume reduction prior to disposal or (2) disposal of low-level radioactive waste. These locations include: GTS Duratek – Barnwell, South Carolina; Envirocare – South Clive, Utah; and GTS Duratek – Oak Ridge, Tennessee.

8.1.3.3.7 Removal of Mixed Wastes

Mixed wastes are being managed according to all applicable federal and state regulations, including NRC handling, storage, and transportation regulations. Mixed wastes from YNPS are being transported only by authorized and licensed transporters and shipped only to authorized 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).

8.1.3.5 Conclusions Regarding Environmental Impact Included in the PSDAR

The PSDAR included a discussion of environmental impacts from decommissioning the YNPS. These conclusions were based largely upon the information provided in the YNPS Decommissioning Environmental Report (DER). The DER was based upon NUREG-0586, "Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities" and the site-specific environmental assessment from the re-capture of the construction time period.

The PSDAR concluded that the impacts due to decommissioning would be bounded by the previously issued environmental impacts statements. This was principally due to the following reasons:

- The postulated impacts associated with the method chosen, DECON, have already been considered in the FGEIS.
- There are no unique aspects of the plant or decommissioning techniques to be utilized that would invalidate the conclusions reached in the FGEIS.
- The methods to be employed to dismantle and decontaminate the site are standard construction-based techniques fully considered in the FGEIS.
- The site-specific person-rem estimate for all decommissioning activities has been conservatively calculated using methods similar to those used in the FGEIS.

Specifically, the review concluded that the YAEC decommissioning will result in generally positive environmental effects, in that:

- Radiological sources that create the potential for radiation exposure to site workers and the public will be eliminated.
- The site will be returned to a condition that will be acceptable for unrestricted use.
- The thermal impact on the Deerfield River from facility operations will be eliminated.
- Noise levels in the vicinity of the facility will be reduced.
- Hazardous material and chemicals will be removed.
- Local traffic will be reduced (fewer employees, contractors and materials shipments than required to support an operating nuclear power plant).

Furthermore, the YNPS decommissioning will be accomplished with no significant adverse environmental impacts in that:

- No site-specific factors pertaining to YNPS will alter the conclusions of the FGEIS.

- Radiation dose to the public will be minimal.
- Radiation dose to decommissioning workers will be a fraction of the operating exposure.
- Decommissioning is not an imminent health or safety problem and will generally have a positive environmental impact.

The Decommissioning Plan estimated the total radiation exposure impact for decommissioning to be 744 person-rem. This estimate was re-evaluated in 1996, resulting in a lower value of 580 person-rem (Reference 8-9). The actual exposure, through December 31, 2002, for decommissioning activities is 555 person-rem (Reference 8-10).

Radiation exposure due to transportation of radioactive waste has been conservatively estimated to be approximately 7 person-rem. This value is bounded by the FGEIS value of 100 person-rem of occupational exposure for transport of radioactive material.

Radiation exposure to offsite individuals for expected conditions, or from postulated accidents is bounded by the Environmental Protection Agency's Protective Action Guidelines and NRC regulations. The public exposure due to radiological effluents will continue to remain well below the 10CFRPart 20 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.

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 to be approximately 132,000 cubic feet. A review of the annual effluent reports filed with the NRC has determined that, through the end of 2002, 144,184 cubic feet of LLW has been shipped offsite for burial (Reference 8-9). 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 cubic feet (Reference 8-11). A final estimate for waste volume will be developed based upon the results of further characterization. The waste volume estimated to be generated by the YNPS decommissioning remains bounded by the FGEIS estimate for a reference PWR of 647,670 cubic feet.

Since the approval of the Decommissioning Plan and the issuance of the Decommissioning Environmental Report, YNPS has identified the presence of polychlorinated biphenyls (PCBs) from some paint coatings in soil. As in the case of radiologically contaminated lead paint, asbestos, and other hazardous materials, contaminated paint that contains PCBs will be managed according to all applicable federal and state regulations.

No significant environmental impacts are anticipated in the event that LLW is required to be temporarily stored onsite because adequate storage space exists and LLW storage will be in accordance with all applicable federal and state regulations.

The non-radiological environmental impacts from decommissioning are temporary and are not significant. The largest occupational risk associated with decommissioning YNPS is related to the risk of industrial accidents. The primary environmental effects are short term: small increases in noise levels and fugitive dust in the immediate vicinity of the site, as well as truck traffic to and from the site for hauling equipment and waste. No socioeconomic impacts, other than those associated with the cessation of operations (loss of jobs and taxes) have been identified. Also, no significant impacts to local culture, terrestrial or aquatic resources, such as the Sherman Reservoir and Deerfield River have been identified.

8.2 Analysis of Site-Specific Issues

8.2.1 Onsite-Offsite Land Uses

8.2.1.1 Onsite Land Uses

The environmental impacts associated with onsite land uses have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of onsite land uses is documented in Section 4.3.1 of Supplement 1 to NUREG-0586.

YNPS is located on a 2200 acre site, of which approximately 10 acres have been developed for plant use. Decommissioning activities involve the same areas used during initial construction and during operations. The use of a small fraction of the total site area land impacted by decommissioning and the re-use of areas used during initial construction are consistent with the NRC's assumptions in Supplement 1 to NUREG-0586, and thus there are no significant environmental impacts associated with YNPS decommissioning.

YAEC has identified no new information or significant environmental change associated with the site-specific termination activities related to the end use of the site.

8.2.1.2 Offsite Land Uses

Only areas within the existing site boundary will be used to support decommissioning and license termination activities (such as temporary storage areas and staging areas). As discussed previously in this section, and in detail in Section 5, isolation and control measures will be instituted to prevent the spread of contamination. These measures will also be monitored to ensure their effectiveness. Thus, no environmental impacts associated with the use of offsite lands are anticipated from YNPS decommissioning and license termination activities.

8.2.2 Water Use

The environmental impacts associated with water use, during decommissioning, have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of water use is documented in Section 4.3.2 of Supplement 1 to NUREG-0586.

During plant operation, an average of 0.4 million gallons of water per day from the Sherman Reservoir was used to cool plants systems. Water use was discussed in the "Environmental Assessment by the U.S. Nuclear Regulatory Commission, Related to the Request to Authorize Facility Decommissioning," dated December 14, 1994 (Reference 8-12). At that point in the decommissioning project, water usage was estimated to be less than 1% of the average water usage during operations.

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.

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.

<u>Worker transportation:</u> 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.

<u>Dismantling systems and removal of equipment:</u> 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.

Movement and open storage of materials onsite: Movement of equipment and open storage of materials during decommissioning may result in fugitive dust. Provisions as discussed in Section 3 and identified above can mitigate these effects. Thus, it is highly unlikely that particulate matter generated as a result of movement or storage of material onsite will be detectable offsite.

Demolition of buildings or structures: As discussed in the FGEIS, demolition of structures and buildings on the YNPS site may result in a temporary increase in fugitive dust. The controlled dismantlement and packaging of site components and structures will minimize the potential for fugitive dust from becoming an ambient air quality concern during decommissioning. Fugitive dust from demolition of buildings and structures generally involves large particles that settle quickly. Dust and smaller particles will be controlled using mitigation methods such as wet suppression. Thus, it is highly unlikely that particulate matter generated as a result of building or structure demolition will be detectable offsite.

Shipments of material to an offsite location: Material, debris, and equipment will be removed from the site during decommissioning. Although the remaining number of shipments to be sent during decommissioning is relatively large, these shipments are taking place over a couple of years, and thus the average number of shipments per day is relatively small. As stated in the FGEIS, it is unlikely that the emissions associated with the small number of daily shipments would be detectable offsite.

Air effluent released from the site is monitored in accordance with the Offsite Dose Calculation Manual (ODCM) which sets limits on doses caused by effluents, based upon the ALARA (as low as reasonably achievable) objectives of 10CFR50.34a, 10CFR50.36a, and Section IV.B.1 of Appendix I to 10CFR50. Effluents are reported annually to the NRC.

Based upon the above considerations, it has been determined that the conclusions of the FGEIS are applicable to YNPS, and decommissioning of YNPS will not noticeably affect offsite air quality. 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.

8.2.5 Aquatic Ecology

8.2.5.1 Activities Within the Operational Area

The environmental impacts associated with aquatic ecology for decommissioning activities within the operational area have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of aquatic ecology for activities within the operational area is documented in Section 4.3.5 of Supplement 1 to NUREG-0586. Any new wetland areas created as a result of the ISFSI construction will remain during decommissioning.

8.2.5.2 Activities Outside of the Operational Area

The FGEIS identifies generation of runoff due to ground disturbances and surface erosion as having the potential to impact aquatic resources. Provisions will be made to reduce surface erosion and runoff.

It is understood that decommissioning of shoreline and in-water structures has the potential to impact aquatic habitats and biota. YAEC will consult with regulatory and resource agencies to obtain permits and plan activities to minimize the duration and extent of these impacts. Regardless, impacts would be limited to those areas previously disturbed during construction and operation, and these areas would be expected to re-colonize as they did following initial construction. Thus, even considering the removal of shoreline and in-water structures, the impacts of decommissioning on aquatic ecology are minimal.

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.

8.2.6 Terrestrial Ecology

8.2.6.1 Activities Within the Operational Area

The environmental impacts of decommissioning associated with terrestrial ecology for activities within the operational area have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of terrestrial ecology for activities within the operational area is documented in Section 4.3.6 of Supplement 1 to the FGEIS.

8.2.6.2 Activities Outside the Operational Area

Only areas within the existing site boundary will be used to support decommissioning and license termination activities (such as temporary storage areas and staging areas). These areas are within those areas that were disturbed during initial construction. The FGEIS states that terrestrial habitats disturbed during the construction of the site often continue to be of low habitat quality during operation and decommissioning.

As discussed previously in this section, and in detail in Section 5, isolation and control measures will be instituted to prevent the spread of contamination, and these measures will be monitored to ensure their effectiveness. Because the YNPS site has been in active decommissioning since the decision to permanently close the facility was made, it is reasonable to conclude that areas disturbed during the construction and operation of the plant have not become new sensitive areas with respect to terrestrial biota. Thus, no environmental impacts associated with the use of offsite lands are anticipated from YNPS decommissioning and license termination activities related to the end use of the site.

8.2.7 Threatened and Endangered Species

While the YNPS site consists of over 2000 acres of land, only a small fraction consisting of approximately 10 acres has been developed for plant use. During planning and construction of the independent spent fuel storage facility (which is adjacent to the areas being decommissioned), the Natural Heritage and Endangered Species Program (NHESP), an agency of the Department of Fisheries, Wildlife, and Environmental Law Enforcement, was contacted to review impacts. This review included activities associated with the installation of the ISFSI pad, road improvements, and improvements to the present storm water system. The NHESP had determined that the activities do not occur within the actual habitat of a state-protected rare wildlife species (Reference 8-15).

However, during recent field surveys to complete the mapping and to characterize natural communities, a late-larval spring salamander (*Gyrinophilus porphyriticus*) was identified on the YAEC property. It was found at the northeast end of the property, in one of the headwater channels of Wheeler Brook and very near the property line (which is also the Massachusetts/Vermont State Line) in a forestry management area.

The spring salamander is a species of Special Concern in Massachusetts. This status means that it is a species that has either been documented as suffering a decline that could threaten the species if allowed to continue or which occurs in small numbers or with a very restricted distribution in the state.

The implications of this species occurring on the site are fairly minimal since (1) this species occurs in a habitat that is already provided a high level of protection under the Massachusetts Wetlands Protection Act and (2) spring salamanders hardly ever stray far from their home streams. Standard best forestry practices include limiting stream crossings, retain tree cover adjacent to streams, and prohibit activities (such as skidding or brush piling) in streams. No evidence of any past forest management activities affecting habitat in this stream was observed during the survey and future forest management activities are not expected to require alteration of the stream.

Only a very small section of Wheeler Brook comes close to the industrial portion of the property, less than 200 feet. In that area, Wheeler Brook is generally of lower gradient than preferred by the spring salamander. Therefore, decommissioning and license termination activities at the YNPS site are not expected to affect the spring salamander.

Thus, decommissioning and license termination activities at the YNPS site does not adversely impact threatened or endangered species.

8.2.8 Radiological

8.2.8.1 Activities Resulting in Occupational Doses to Workers

The environmental impacts associated with radiological activities resulting in occupational doses to worker have been determined by the NRC to be generically applicable with a SMALL impact, because of the existence of guidance regulating doses to workers (10CFR20) which remain applicable to the YNPS. The NRC's analysis of the environmental impacts of radiological activities resulting in occupational doses to workers is documented in Section 4.3.8 of Supplement 1 to NUREG-0586.

8.2.8.2 Activities Resulting in Doses to the Public

The environmental impacts associated with radiological activities resulting in doses to the public have been determined by the NRC to be generically applicable with a SMALL impact, because of the existence of guidance regulating and documenting doses to members of the public (10CFR20). The NRC's analysis of the environmental impacts of radiological activities resulting in doses to the public is documented in Section 4.3.8 of Supplement 1 to NUREG-0586. 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.

Potential doses to the public following license termination are not covered by the Supplement to the FGEIS but were evaluated during promulgation of rulemaking for the radiological criteria for license termination (10CFR20.1402). The basis for public health and safety considerations associated with the license termination rule is discussed in NUREG-1496.

8.2.9 Radiological Accidents

The environmental impacts associated with radiological accidents have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of radiological accidents is documented in Section 4.3.9 of Supplement 1 to NUREG-0586. 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.

The NRC concluded that radiological impacts, due to accidents, are considered to be undetectable and non-destabilizing, in the National Environmental Policy Act (NEPA) sense, if the doses remain within regulatory limits. The YNPS FSAR provides a summary of the evaluation of plant transients that have a potential impact on both occupational and public safety and health. The risk of accidents resulting in a significant radiological release during decommissioning activities is considerably less than during plant operations.

The analysis of decommissioning events includes all phases of decommissioning activities: decontamination, dismantlement, packaging, storage, radioactive materials handling, and license termination activities (including final status surveys). The following radiological events were identified as having the potential to affect public health and safety:

- Decommissioning activity events.
- Loss of support system events, including loss of offsite power, cooling water and compressed air.
- Fire and explosion events.
- External events.
- Spent fuel storage events.

YAEC requested and received an exemption from the emergency preparedness requirements of 10CFR50.47 (Reference 8-16); however, approval of the exemption request was predicated on the absence of any accidents where the offsite dose consequences could exceed the EPA protective action guidelines (PAGs). Releases resulting from accidents postulated in the decommissioning accident analysis were evaluated using the EPA PAGs as an upper limit and found to be bounded by this criterion. Use of the EPA PAGs as an administrative limit also ensure that postulated accident offsite doses are significantly less than the 10CFR100 reference values.

Thus, because the dose consequences resulting from radiological events, identified as having the potential to affect public health and safety, are below the EPA PAGs and the criteria of 10CFR100, the associated impacts on the environment are minimal.

8.2.10 Occupational Issues

The environmental impacts of occupational issues have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of occupational issues is documented in Section 4.3.10 of Supplement 1 to NUREG-0586. 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.

As Supplement 1 to the FGEIS indicates, the Occupational Safety and Health Act of 1970 was enacted to protect the health of workers, and applicable regulations are administered by the Occupational Safety and Heath Administration (OSHA). YNPS is subject to 29 CFR 1910 and 1926 for worker health and safety protection under OSHA regulations. These requirements are implemented under existing plant programs and procedures.

8.2.11 Socioeconomic Impacts

The environmental impacts of socioeconomic impacts have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of socioeconomic impacts is documented in Section 4.3.12 of Supplement 1 to NUREG-0586.

The impacts that are observed by the community are primarily those resulting from plant closure rather than from decommissioning, although some decommissioning activities began very shortly after closure. These impacts occur either through changes in employment levels and local demands for housing and infrastructure, or through decline of the local tax base and the ability of local government entities to provide public services. Supplement 1 to NUREG-0586 states that decommissioning, itself, has no impact on the tax base and no detectable impact on the demand for public services.

Additionally Supplement 1 to NUREG-0586 concludes that the effects of employment changes on population growth are:

1. not detectable if population changes (reductions or increases) are less than 3% per year,
2. detectable but not destabilizing if the population change is between 3% and 5%, and
3. de-stabilizing if the population change is greater than 5% per year.

Table 8-2 shows the change in population over the last two decades. For the decade 1990 to 2000, which includes the period of shutdown and partial decommissioning, the overall change in population in the vicinity of the site was a 5% decrease over this ten-year period. The average annual population change, based upon the data from 1990 and 2000, does not exceed the NRC's threshold of 3%, and thus signifies that the changes are neither detectable nor destabilizing. Thus no significant socioeconomic impacts are associated with YNPS decommissioning and license termination activities related to the end use of the site.

8.2.12 Environmental Justice

Radioactive waste shipments, from the site to an interstate highway, traverse a six-county area including the following counties: Berkshire, Franklin, and Hampshire in Massachusetts; Bennington in Vermont; and Columbia and Rensselear in New York. The total population of this area is approximately 611,400 people. The number of minority (non-white) persons is about 7% of the total population, and the percentage of people below the poverty level is about 9% of the total population. The area is generally rural along the shipping routes. These data were derived from the Bureau of the Census 2000 Reports (References 8-17, 8-18, and 8-19).

Environmental Justice was addressed by the NRC during the review and approval of the YNPS Decommissioning Plan (Reference 8-20). The NRC concluded that there are no significant environmental impacts associated with the proposed decommissioning activity that would have a significant effect on the quality of the human environment. The NRC included consideration of the transportation of radioactive wastes from the YNPS site to the interstate transportation corridor (both rail and highway) and concluded that such transportation will not have a disproportionate effect on minority or low income populations.

These conclusions remain valid. The types of decommissioning and license termination activities, conducted or planned at YNPS, are not significantly different than those described in the Decommissioning Plan and the assumptions related to affected populations remain valid, considering the information from the 2000 Census, presented above. Thus, there are no environmental justice impacts introduced by decommissioning or license termination.

8.2.13 Cultural and Historic Resource Impacts

8.2.13.1 Activities Within the Operational Area

The environmental impacts associated with cultural and historic resource impacts from activities within the operational area have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of cultural and historic resource impacts from activities within the operational area is documented in Section 4.3.14 of Supplement 1 to NUREG-0586. 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.

8.2.13.2 Activities Outside the Operational Area

An independent review of files from the Massachusetts Historic Commission, the Massachusetts State Archives, and the State House Library was performed to determine the significance of buildings and areas in the vicinity of the YNPS site. There are no historic or cultural resources which are listed in the National Register of Historic Places within five miles of the plant (References 8-21, 8-22, 8-23 and 8-24). The Hoosac Tunnel, just beyond five miles of the site to the southwest, is designated as a National Register Property. The closest locale considered to have local historic significance is the Brigham Young birthplace monument located in Whitingham, Vermont, approximately five miles northeast of YNPS. The Sherman Dam Development District (including individual structures) and the Monroe Bridge Development/ Glassine Paper Company/Deerfield Dam District (including individual structures) have been deemed eligible to be on the State Register of Historic Places. The YNPS structures have not been identified as a historic site or asset, and decommissioning and license termination activities will not involve or impact any site or structure listed in the State Register of Historic Places.

8.2.14 Aesthetics

The environmental impacts associated with aesthetics have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of aesthetics is documented in Section 4.3.15 of Supplement 1 to NUREG-0586.

Aesthetic resources include natural and man-made landscapes and the way the two are integrated. As a part of construction and operation of the facility, the landscape was previously altered. Decommissioning activities will be conducted onsite, both inside and outside of existing buildings (in the case of dismantlement or shipping activities). The NRC has concluded that any visual intrusion resulting from decommissioning will be temporary and would serve to reduce the aesthetic impacts of the facility. YAEC will use best management practices to control many of the potentially adverse impacts of decommissioning on aesthetics (such as dust and noise), as discussed in other sections.

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.

8.2.15 Noise

The environmental impacts associated with noise have been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of noise is documented in Section 4.3.16 of Supplement 1 to NUREG-0586.

As stated in the "Environmental Assessment by the U.S. Nuclear Regulatory Commission, Related to the Request to Authorize Facility Decommissioning," dated December 14, 1994, decommissioning activities at YNPS will add minimally to the ambient noise of the surrounding environment, beyond the security fence.

Decommissioning activities will, in general, be intermittent and temporary, and limited to a relatively small portion of the entire YNPS site. Noise is attenuated by the mature forests surrounding the plant. During fall and winter, absence of foliage will allow some additional transmission of noise, and, to the areas north and west of the plant, the presence of Sherman Reservoir will allow some transmission of noise over the water before attenuation by forest. However, a review of wildlife species existing in the vicinity of the plant indicates an assemblage consistent with that found within similar regional habitats. This indicates that the noise levels generated at YNPS during decommissioning have added only minimally to the ambient noise levels and have had a negligible effect on the vicinity and the environment. 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.

8.2.16 Transportation

The environmental issue of transportation has been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of transportation is documented in Section 4.3.17 of Supplement 1 to NUREG-0586.

The number of shipments and the volume of waste shipped are greater during decommissioning than during operations. In Supplement 1 to the FGEIS, the public health and safety impacts of transportation of radioactive wastes are evaluated on the basis of compliance with regulation. The NRC has concluded that compliance with regulation is adequate to protect the public against unreasonable risk from the transportation of radioactive materials. The supplement to the FGEIS notes that the evaluation leading to that conclusion was based, in part, on information in NUREG-0170 and that recent re-evaluation of transportation risks, using updated information and assessment tools, found that risks are lower than those estimated in NUREG-0170. Because YNPS will comply with all applicable regulations when shipping radioactive wastes from decommissioning, the effects of transportation of that radioactive waste on public health and safety are considered to be neither detectable nor destabilizing.

Non-radiological impacts of transportation include increased traffic and wear and tear on roadways. Because the average number of shipments from the site will be relatively small, there will be no significant effect on traffic flow or road wear. Additionally, because of the industry's emphasis on training and adherence to established procedures, truck accident rates for activities at nuclear facilities has been lower than the national average for similar activities. The NRC has concluded that impacts of transportation accidents would neither be detectable nor destabilizing.

Thus, transportation of wastes associated with the YNPS decommissioning and license termination activities do not present significant adverse impacts.

8.2.17 Irretrievable Resources

The environmental issue of irretrievable resources has been determined by the NRC to be generically applicable with a SMALL impact. The NRC's analysis of the environmental impacts of irretrievable resources is documented in Section 4.3.18 of Supplement 1 to NUREG-0586.

Supplement 1 to the FGEIS indicates that land associated with a site released for unrestricted use is available for other uses, regardless of whether or not the decommissioning process returned the land to an open space or to an industrial complex. Thus the land resource would not be considered "irretrievable." The Supplement to the FGEIS evaluated other irretrievable resources such as the materials/equipment used to decontaminate the facilities and the fuel used for construction machinery and for transporting wastes and concluded these resources are minor.

Thus, the impact of decommissioning and license termination on irretrievable resources is neither detectable nor destabilizing.

8.3 References

8-1 YNPS Decommissioning Environmental Report., dated December 1993.

8-2 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated August 1988.

8-3 NUREG-1575, NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual," Revision 1, dated August 2000.

8-4 Supplement 1 to NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," dated November 2002.

8-5 Title 10 to the Code of Federal Regulations, Subpart E to Part 20.

8-6 Attachment E to the "Contract for the Performance of Demolition and Disposal and Related Services, By and Between DEMCO, Inc. and Yankee Atomic Electric Company," dated February 28, 2003

8-7 Yankee Nuclear Power Station Decommissioning Plan, Revision 0.0.

8-8 Regulatory Guide 1.185, "Standard Format and Content for Post-shutdown Decommissioning Activities Report," dated July 2000.

8-9 YNPS Post-Shutdown Decommissioning Activities Report, dated June 2003.

8-10 USNRC Atomic Safety and Licensing Board Docket No. 50-029-DCOM, Supplemental Affidavit of Russell A. Mellor, September 3, 1996.

8-11 Memorandum RP-03-045 from Greg Babineau to Jim Kay, dated November 19, 2003.

8-12 "Environmental Assessment by the U.S. Nuclear Regulatory Commission, Related to the Request to Authorize Facility Decommissioning," dated December 14, 1994.

8-13 R. Janson (EPA) to J. A. Kay (YNPS), dated July 29, 2003, "Issuance of NPDES Permit No. MA0004367.

8-14 Offsite Dose Calculation Manual, Revision 15.

8-15 NHESP 99-5798, "Installation of an on-site storage pad, road improvements, and improvements to present storm water system," dated November 30, 1999, from Patricia Huckery, NHESP Wetlands Environmental Review to the Rowe Conservation Commission.

8-16 NYR 92-144, Exemption From 10CFR Part 50 - Appendix E - Emergency Preparedness Training Exercises at the Yankee Nuclear Power Station (TAC No. M83415), M. B. Fairtile (USNRC) to J. M. Grant, July 24, 1992.

8-17 "Massachusetts: 2000, Summary Population and Housing Characteristics," U.S. Department of Commerce, issued September 2002

8-18 "Vermont: 2000, Summary Population and Housing Characteristics," U.S. Department of Commerce, issued October 2002.

8-19 "New York: 2000, Summary Population and Housing Characteristics," U.S. Department of Commerce, issued July 2002.

8-20 NRC Letter, "Order Approving the Decommissioning Plan and Authorizing Decommissioning of the Yankee Nuclear Power Station," dated February 14, 1995.

8-21 State Register of Historic Places/1988, Massachusetts Historical Commission.

8-22 BYR 2003-063 "Project Notification Form, Request for Determination of No Adverse Effect," from Gregg Demers and John McTigue, ERM, to Brona Simon, Massachusetts Historical Commission, dated July 11, 2003.

8-23 National Register Survey Books, Bennington County and Windham County Listings, Vermont Division of Historic Preservation.

8-24 Deerfield River Project, Deerfield River, Vermont and Massachusetts—Information for the Initial Stage of Consultation FERC Project No. 2323, Volumes I and II, New England Power Company, March 1988.

Table 8-1

Summary of Environmental Impacts from Decommissioning

Issue	Generic	Impact	LTP Section
Onsite-Offsite Land Uses			8.2.1
• Onsite Land Uses	Yes	Small	8.2.1.1
• Offsite Land Uses	No	Site-Specific	8.2.1.2
Water Use	Yes	Small	8.2.2
Water Quality	Yes	Small	8.2.3
Air Quality	Yes	Small	8.2.4
Aquatic Ecology			8.2.5
• Activities within the operational area*	Yes	Small	8.2.5.1
• Activities outside the operational area	No	Site-Specific	8.2.5.2
Terrestrial Ecology			8.2.6
• Within the operational area	Yes	Small	8.2.6.1
• Outside the operational area	No	Site-Specific	8.2.6.2
Threatened and Endangered Species	No	Site-Specific	8.2.7
Radiological			8.2.8
• Activities resulting in occupational doses to workers	Yes	Small	8.2.8.1
• Activities resulting in doses to the public	Yes	Small	8.2.8.2
Radiological accidents	Yes	Small	8.2.9
Occupational issues	Yes	Small	8.2.10
Cost	N/A	N/A†	7
Socioeconomic	Yes	Small	8.2.11
Environmental Justice	No	Site-Specific	8.2.12
Cultural and Historic Resource Impacts			8.2.13
• Activities within the operational area	Yes	Small	8.2.13.1
• Activities outside the operational area	No	Site-Specific	8.2.13.2
Aesthetics	Yes	Small	8.2.14
Noise	Yes	Small	8.2.15
Transportation	Yes	Small	8.2.16
Irretrievable Resources	Yes	Small	8.2.17

* The operational area is defined as the portion of the plant site where most or all of the site activities occur, such as reactor operation, materials and equipment storage, parking, substation operation, facility service, and maintenance. This includes areas within the protected area fences, the intake, discharge, cooling, and associated structures as well as surrounding paved, graveled, maintained landscape, or other maintained areas.

† A decommissioning cost assessment is not a specific National Environmental Policy Act (NEPA) requirement.

Table 8-2

Population Changes in the Vicinity of YNPS

Location	1980 (Ref 8-1)	1990 (Ref 8-1)	2000 (Ref 8-17 & 8-18)	% change in decade before shutdown	% change in decade including shutdown
Massachusetts					
Adams	10,381	9,445	8,809	-9%	-7%
Clarksburg	1,871	1,745	1,686	-7%	-3%
Florida	730	732	676	0%	-8%
North Adams	18,063	16,797	14,681	-7%	-13%
Savoy	644	634	705	-2%	11%
Buckland	1,864	1,928	1,996	3%	4%
Charlemont	1,149	1,249	1,358	9%	9%
Colrain	1,552	1,757	1,813	13%	3%
Hawley	280	317	336	13%	6%
Heath	482	716	805	49%	12%
Monroe	179	115	93	-36%	-19%
Rowe	336	387	351	15%	-9%
Vermont					
Halifax	488	782	782	60%	0%
Whitingham	1,043	1,298	1,298	24%	0%
Wilmington	1,808	1,968	2,225	9%	13%
Readsboro	638	762	809	19%	6%
Stamford	773	773	813	0%	5%
Overall	42,281	41,405	39,236	-2%	-5%