January 13, 2000
MN-00-004 GAZ-00-003
Proposed Change No. 210

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

Reference: (a) License No. DPR-36 (Docket No. 50-309)
(b) MYAPC Letter to USNRC "Updated Draft Copy of Maine Yankee's License Termination Plan (MYLTP), dated November 15, 1999 (MN-99-32)

Subject: Maine Yankee's License Termination Plan (MYLTP)

In accordance with 10 CFR 50.82(a)(9), Maine Yankee hereby submits its License Termination Plan. This License Termination Plan demonstrates that the remainder of decommissioning activities will be performed in accordance with Title 10 Code of Federal Regulations, will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment.

Maine Yankee hereby submits, pursuant to 10 CFR 50.90, an application to amend the License to add a license condition which requires Maine Yankee to implement and maintain in effect all provisions of the License Termination Plan.

Attachment I provides the background and reason for the proposed change, a description of the proposed change, a no significant hazards consideration determination, and an environmental impact consideration determination. Attachment II provides a copy of the affected Facility Operating License No. DPR-36 page with changes annotated. Attachment III provides the Maine Yankee License Termination Plan for NRC review and approval in accordance with 10 CFR 50.82. In accordance with 10 CFR 50.82(a)(9)(i), the MYLTP is being submitted as a supplement to the FSAR. Therefore, pursuant to 10 CFR 50.4(b)(6), ten additional copies of the License Termination Plan are provided.

Maine Yankee has included in the MYLTP a preface section entitled "Beyond Regulatory Requirements - A Non-Technical Summary." This section is being transmitted for NRC information only. It reflects the preliminary results of the continuing dialogue which has been established between Maine Yankee and the various MYLTP stakeholders. This section of the LTP is not intended to support NRC's review of the MYLTP pursuant to 10 CFR 50.82 and 10 CFR 20 Subpart E.

ADMIN RECORD

1/405

This change does not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in the margin of safety.

This proposed change has undergone an Independent Safety Review. The Independent Review and Audit Committee has also reviewed this proposed change. A representative of the State of Maine is being informed of this request by a copy of this letter.

If you have any questions, please contact us.

Very truly yours,

George A. Zinke
Director, Nuclear Safety and
Regulatory Affairs

Attachments

c: Mr. H. J. Miller, NRC Regional Administrator, Region I
Mr. C. J. Paperiello, NRC Director, Office of Nuclear Material Safety and Safeguards
Mr. M. K. Webb, NRR Project Manager
Mr. J. T. Greeves, Director, Division of Waste Management
Mr. M. C. Roberts, NRC Region I
Mr. C.L. Pittiglio, NMSS Project Manager, Decommissioning
Mr. M.T. Masnik, Section Chief, Project Directorate IV & Decommissioning
Mr. P. J. Dostie, State of Maine, Division of Health Engineering
Mr. U. Vanags, State of Maine, Nuclear Safety Advisior
Mr. D. Rothstein, USEPA Region I, Office of Regional Counsel

STATE OF MAINE

Then personally appeared before me, George A. Zinke, who being duly sworn did state that he is the Director, Nuclear Safety and Regulatory Affairs of Maine Yankee Atomic Power Company, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Maine Yankee Atomic Power Company, and that the statements therein are true to the best of his knowledge and belief.

_____

Notary Public

ATTACHMENT I
DESCRIPTION OF CHANGES
PROPOSED CHANGE NO. 210

BACKGROUND

On August 6. 1997 the Maine Yankee Atomic Power Co. Board of Directors voted to permanently cease operations of Maine Yankee and begin the decommissioning process. On August 7, 1997, Maine Yankee submitted certifications of permanent cessation of power operations and permanent removal of fuel from the reactor in accordance with 10 CFR 50.82(a)(1). On August 27, 1997, Maine Yankee submitted its Post Shutdown Decommissioning Activities Report in accordance with 10 CFR 50.82(a)(4). In this report, Maine Yankee signaled its intention to decontaminate and dismantle the plant in a manner that results in the prompt removal of the existing nuclear plant. This approach is known as the DECON alternative. In its PSDAR, Maine Yankee provided a description and schedule of planned decommissioning activities, an estimate of expected costs and a discussion providing the reasons for concluding that the environmental impacts associated with the site-specific decommissioning activities are bounded by appropriate previously issued environmental impact statements. A public meeting was held on November 6, 1997 to solicit public comments on the PSDAR. On November 3, 1998, Maine Yankee submitted its Site Specific Cost Estimate and updated the PSDAR.

Pursuant to 10 CFR 50.82(a)(9), nuclear power reactor licensees are required to submit a license termination plan prior to or along with their application for termination of license. Pursuant to 10 CFR 50.82(a)(10), the Commission shall approve the plan if it demonstrates that the remainder of decommissioning activities will be performed in accordance with the regulations in 10 CFR Chapter I, will not be inimical to the common defense and security or to the health and safety of the public, will not have a significant effect on the quality of the environment, and after notice to interested persons. Accordingly, Maine Yankee is hereby submitting its License Termination Plan for NRC review and approval. This plan is being submitted as a supplement to the FSAR in accordance with 10 CFR 50.82(a)(9)(i). Maine Yankee is not submitting at this time its application for termination of license.

Pursuant to 10 CFR 50.82(a)(10), the Commission's approval of the plan shall be executed by license amendment, subject to such conditions and limitations as it deems appropriate and necessary. This approval will authorize the implementation of the license termination plan. Attached is a proposed change to the license authorizing the implementation of the license termination plan and providing appropriate and necessary conditions regarding the licensees authority to make changes to it without prior NRC review and approval.

Sort	Origin Doc	Origin Section	Author	Year	Reference	AR#	Comments	Action	PDFs
426	RI/FS	CRA	Kaiser-Hill	1999	1998 Study of the Preble's Meadow Jumping Mouse at the Rocky Flats Environmental Technology Site. Appendix B, 1998 Annual Wildlife Survey Report for the Rocky Flats Environmental Technology Site, Kaiser-Hill Company, L.L.C., Rocky Flats Environ	Have .pdf	On CD (J. Nelson)	Jody Nelson has and will submit to Hollies	
360	RI/FS	CRA	MacDonald, D.D., T. Berger, K. Wood, J. Brown, T. Johnsen, M.L. Haines, K. Brydges, M.J. MacDonald, S.L. Smith, and D.P. Shaw	1999	A Compendium of Environmental Quality Benchmarks. GBEI/EC-99-001. ISBN 0-662-28624-3. Prepared for Environment Canada, Van	Have .pdf	pdf_Ryan	Black Binder	
642	HRR		MYAPC	1999	Maine Yankee License Termination Plan, November	Have hard copy and .pdf		Rebecca found it	W: /AR Documents/MYLTP Sections 1 through 9 .pdfs plus PREFACE.pdf and TRANSMITTAL.pdf
159	RIFS	7	Pitten, F.A, G. Müller, and P. König	1999	Risk Assessment of Former Military Base Contaminated with Organoarsenic-based Warfare Agents: Uptake of Arsenic by Terrestrial Plants, Sci. Total Environ. 226:237-245.		CU-Boulder Library Science RA565.S365		

DESCRIPTION OF CHANGES

Maine Yankee proposes to amend the license to incorporate a new License Condition: 2.B.(9). Attachment II provides a copy of the affected Facility Operating License No. DPR-36 page with the proposed changes annotated.

"2.B.(9) License Termination

The licensee shall implement and maintain in effect all provisions of the approved License Termination Plan submitted on January 13, 2000 as supplemented and as approved in the SER dated_____. The licensee may make changes to the License Termination Plan without prior approval provided the proposed changes do not:

(a) Involve an unreviewed safety question or changes to the technical specifications as specified in 10 CFR 50.59; or
(b) Violate the requirements of 10 CFR 50.82(a)(6)
(c) Increase the specified derived concentration guideline levels (DCGL's) or area factors
(d) Increase the probability (α) of making a Type I decision error
(e) Increase the action thresholds associated with survey unit classification investigations. (Investigation Levels)

The licensee shall submit an updated License Termination Plan in accordance with 10 CFR 50.71(e)."

A method is proposed to allow the licensee to make changes to the License Termination Plan without prior NRC approval similar to the flexibility afforded licensees to make changes to the facility or procedures as described in the safety analysis report. This method includes five change criteria elements. The first two elements regarding an unreviewed safety question and 50.82(a)(6) are clearly established in current regulation. The third element regarding DCGL's and area factors is established to control the primary outputs from the dose models. The DCGL's for the Soil, the Standing Building, Activated Concrete and Embedded/Buried Piping are provided in License Termination Plan Section 6.6.1, 6.7.2 Table 13, 6.8.1 Table 15, and 6.9 respectively. The DCGL associated with concrete rubble is described below in the next paragraph. The area factors are described in License Termination Plan Sections 6.4 Table 8, 6.6.1(d) Table 11, and 6.7.3 Table 14. The fourth element limits the probability of releasing a survey unit which contains residual radioactivity above the release criterion. This value is discussed in License Termination Plan Section 5.4.2. The fifth element regarding investigation level sets a limit on the action thresholds which would trigger an investigation. These thresholds are specified in License Termination Plan Section 5.6, Table 6.

The DCGL associated with concrete rubble is described in License Termination Plan Section 6.2.5. The total activity in a given volume of concrete rubble (1929 m^3) is the basic compliance DCGL and is called the "Fundamental DCGL." The Fundamental DCGL is directly linked to dose modeling and is the underlying basis for compliance with the unrestricted use criteria. The DCGL for rubble which may not be increased with out prior NRC approval is the Fundamental DCGL. In LTP Section 6.2.5, the Generic DCGL is calculated to be 113,500 dpm/100 cm^2 gross beta. Consistent with the Generic gross beta DCGL assumptions, the total gross beta activity for 1929 m^2 of concrete surface area is 5.1 uCi/m^2. Since this area represents 1929 m^3 of concrete under the generic assumption of 1 m^2/m^3, the Fundamental DCGL can be expressed as 5.1 uCi/m^3. The gross beta DCGL's listed in LTP Section 6.2.5 Table 5 may be changed without prior NRC review as long as the Fundamental DCGL of 5.1 uCi/m^3 gross beta is demonstrated to be met. If measurements other than gross beta are used, such as in-situ gamma spectroscopy, it may be more appropriate to directly apply the Fundamental DCGL as opposed to the gross beta DCGL.

SIGNIFICANT HAZARDS EVALUATION

The proposed change to the Technical Specifications, has been evaluated against the standards of 10 CFR 50.92 and has been determined not to involve a significant hazards consideration. An evaluation against these standards is provided below:

The proposed change does not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated.

   The requested license amendment does not authorize any plant activities beyond that allowed by 10 CFR Chapter I or beyond that considered in the DSAR. The bounding accident described in the DSAR for potential airborne activity is the postulated resin cask drop accident in the Low Level Radioactive Waste Storage Building. This accident is expected to contain more potential airborne activity than can be released from other decommissioning events. The radionuclide distribution assumed for the spent resin cask has more transuranics (the major dose contributor) than the distribution in the components involved in other decommissioning accidents. The accidents considered in the DSAR include: 1) Explosion of Liquid Petroleum Gas (LPG) Leaked from a Front End Loader or Forklift, 2) Explosion of Oxyacetylene During Segmenting of the Reactor Vessel Shelf, 3) Release of Radioactivity from the RCS Decontamination Ion Exchange Resins, 4) Gross Leak During In-Situ Decontamination, 5) Segmentation of RCS Piping with Unremoved Contamination, 6) Fire Involving Contaminated Clothing or Combustible Waste, 7) Loss of Local Airborne Contamination Control During Blasting or Jackhammer Operations, 8) Temporary Loss of Services, 9) Dropping of Contaminated Concrete Rubble, 10) Natural Phenomena and 11) Transportation Accidents. The probabilities and consequences for these accidents are estimated in the basis documentation for DSAR Section 7. No

systems, structures, or components that could initiate or be required to mitigate the consequences of an accident are affected by the proposed change in any way not previously evaluated in the DSAR. Since Maine Yankee does not exceed the salient parameters associated with the plant referenced in the basis documentation in any material respects, it is concluded that these probabilities and consequences are not increased. Therefore, the proposed change to the Maine Yankee License does not involve any increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   The requested license amendment does not authorize any plant activities which could precipitate or result in any accidents beyond that considered in the DSAR. The accidents previously evaluated in the DSAR are described above. These accidents are described in the basis documentation for DSAR Section 7. The proposed change does not affect plant systems, structures, or components in any way not previously evaluated in the DSAR. Since Maine Yankee does not exceed the salient parameters associated with the plant referenced in the basis documentation in any material respects, it is concluded that these accidents appropriately bound the kinds of accidents possible during decommissioning. Therefore, the proposed change to the Maine Yankee License would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

   The margin of safety defined in Maine Yankee's license basis for the consequences of decommissioning accidents has been established as the margin between the bounding decommissioning accident and the dose limits associated with the need for emergency plan offsite protection, namely the Environmental Protection Agency Protective Action Guidelines EPA-PAGs. As described above the bounding decommissioning accident is the postulated resin cask drop accident in the Low Level Radioactive Waste Storage Building. Since the bounding decommissioning accident is expected to contain more potential airborne activity than can be released from other decommissioning events and since the radionuclide distribution assumed for the spent resin cask has more transuranics (the major dose contributor) than the distribution in the components involved in other decommissioning accidents, the margin of safety associated with the consequences of decommissioning accidents can not be reduced. The margin of safety defined in the statements of consideration for the final rule on the Radiological Criteria for License Termination is described as the margin between the 100 mrem/yr public dose limit established in 10 CFR 20.1301 for licensed operation and the 25 mrem/yr dose limit to the average member of the critical group at a site considered acceptable for unrestricted use. This margin of safety accounts for the potential effect of multiple sources of radiation exposure to the critical group. Since the license termination plan was designed to comply

with the radiological criteria for license termination for unrestricted use, the margin of safety can not be reduced. Therefore, the proposed changes to the Maine Yankee License would not involve a significant reduction in any margin of safety.

Conclusion

Maine Yankee has concluded that the proposed change to the Technical Specifications does not involve a significant hazards consideration as defined by 10 CFR 50.92.

ENVIRONMENTAL IMPACT DETERMINATION

This amendment request meets the criteria specified in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements to perform an environmental assessment or to prepare an environmental impact statement. The specific criteria of 10 CFR 51.22(c)(9) are discussed below:

1. The amendment involves no significant hazards consideration. As demonstrated above, this requested amendment does not involve any significant hazards considerations.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The environmental impacts associated with doses to members of the public as a result of decommissioning activities and site release for unrestricted use were considered in the Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (NUREG-0586) and the Generic Environmental Impact Statement in Support of the Rulemaking on Radiological Criteria for License Termination (NUREG-1496). In Maine Yankee's Post Shutdown Decommissioning Activities Report, Maine Yankee concluded that the environmental impacts associated with the site-specific decommissioning activities will be bounded by appropriate previously issued environmental impact statements. In particular, the decommissioning activities covered by the License Termination Plan will result in radiation doses to the public below a comparable level when the plant was operating. Radiation dose to the public will be minimal. The release of effluents will continue to be controlled by plant procedures throughout decommissioning. Maine Yankee will continue to operate in accordance with the Offsite Dose Calculation Manual (ODCM) during the decommissioning activities. In addition, because of the decay of short-lived radionuclides, the types of nuclides which could potentially be released in effluent have decreased.

3. There is no significant increase in individual or cumulative occupational radiation worker exposure. The environmental impacts associated with individual or cumulative occupational radiation exposure were also considered in the Generic Environmental Impact Statements referred to above. It was similarly concluded that these GEIS appropriately bounded the environmental impacts associated with the Maine Yankee site-specific decommissioning activities. Radiation dose to decommissioning workers will be a small fraction of the operating experience. The total decommissioning dose including exposure from decontamination and dismantlement activities and the exposure during transportation of the low-level wastes is estimated to be less that the total dose that was found acceptable for decommissioning the reference PWR in the GEIS on decommissioning of nuclear

facilities. Radiation protection principles used during plant operation remain in effect during decommissioning to ensure that protective techniques, clothing, and breathing apparatus are used as appropriate.

On the basis of the preceding discussion, it has been concluded that the proposed amendment is acceptable and meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9), and thus no environmental impact statement or environmental assessment need to be prepared in connection with issuance of this amendment.

ATTACHMENT II
PROPOSED CHANGE NO. 210

Maine Yankee
Proposed Technical Specification
Pages 3-3

2.B.(7) This amended license is subject to the following conditions for protection of the environment:

(f) Deleted

(g) Deleted

2.B.(8) This amended license is effective as of the date of issuance and shall expire at midnight October 21, 2008.

2.B.(9) License Termination

The licensee shall implement and maintain in effect all provisions of the approved License Termination Plan submitted on January 13, 2000 as supplemented and as approved in the SER dated_____. The licensee may make changes to the License Termination Plan without prior approval provided the proposed changes do not:

(a) Involve an unreviewed safety question or changes to the technical specifications as specified in 10 CFR 50.59; or
(b) Violate the requirements of 10 CFR 50.82(a)(6)
(c) Increase the specified derived concentration guideline levels (DCGL's) or area factors
(d) Increase the probability (α) of making a Type I decision error
(e) Increase the action thresholds associated with survey unit classification investigations. (Investigation Levels)

The licensee shall submit an updated License Termination Plan in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by A. Giambusso

A. Giambusso, Deputy Director
For Reactor Projects
Directorate of License

Attachments:
Appendices A & B - Technical Specifications

Date of Issuance: June 29, 1973

ATTACHMENT III
PROPOSED CHANGE NO. 210

MAINE YANKEE LICENSE TERMINATION PLAN

MAINE YANKEE

LTP PREFACE

BEYOND REGULATORY REQUIREMENTS - A NON-TECHNICAL SUMMARY

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P.0 Beyond Regulatory Requirements - A Non-Technical Summary

P.1 Introduction

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

The feedback has generally indicated a desire for Maine Yankee to go beyond regulatory requirements (including ALARA) in reducing residual radiation exposure on-site. At the same time, virtually all commenters recognized that it was a difficult task to decide where to draw the line and that it is unrealistic to expect elimination of all residual radioactivity.

This section of the LTP has two purposes:

- To discuss key elements of the LTP while lending perspective with respect to public health and safety, and
- To review the steps beyond regulatory requirements Maine Yankee is proposing to implement in being responsive to stakeholder feedback.

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

P.2 Background and Definitions

NRC regulations require that decommissioning nuclear facilities clean up residual radioactivity (i.e., plant derived radioactive contamination above natural background radiation) so that any member of the public would receive no more than a 25 millirem (mrem) dose over a year's period of time. This is total dose from any exposure pathway (e.g., drinking water, food, etc.).

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

In addition to the NRC's 25 mrem site release limit, the NRC also requires that the residual radioactivity be ALARA - "As Low As Reasonably Achievable". Using NRC guidance, "reasonably achievable" is determined by the amount of dose reduction achieved compared to the cost of additional dose reduction.

The Environmental Protection Agency (EPA) has also issued site release guidance for facilities other than commercial nuclear power plants. Their criterion is risk-based rather than dose-based.

Without accounting for radioactive decay, the EPA calculates a "surrogate" limit of 15 mrem/year which, when decay is accounted for results in guidance in excess of 30 mrem/year. The EPA also fosters an additional criterion of 4 mrem/year due to groundwater ingestion. The EPA does not have an ALARA standard.

A limit in the range of 25-30 mrem/year above background is such a low value that it cannot be directly measured, particularly when you consider that total radiation exposure is the sum of many different exposure pathways such as eating, drinking and direct exposure.

In order to demonstrate compliance with this dose limit, one must convert it to a surrogate value that can be directly measured. The value is called the DCGL - Derived Concentration Guideline Level. The DCGL is a limit for residual radioactive contamination levels in soil, buildings, etc. that, when put into a computer model to account for all exposure pathways, will result in doses less than a pre-defined limit (e.g., 25 mrem/year).

In order to identify the exposure pathways, one must answer the question "Who receives the 25 mrem/year?". The answer is found in regulatory guidance which requires the dose calculations to model the "average member of the group reasonably expected to receive the greatest exposure". In other words, a hypothetical, conservative scenario is created which includes theoretical individuals who could receive more radiation dose than could be expected for a member of the public.

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

All of the exposure pathways applicable to the resident farmer scenario are considered in the computer model. This model results in the calculation of the DCGL. The DCGL is a value that can be directly measured. For instance, the residual contamination on a building wall may be 20,000 dpm/100 cm^2. The term "dpm" stands for "disintegrations per minute" and is the number of radioactive atoms that decay in a minute. The "100 centimeter squared" provides an area over which the measurement is made. If the DCGL for this building example is 36,000 dpm/100 cm^2 (eg. Containment Building) then, under MARSSIM, we can be assured that radiation exposure due to this portion of the building, combined with the remainder of the site, will be below 25 mrem/year.

Once the dose limit is converted to a readily measurable value, the question is "How should one measure it in a widely recognized manner with high confidence that the site meets the limit for release?" To solve this problem, various government agencies including the NRC, the EPA, the Department of Energy and the Department of Defense spent a number of years pooling their resources to come up with a solution. They developed a method that would guarantee, on a rigorous statistical basis, that any site areas to be released meet the site release criteria. These

methods were published in December, 1997 under the title "Multi-Agency Radiation Survey and Site Investigation Manual", or MARSSIM for short. NRC and EPA recommend, and Maine Yankee has committed to, the use of MARSSIM.

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

The LTP's primary purpose is to demonstrate compliance with the NRC's 25 mrem ALARA requirement. Surprisingly, due to conservatisms employed in demonstrating compliance, it has little relevance to the ultimate site cleanup level which will be much lower.

Under NRC guidance and MARSSIM, Maine Yankee assumes that the site and buildings are contaminated. We further assume the contamination is at the DCGL limit. (Remember that the DCGL is that measured value that ensures that dose to a member of the maximally exposed group is less than 25 mrem/year.)

In reality, some areas will have no contamination and many areas that are contaminated will be below the DCGL limits. Using the containment as an example, LTP Section 6, Table 5 indicates that the containment DCGL is 36,000 dpm/100 cm^2.

- Nonetheless, there will be some areas of containment that will be higher than the DCGL - approximately 8-9% of the surface area. In those cases, Maine Yankee will scabble (i.e., remove a layer of concrete - typically ¼ inch) to lower the radioactive contamination level. Since the vast majority of contamination is contained in the first millimeter of a wall (see Section 2.5.2(f), pg. 2.43, Table 5), scabbling will result in lowering the contamination level far below the DCGL (in our example, to the 10,000-20,000 dpm/100 cm^2 range or lower).

- Most areas of containment are below or much below the DCGL to begin with. A rough estimate would put 70% well below the DCGL.

In general, remediating higher contamination levels combined with pre-existing low contamination levels will result in actual contamination levels being a medium to small fraction of DCGLs. Recognizing that contamination levels lower than DCGLs translate directly into lower doses, we can also say that dose to the resident farmer will be a medium to small fraction of 25 mrem/year. In this sense, the LTP and DCGLs are very conservative assumptions only useful for proving prior to decommissioning that Maine Yankee's approach will meet regulatory requirements.

P.4 Relationship Between Resident Farmer Dose and Actual Public Dose

Just as the LTP DCGLs are upper limits which don't reflect reality and just as the LTP DCGLs bear no relationship to the true dose to the resident farmer after decommissioning, so does the resident farmer dose assumption in the LTP bear no relationship to the true public dose since the

resident farmer is not representative of the likely end use of the Maine Yankee site.

The Maine Yankee site consists of 800+ acres of land, virtually all of which is non-contaminated. Of that acreage, only approximately one acre following decommissioning will contain some minimal level of residual contamination.

The preferred use of the one acre is as a gas generated power plant. Should this happen, we can estimate the worst case dose to a worker in the facility by assuming that the worker spends 8 hours a day standing in the most contaminated location of the power plant. The worker's occupational dose works out to be about 0.002 mrem/year - about 10,000 times lower than the NRC's required limit of 25 mrem/year.

Should the one acre in question simply be opened for public access following decommissioning, expected uses may include recreation and similar activities. Assuming that a nearby resident decides to walk their dog every day for 4 hours on the one acre of land. The dose to this member of the public is about 0.001 mrem/year - about 20,000 times below the NRC's required limit of 25 mrem/year.

P.5 Comparison of On-Site Rubblization With Other Fill Materials

The technical and dose calculation aspects of rubblization are discussed in Section 6.

Even at the LTP DCGLs, the amount of radioactive contamination remaining in concrete rubble is quite small. In fact, approximately 99.9% of radioactive material will be removed from the site during the course of remediation. The residual contamination in the rubble to be used as fill will be approximately 0.3 curies (see below).

Maine Yankee estimated the dose effects from filling the foundation hole with other common materials that contain naturally occurring radioactivity. We examined filling the hole with granite (a substance common in Maine) and filling the hole with fly ash (a byproduct of some fossil fueled power plants that is unregulated by the EPA and commonly used in road materials). Either approach results in resident farmer doses well in excess of NRC requirements.

P.6 Summary of Dose/Residual Contamination Information

To lend perspective, the information in the previous several sections is summarized below.

TABLE 1	Dose (mrem/yr)
Fly Ash Fill (resident farmer)	2300
Granite Fill (resident farmer)	300

LTP (compliance - resident farmer)	25
LTP (actual - resident farmer)	8 -12
Industrial - actual	0.002
Recreational - actual	0.001

Quantity of Radioactive Material On-Site

3,000,000 Ci. - no fuel
30,000 Ci. - no GTCC
300 Ci. - no vessel

0.3 Ci. - after remediation

Reduction of 99.9 % of radioactive material

P.7 Performing Beyond Regulatory Requirements

As noted above, just by meeting the federal requirements and taking into account actual use of the site for industrial or other public purposes, true occupational or public dose is thousands of times lower than NRC requirements.

On the other hand, stakeholder perception and comfort is an important element of a successful decommissioning. Since the beginning of decommissioning, Maine Yankee has incorporated "responsiveness to the community" as a key criterion in its mission statement. To meet that mission, Maine Yankee has purposely provided draft LTP information months in advance to all interested stakeholder groups in order to solicit, understand and act upon stakeholder viewpoints prior to submitting the LTP to the NRC.

One consistent sentiment stands out: "do what you can to go beyond NRC requirements, including ALARA, in restoring the Maine Yankee site".

Maine Yankee agrees.

In order to achieve a consensus about the end state of the site, Maine Yankee proposes four additional measures which will substantially lower the dose to the public from the remaining 0.1% of the radiation and provide even greater protection of the environment.

First, as recommended by one member of the Technical Advisory Panel, Maine Yankee proposes, through encapsulation, to achieve a dose of 10 mrem or less for all pathways and 4 mrem or less for groundwater pathway. This will bring the site remediation to a final state considerably below the NRC and EPA standards, and will meet or better the EPA's separate groundwater criterion.

Maine Yankee proposes to inject the concrete rubble which has been decontaminated to below the DCGL levels but greater than the minimum detectable activity (MDA) level (i.e., not capable of going to a local landfill) with "flowable fill" to create a solid mass in the existing building foundations. This additional step will virtually eliminate groundwater as a pathway for radionuclides. These areas will then be capped with soil and vegetated. Importantly, by creating a solid mass and virtually eliminating ground water pathways, Maine Yankee will achieve a clean

up of the site with a dose of less than 10 mrem for all pathways and less than 4 mrem to groundwater pathway.

Second, Maine Yankee proposes to reuse much of the crushed clean concrete (i.e., concrete below the MDA and capable of going to a local landfill) as fill. Use of this clean concrete as fill will have the environmentally responsible effect of eliminating the need to import clean fill for the site and it will also avoid using hundreds of thousands of cubic feet of scarce Maine landfill capacity for clean concrete generated by the project.

Third, the Maine Yankee site currently consists of more than 800 acres, of which only a few acres south of the Ferry Road contain low levels of radioactivity requiring remediation. However, in response to the suggestions of state officials to provide redundant protection to future users of the site, Maine Yankee is proposing to convey certain property south of the Ferry Road with a deed restriction ensuring that only commercial and/or industrial activities are permitted in the particular areas indicated. This restriction could run with the land for an agreed upon period, factoring in decay of radionuclides, so as not to burden the land in perpetuity. Such deed restriction will also prohibit well drilling for drinking water or other domestic purposes. The real estate is served by public water and we would expect any future occupants to derive their water from the public supply. (In determining whether we will meet the 10/4 mrem dose to the public, we will not take credit for the deed restriction. It will, however, have the real effect of further reducing dose.)

Fourth, Maine Yankee proposes that the state participate in the final site survey by taking independent measurements (and split samples, as appropriate) to verify final status survey activities consistent with Maine Yankee's decommissioning schedule. Measurement techniques, protocols, MDAs, quality assurance and instrumentation would be in accordance with the NRC approved License Termination Plan. The measurement results shall be analyzed by utilizing the NRC approved dose model, as adjusted for use with a solid mass. As always, the state inspectors may also accompany the NRC inspectors on any of their site verification inspections.

Taking these four additional steps of dose reduction, encapsulation, deed restriction and independent state verification should enhance public confidence in the clean up process and hasten the redevelopment of the site for the benefit of the state and the community. Moreover, while Maine Yankee is proposing to achieve a clean up to the 10/4 mrem dose level, it expects, by taking these measures, to achieve a much lower dose, although the final dose cannot be known with certainty until the decommissioning work is complete.

P.8 Additional Information

As discussed at the outset, this Section of the LTP is provided for stakeholders other than the NRC. It is intended to provide a point of reference and perspective on license termination issues associated with public health and safety. As important, it provides documentation of the additional steps Maine Yankee will be taking to address stakeholder feedback on site cleanup.

We recognize that it is not possible here to address each issue in the level of detail that would be satisfying to all stakeholders. Additional information is available through several means: raising questions during meetings of Maine Yankee's Community Advisory Panel, reviewing Maine Yankee's web site at www. maineyankee.com, written correspondence via mail or e-mail, or simply calling us. Please feel free to use whatever communications means is available and we'll

do our best to answer your question.

MAINE YANKEE

LTP SECTION 1

GENERAL INFORMATION

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ATTACHMENT A Documentation In Support of Request for Release Of Site Lands

1.0 GENERAL INFORMATION

1.1 Introduction

This License Termination Plan (LTP) has been prepared for the Maine Yankee Atomic Power Company (MYAPC) nuclear power plant; located at 321 Old Ferry Road, Wiscasset Maine, 04578. In accordance with requirements of 10CFR50.82(a)(9), it has been prepared and submitted as a supplement to the MYAPC Defueled Safety Analysis Report (DSAR), and is intended to support application for amendment of license number DPR-36; docket number 50-309. An application for amendment of license is being provided to facilitate authorization/approval of the LTP as required by 10CFR50.82 (a)(9).

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

1.2 Operating History

The plant is owned by a consortium of 11 New England electric utilities representing consumers in Maine, New Hampshire, Vermont, Massachusetts, Connecticut and Rhode Island. It began commercial operation in December 1972 under Atomic Energy Commission Docket no. 50-309 License No. OL-FP DPR36, and permanently ceased operation in December 1996. Over its lifetime, the plant operated for a total of approximately 16 effective full power years based on its rated thermal power. The MYAPC board of directors voted to permanently cease further operation, and decommission the plant in August 1997.

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

The Final Safety Analysis Report (FSAR) was revised to reflect the permanently defueled plant condition and was re-titled "Defueled Safety Analysis Report" (DSAR). The DSAR was submitted to the NRC on February 6, 1998. Additional licensing basis documents were also revised and submitted to reflect the plant's defueled condition (Defueled Security Plan, Fire Protection Plan, QA Plan, Training Plan and Emergency Plan).

On August 27, 1997, MYAPC submitted the Post Shutdown Decommissioning Activities Report (PSDAR) and discussion on Environmental Impact. On November 6, 1997, a public meeting was held in Wiscasset to hear public comments on the PSDAR. On

November 3, 1998, MYAPC submitted the Site-Specific Decommissioning Cost Estimate along with a PSDAR Update.

1.3 Plant Description

The plant is a three-loop pressurized water reactor with a power rating of 2,700 Megawatts thermal. It is located on an 820-acre site in Lincoln County Wiscasset, Maine. It has a Nuclear Steam Supply System supplied by Asea Brown Boveri/Combustion Engineering. The secondary plant consists of three Asea Brown Boveri turbines, one high pressure and two low pressure, coupled with a 950 MVA Westinghouse electric generator and associated auxiliary systems. The site also includes ancillary facilities used to support normal plant operations. These facilities consist of warehouses, administrative office buildings, security structures, an environmental sampling complex, a substation and a fire protection system.

1.4 LTP Submittal Change, Early Release of Land

1.4.1 LTP submittal and changes

MYAPC is submitting this LTP as an amendment to the DSAR.

Changes requiring NRC approval will be submitted via submittal of an updated LTP in accordance with 10 CFR 50.71(e).

The licensee may make changes to the LTP without prior approval provided the proposed changes do not:

a. Involve an unreviewed safety question or changes to the technical specifications as specified in 10 CFR 50.59; or

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

c. Increase the specified derived concentration guideline levels (DCGL's) or area factors

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

e. Increase the action thresholds associated with survey unit classification investigations. (Investigation Levels)

1.4.2 Phased Release and License Termination

MYAPC will make DSAR changes to the site boundary footprints to allow

unrestricted release and license termination of parcels of property. The following process will be used for making these changes:

a. Following the completion of LTP activities in a given area, MYAPC will provide to the NRC a report covering the area which it seeks to release from the Part 50 license. This report will contain the information which the NRC needs to make its 10 CFR 50. 82(a)(11) determination and will include:

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

2. A certification that the remaining dismantlement activities described in the license termination plan have been performed.

3. FSS results for the area.

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

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

6. An evaluation of the impact on the following license programs for the site lands remaining within the domain of the Part 50 license: ODCM, Emergency Plan, Security Plan, Fire Protection Plan, QA Plan, Training Plan, and Post Shutdown Decommissioning Activities Report.

Upon satisfactory NRC review, the NRC will provide a duly authorized certification or license amendment to MYAPC that the NRC has made the required 10 CFR 50.82(a)(11) determination regarding the area to be released from the Part 50 license and that the area is henceforth released from the Part 50 license. This certification or license amendment will carry the same authority as that associated with terminating a license under 10CFR50.82(a)(11).

Once an area is so released, it is understood that the NRC will not require any additional surveys or decontamination of these areas by MYAPC in response to future NRC criteria or standards, or third party survey results,

unless, pursuant to 10 CFR 20.1401(c), the NRC determines that the criteria of this subpart were not met and residual activity remaining at the site could result in significant threat to public health and safety

1.5 Plan Description

1.5.1 General Information

This Section gives an introduction, operating history, plant description, of MYAPC in addition description of the LTP change process and the phased release of land. Each of the seven (7) 10CFR 50.82 (a) (9) (ii) required sections are briefly described

1.5.2 Site Characterization

This section summarizes the radiological surveys that have been conducted to characterize the nature and extent of contamination at MY. A site radiological characterization was performed during November 1997 through March 1998. This resulted in a Site Characterization Survey Report dated April 1998. From March through July 1999, supplemental site characterization activities were conducted. Additional charaterization has been completed and will be performed as required during the term of the decommissioning project. The site characterization results will be used to identify areas of the site that are likely to require remediation, and to plan remediation strategies, and to support final status survey and dose assessment activities.

1.5.3 Identification of remaining site dismantlement activities

This section presents the sequence of dismantlement and decontamination (D&D) activities for the remaining systems, structures, and components at MY.

The overall project schedule defines the current status and remaining activities. Four phases of site dismantlement have been defined. As part of preparing the site, Phase 1 removes structures to increase the free area needed for large vehicles and equipment. Phase 2 initiates activities for commodity dismantlement and removal and structure decontamination. Phase 3 consists primarily of demolition activities as well as site restoration activities. Phase 4 consists of the movement of spent fuel to dry storage. These phases, in some cases, run in parallel.

The strategies for disposal of waste generated during decommissioning are included in the MY Decommissioning Waste Management Plan. The plan addresses
issues such as; estimates of the quantity of radioactive material to be released, control mechanisms, and radioactive waste characterization.

The approach to licensing the Independent Spent Fuel Storage Installation (ISFSI) is to maintain the 10CFR Part 50 license (encompasses Part 70 and Part 30).

1.5.4 Remediation Plans

The methods used to reduce the levels of radioactivity to meet the radiological release criteria of 10CFR20.1402 (Radiological Criteria for Unrestricted Use) are described in this section. The calculations used to verify that the residual activity levels have been reduced to levels that are as low as reasonably achievable ALARA are presented. These calculations, and the applied methodology generally conform to the guidance provided in Draft Regulatory Guide DG-4006 (Demonstrating Compliance with the Radiological Criteria for License Termination).

This section describes the areas on site that will be subject to remediation (as necessary) and identifies the methods that will be used.

1.5.5 Final Status Survey (FSS)

The FSS section of this LTP describes the methods that will be used by MYAPC to demonstrate that residual contamination levels at MY have been reduced to levels below the site release criteria. The derived concentration guideline (DCGL) is calculated in Section 6 of this LTP and represents the residual contamination levels that will result in a Total Effective Dose Equivalent (TEDE) to the average member of the critical population group that is less than 25 mrem per year. The methods for conducting the final status survey generally follow the guidance in Draft Regulatory Guide 4006. NUREG – 1575 (Multi-Agency Radiation Survey and Site Investigation Manual [MARSSIM]) is also used to the extent it is referenced in DG-4006 as appropriate. Additional sections of NUREG - 1575 are followed as required for specific applications. The FSS plan describes methodology for the division of the site into survey units, the classification of survey areas, and the requirement that all survey units meet the DCGL with a 95 % confidence level. Management controls over all aspects of the project are discussed in detail, including quality assurance, data processing, and final status survey reports.

1.5.6 Compliance with the radiological criteria for license termination

This section describes the methods used for conducting a dose assessment to develop the DCGLs for demonstrating compliance with the unrestricted use criteria in Subpart E of 10CFR20.

"Radiological Criteria for Unrestricted Use," allows termination/amendment of a license and release of a site for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent to an average member of a critical group that does not exceed 25 millirem per year and the residual radioactivity has been reduced to levels that are ALARA.

1.5.7 Update of the Site-Specific Decommissioning Costs

This section provides an updated estimate of remaining decommissioning costs and a comparison of these estimated costs with the present funds set aside for decommissioning. A site-specific decommissioning cost analysis was prepared by TLG Services in October of 1997. Subsequent to that, a revision to the decommissioning cost estimate was presented in the MY Site Specific Decommissioning Cost Estimate, dated November 1, 1998. As decommissioning activities are initiated and completed, the actual costs will be compared against the estimates previously submitted.

1.5.8 Supplement to the Environmental Report

This section demonstrates that decommissioning activities will be accomplished with no significant adverse environmental impacts. Decommissioning and license termination/ amendment activities remain bounded by previously issued environmental impact statements and Generic Environmental Impact Statement (GEIS).

The MY PSDAR was submitted as required by 10CFR 50.82 (a) (4) (i). That report provides a discussion of reasons for concluding that the environmental impacts associated with the site-specific decommissioning activities will be bounded by appropriate previously issued environmental impact statements. This LTP does not describe any activities that are not bounded by the PSDAR and the referenced GEIS.

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

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

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

APPENDIX A

Documentation In Support of Request for Release Of Site Lands

I. Description of Area Boundaries

II. Certification of Completion of Dismantlement Activities

III. Final Radiation Survey Results

IV. Recontamination Evaluation and Controls

V. Evaluation of impacts on Exclusion Area

VI. Evaluation of impact on License Programs

MAINE YANKEE

LTP SECTION 2

SITE CHARACTERIZATION

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ATTACHMENT B **Characterization Data**

ATTACHMENT C **Discussion of GTS Duratek Methods of Calculating Scan MDCs**

ATTACHMENT D **Summary of Continued Characterization Data**

2.0 SITE CHARACTERIZATION

2.1 Overview

The radiological characterization of the MY site has been going on since pre-operational sampling was begun in 1970. Initial site characterization was begun in the fall of 1997 and ran through the spring of 1998. Historical information, including the 10CFR50.75(g) file, employee interviews, Radiological Incident files, pre-operational survey data, special surveys (e.g., site aerial surveys, marine fauna and sediment surveys), operational survey records and Annual Environmental Reports to the NRC were reviewed and compiled into the Historical Site Assessment (HSA). Using the information collected during the HSA, a characterization plan was developed to collect measurements and samples from plant structures, systems and open land areas to cover all areas where contamination existed, remained or had the potential to exist.

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

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

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

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

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

2.2 Historical Site Assessment

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

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

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

2.2.1 Historical Data Review

Historical records contained in the radiation protection files, 10CFR50.75(g) file, Annual Environmental Reports to the NRC and one 10CFR20.302 submittal were reviewed to determine the location and extent of leaks and spills on site. The results of

the record reviews and employee interviews were captured in the Historical Site Assessment (HSA). The HSA was a compilation of the 118 potential events occurring over the 26 year operating history of the plant requiring investigation during site characterization. Of the 118 events, 79 were potential radiological issues and 39 were chemical or hazardous material events. Most of these events had already been identified and quantified in various event report records or survey records and copies of these records were added to the HSA. None of the event records indicated the uncontrolled release of radioactive material beyond Bailey Point (i.e., south of Ferry Road and east of Bailey Cove).

2.2.2 Decommissioning File 10CFR50.75(g)

Even though MY was in operation well before the requirement to maintain a decommissioning file, the 50.75(g) file contained documentation of three areas of soil contamination and one record of a 10CFR20.302 submittal for burial in place of residual soil activity. The information in the decommissioning file was added to the HSA so that the affected areas could be properly addressed during site characterization.

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

Table 1
Soil Contamination Events

Event	Date	Location	Volume	Disposition
RWST siphon heater leak	2/23/88	Area south and west of RWST	8200 ft^3	Remediated 600 ft^3. 7600 ft^3 left in place under 10CFR20.302.
Removal of Low Level Waste Storage Area	7/92	Outside the RCA Storage Bldg and west to high rad bunker	2000 ft^3	Residual contamination evaluated and entered into 50.75(g) file.

Table 1 Soil Contamination Events				
Event	**Date**	**Location**	**Volume**	**Disposition**
Silt spreading area	1992, 1993 Outages	Land adjacent to and south of ballfield.	1250 ft^3	Residual contamination evaluated and entered into 50.75(g) file.
Waste Neutralization Tk drain pipe leak	3/95	Underneath service building slab	193 ft^3	Residual contamination evaluated and entered into 50.75(g) file.

2.2.3 10CFR20.302 Submittal

MY applied to the NRC on 11/2/88 (MN 88-107) to allow residual soil contamination to remain in place under the provisions of 10CFR20.302. The NRC approved the submittal on 8/31/89. The details of the soil contamination are presented below.

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

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

Approximately 600 cubic feet of radioactive waste was generated from the excavation. Residual activity in an estimated 7600 cubic feet of remaining affected soil was 6 mCi. The location of this contaminated soil is well known and the need for further remediation will be evaluated during decommissioning to ensure compliance with the unrestricted use criterion.

2.2.4 Historical Radiological Status Including Original Shutdown Status

MY ran for approximately 16 full power years, had an early history of fuel clad failures and was known as a high source term plant. Dose rates in the loop areas in

Containment were approximately 1000 to 2000 mR/hr with surface contamination levels averaging in the 10,000 to 100,000 dpm/100 cm^2 range. Routinely-accessed areas of the PAB, Spray and Fuel Buildings had dose rates of 10 to 50 mR/hr, walkways were kept less than 1000 dpm/100 cm^2, and equipment spaces had dose rates of up to 1000 mR/hr and contamination levels on average of 5000 to 50,000 dpm/100 cm^2. The LSA, RCA Storage and LLWS Buildings had dose rates of 10 to 200 mR/hr depending on the type and quantity of waste in storage and contamination levels ranged from 5000 to 50,000 dpm/100 cm^2 in liquid waste processing areas to less than 1000 dpm/100 cm^2 in walkways.

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

In the late 80s and early 90s the plant began measures to reduce both the source term and surface contamination levels. Floor to ceiling area decontaminations were undertaken. High efficiency filters were installed in primary systems. One primary system chemical decontamination was performed which reduced primary system piping radioactivity levels by a factor of two.

In 1990, the plant experienced a primary to secondary steam generator tube leak. Prompt operator actions limited the secondary plant contamination to only volatile radionuclides. Following the steam generator tube leak, secondary systems were extensively surveyed during recovery activities and no residual activity was identified.

The plant was shutdown in December 1996 for evaluation of cable separation problems. During the extended outage, other equipment and regulatory problems were discovered which ultimately led to the decision to permanently shutdown. A second chemical decon was performed when the decision to decommission the plant was made. The decontamination factors for the second decon improved to five to ten which resulted in loop area dose rates in the range of 50 to 200 mR/hr. Contamination levels throughout the plant remained consistent with pre-shutdown values.

2.2.5 Current Radiological Status

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

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

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

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

2.2.6 Hazardous and Chemical Material Contamination

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

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

Decommissioning of the plant will include removal of additional known contaminants in plant systems and structures. Mercury switches, lead components, and PCB light

ballasts are some examples of hazardous materials that will be removed along with other plant components. Polychlorinated biphenyls (PCBs) found at other nuclear facilities are also present at MY but are limited to specific low concentration painted surfaces within the reactor containment building and in some cable insulation material. Asbestos abatement will continue to play a part in the removal of various components and building materials.

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

In these areas and throughout the site, MY will continue to work with the MDEP to demonstrate that areas have been adequately characterized, remediated if necessary, and are sufficiently clean to insure public health and safety.

2.3 Site Characterization Survey Methods

Initial Characterization was performed by GTS Duratek and its subcontractor. The DOC (Stone & Webster) and its subcontractor then performed the Continued Characterization based on the GTS Duratek Study and built the FSS plan based on this information. Site characterization was performed by two different contractors using similar but not identical methods and techniques. These differences are noted within the methods and results sections of this report.

2.3.1 Organization and Responsibilities

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

- IT Corporation performed the hazardous materials characterization survey and drive over scans.

- Duke Engineering & Services performed the activation analysis.

- Canberra Industries provided on site laboratory instruments.
- Team Associates performed the asbestos characterization.
- Quanterra performed off site laboratory analyses.

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

2.3.2 Characterization Data Categories

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

a. Surfaces and Structures

This category included building interiors and exteriors with associated structures, and, where applicable, the exterior surfaces of plant systems and components because these surfaces have the same potential for residual levels of radioactive material as the building surfaces in which they are located. Surface and structure survey packages also included ancillary buildings and structures. Structural background measurements were also included in this category.

In total, the survey category included approximately 7,850 measurements in unaffected areas and approximately 6,350 measurements in affected areas. This intentional bias toward unaffected surfaces and structures ensured no unsurveyed or undetected locations were likely to exist. Affected structure surveys included 18 concrete core samples. Because rubblized concrete was considered to be a primary source for evaluating the radiological release criteria, an additional 40 concrete core samples were taken to improve the nuclide data.

b. Systems

This category included interior surfaces of process piping, components, ventilation ductwork, and installed drains and sumps. The levels of radioactive

material on the internal surfaces of plant systems and components primarily depend on process operations. Therefore, these survey packages were separate from surface and structure survey packages. Plant system survey packages generally included one plant system.

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

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

c. Environs

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

Sediment, ground water and surface water samples were also included in this category.

Additional soil samples were taken for determining the amounts and ratios of the hard-to-detect radionuclides in soils.

Soils from beneath foundation slabs will be sampled and analyzed when the soil is exposed during remediation. Sub-slab soils beneath building foundations that remain in place after demolition will be evaluated by drilling holes through the foundation or by sampling soil at the edge of the foundation wall. Scan and fixed surveys of pavement were performed to identify potential sub-surface contamination. Two areas of soil contamination beneath pavement were documented in the HSA. One area of sub-slab leakage from the liquid waste effluent line occurred underneath the Service Building floor. The results of this soil contamination were contained in the 50.75(g) file.

2.3.3 Characterization Survey Design

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

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

The MY facility was divided into five groups and 127 packages which are listed in section 2.3.7. Groups were organized according to media type and contamination potential. Each group was further subdivided into survey packages that correspond to specific plant areas with similar operational history or physical location.

The survey package breakdown is contained in Attachment B. All plant areas are included in one of the survey groups/packages. The five groups are listed below.

- Group A-Affected Surfaces and Structures
- Group B-Unaffected Surfaces and Structures
- Group C-Affected Systems
- Group D-Unaffected Systems
- Group R-Radiologically Affected or Unaffected Environs

These group designators were also used during continued characterization for survey

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

- Group E-Hazardous Materials on Structures, Systems or Surfaces
- Group H-Hazardous Materials in Environs

Activation analysis calculations were also performed for the reactor vessel and internals.

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

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

a. Survey Methods

Direct measurements and surface scans of structures were performed with 126 cm^2 gas flow proportional detectors for beta contamination. The detector was kept within 1 cm of the surface. Measurements of surface activity on small or restricted access areas were made using small Geiger-Mueller detectors or and array of multiple detectors for large bore systems or components. Measurement times were controlled in order to achieve the required MDCs.

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

Samples of building materials, sediments, sludges and water were taken and analyzed using standard procedures and laboratory instruments. Smears for removable contamination were taken using standard techniques and laboratory counters. Exposure rates at one meter were measured using a NaI detector

and a pressurized ion chamber. Soil samples of approximately 1000 g were cleaned to remove large debris and dried to remove moisture. Samples were counted in Maranelli beakers using GeLi detectors for gamma emitters. Samples were analyzed by off site labs for hard-to-detect radionuclides.

b. Minimum Detectable Activities for Volumetric Measurements

Minimum detectable activities (MDAs) were defined for measurements and analyses used to quantify soil and other volumetric activity. Similar instruments, procedures and MDAs applied to continuing characterization. MDAs for volumetric materials ranged from less than 0.01 pCi/g or pCi/ml for the easily detected radionuclides to 20 pCi/g or pCi/ml for the harder to detect nuclides (including transuranics) except for Ni-59 which was in the 100 -500 pCi/g. These MDAs are well below the expected DCGLs for the nuclides of concern.

Table 2
Volumetric MDAs

Type of Analysis	**MDA (pCi/g)**	
	GTS	**DOC**
Gamma Spectroscopy	0.10	0.01 - 0.1
Liquid Scintillation	2.0 to 3.0	8.0
Alpha Spectroscopy	0.10	1.0
Radio Chemical Analysis	* 1 - 20 pCi/g	* 1 - 20 pCi/g

* except Ni-59

c. Structure and Surface Scan Sensitivities

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

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

The NUREG-1575/NUREG-1507 method was used to calculate scan sensitivities in the continuing characterization program. This method yielded surface scan MDCs of 800 dpm/100 cm^2 for large area gas flow detectors (Ludlum Model 43-68) and 4500 dpm/100 cm^2 for system internals surveys.

d. Open Land Area Scans

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

Table 3
Theoretical Scanning Sensitivities

Instrument	**Minimum Detectable Concentration/Activity**
Ludlum 44-2	14 pCi/g
VRM-1X	11 pCi/g*
SPA-3	5 pCi/g

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

Theoretical minimum detectable concentrations or minimum detectable activities for scans performed with a vehicle mounted VRM-1X detector, traveling at less than 5 mph, were calculated for several geometries based on empirical data and numerical integrations.

The SPA-3 detectors (2 inch by 2 inch NaI) were used for land area scans

during continuing characterization with scan MDCs of approximately 5 pCi/g as calculated according to paragraph 6.7.2.1 of NUREG-1575.

e. Instrument Calibrations

Analytical and field instruments were calibrated using National Institute of Standards and Technology traceable sources representative of the assumed radionuclide mix at the MY site. Instruments were calibrated at the MY site and, for GTS, at the GTS Duratek Central Calibration Facility in Oak Ridge, Tennessee or by vendors in accordance with the GTS Duratek Quality Assurance Project Plan for Site Characterization. Approved procedures were employed, to specify on-site instrumentation calibration requirements for continuing characterization.

Tc-99 and Am-241 sources were used for calibrating gas flow proportional instruments used to perform scans and direct measurements. Cs-137 sources were used to calibrate exposure rate instruments. The calibration program ensured that equipment was of the proper type, range, accuracy and precision to provide data to support the MY site characterization activities. The response of exposure rate instruments to Co-60 was also determined in order to detect discrete Co-60 particles.

2.3.5 Quality Assurance

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

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

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

a. Quality Control Samples and Measurements

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

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

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

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

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

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

An approved CCS Quality Control procedure for the sample quality control criteria was developed. This procedure covered instrument daily checks, split or spiked sample requirements and acceptability criteria. Five percent of all survey units were chosen for repeat surveys with 10% of scans, fixed point and exposure rate measurements being replicated. Agreement for replicates was considered to be values within $\pm$ 2 standard deviations. Instruments not passing the daily source check requirements were tagged "do not use" and were removed from service until repaired. Data not meeting the replicate count criteria were removed from the data base until evaluated by an FSS specialist or engineer.

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

The methods used by the off site laboratory for analysis of hazardous materials were based on the EPA method for solid waste analysis SW-846. Specific

quality control samples, analysis, and acceptance criteria are specified in the analysis methods.

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

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

b. Audits and Surveillances

MY provided oversight of survey and sample activities to determine whether the characterization plan was implemented as designed. External audits of project activities included assessments by MY personnel and subcontractors. These included an audit of the GTS Duratek facility in Kingston, TN and project-specific audits based on the Quality Assurance Program Plan and other project plans. These audits did not identify any project-specific nonconformances. In addition, MY personnel and their contractors performed surveillances on daily project operations. Characterization personnel identified, tracked, and corrected concerns generated by these surveillances.

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

procedures.

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

2.3.6 Data Quality Objectives

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

The DQO process was used during continuing characterization to obtain the inputs for FSS. Contamination boundaries, radionuclide profiles, data standard deviations and projected sample sizes were determined during continuing characterization.

2.3.7 Survey Findings And Results

The results of the characterization surveys are reported by survey group and package number as identified below.

PACKAGE NUMBER	GROUP "A" Affected Structures and Surfaces Survey Packages
A0100	Containment Building - Elevation -2 ft.
A0200	Containment Building - Elevation -20 ft.
A0300	Containment Building - Elevation 46 ft
A0400	Fuel Building - Elevation 21 ft.
A0500	Demineralized Water Storage Tank TK-21 - Elevation 21 ft.
A0600	Primary Auxiliary Building - Elevation 11 ft.
A0700	Primary Auxiliary Building - Elevation 21 ft.
A0800	Primary Auxiliary Building - Elevation 36 ft.

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

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

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

PACKAGE NUMBER	GROUP "D" Unaffected Plant Systems Survey Packages
D0100	Condensate System
D0200	Water Treatment Plant Systems
D0300	Potable Water System
D0400	Sanitary Sewer System
D0500	Circulating Water and Screen Wash System
D0600	Service Water System
D0700	Fire Protection System
D0800	Lube Oil System
D0900	Compressed Air System
D1000	Auxiliary Boiler System
D1100	Steam Generator System
D1200	Main and Reheat Steam System
D1300	Auxiliary Steam System
D1400	Main Turbine and Turbine Control System
D1500	Steam Dump and Turbine Bypass System
D1600	Main Feedwater System
D1700	Emergency/Auxiliary Feedwater System
D1800	Heater Drain and Extraction Steam System
D1900	Component Cooling Water System
D2000	Vacuum Priming and Air Removal System
D2100	Amertap System
D2200	Secondary Plant Sealing System
D2300	Auxiliary Diesel Generator
D2400	Secondary Sample and Chemical Addition System
D2500	High Pressure Drains
D2600	Environmental Services Laboratory Systems
D2700	Administration Building HVAC System
D2800	Information Building HVAC System
D2900	Turbine Building Ventilation System
D3000	Staff Building HVAC System
D3100	Service Building HVAC System
D3200	Hydrogen and Nitrogen System

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

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

PACKAGE NUMBER	GROUP "R" Environs Affected and Unaffected Survey Packages
R2500	Driveover Elevated Areas
R2501	Follow-up sampling at Elevated Soil Sample Locations (south of Refueling Water Storage Tank and Contractor Parking Lot)

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

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

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

2.4 Summary of Characterization Survey Results

The operational history as well as the range of contamination determined during site characterization is summarized below for each survey group. More detailed data including

mean, maximum, and standard deviation are presented by survey package in Attachment B.

2.4.1 Group A "Affected Structures and Surfaces"

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

Maximum surface activities ranged from greater than 100,000 dpm/100 cm^2 in the RCA Storage, Containment and Spray Buildings to less than 1000 dpm/100 cm^2 in auxiliary support areas (e.g., electrical/mechanical penetrations). No removable alpha sample activities were above the MDA values which indicated little or no transuranic (TRU) surface contamination. Maximum exposure rates reported in Attachment B. ranged from 12,400 uR/hr in the RCA Storage Building to 14 uR/hr in the mechanical penetration area. Operational surveys reported containment exposure rates ranging from 1 mR/hr to 1000 mR/h.

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

2.4.2 Group B "Unaffected Structures and Surfaces"

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

stored at the Bailey House environmental laboratory..

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

Survey results ranged from a high of 8600 dpm/100 cm^2 (in the chemical addition tank from naturally occurring material) to a low of 275 dpm/100 cm^2 in the cable vault. No areas had alpha activity above the MDA level. Maximum exposure rates ranged from 26 uR/hr in the Service Building to 9 uR/hr in the Cold Shop. Tritium was detected slightly above MDA not surprisingly in several water-containing systems.

Group B surveys verified that most of the Turbine Hall was free of residual radioactivity and established the extent and limits of radioactivity in the areas in which it was found. This ensured proper control of radioactive material prior to and during decommissioning.

2.4.3 Group C "Affected Plant Systems"

This group was comprised of the radioactive systems such as the RCS, CVCS, ECCS, liquid and solid waste, containment ventilation and primary vents and drains. The survey packages in this group consisted of systems and components that will be removed and disposed of as radioactive waste during decommissioning. These are the highest activity systems at MY.

Internal system surfaces were found to be contaminated up to 500,000 dpm/100cm ^{2} removable beta activity. Alpha activity was present at as much as 35 dpm/100 cm^2 above the MDA in the CVCS indicating possible TRU contamination. Exposure rates in these areas ranged from a low of 2 mR/hr in the Containment ventilation system to more than 16,000 mR/hr in the Spent Fuel Cooling system.

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

2.4.4 Group D "Unaffected Plant Systems"Including The Sewerage Treatment System

This group consisted of secondary side systems that were designed to remain non-contaminated. Examples of these systems are main steam, feedwater, compressed air and potable water. However, certain parts of the secondary side systems do contain minor levels of contamination. The auxiliary condensate system was known to be slightly contaminated due to aux boiler problems early in plant life (maximum removable beta activity of 115 dpm/100 cm^2). Turbine Hall sumps were known to be slightly contaminated (maximum direct beta activity of 5800 dpm/100 cm^2) due to reactor coolant pump motor refurbishment activities taking place in the Turbine Hall. Steam and feedwater systems were potentially impacted by the 1990 steam generator tube leak. Group D systems were generally the lowest in activity of all those surveyed.

Until the early 1980s when they were disconnected, hot side shower drains and toilets were directed to the sewerage treatment plant. Characterization surveys showed 5600 dpm/100 cm^2 in one shower drain. Over the past two years, routine chemistry analyses of both the on site holdup tank and the municipal treatment facility have shown no plant-derived radionuclides. Radionuclides have been detected in the sewage plant as a result of employees receiving medical isotope therapy.

Several of the systems had elevated readings indicating the possible presence of radioactive material. Further measurements were made on these systems as part of the continuing characterization plan to properly evaluate the level and extent of contamination. Upon investigation, nine of the 34 systems in group D had detectable levels of plant derived radioactive material.

Survey results from Group D established the limit and extent of residual activity in systems expected to be clean and provided necessary information to properly control the systems as well as classify the waste during decommissioning. Systems in this group found to be contaminated with plant related radioactivity will be disposed of as radioactive waste.

2.4.5 Group R "Environs Affected and Unaffected

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

sampling.

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

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

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

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

2.4.6 Ventilation Ducts and Drains

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

Affected System Vents and Drains (C0600, C1500, C1600 and C1800) showed mean removable contamination values ranging from 2000 to 400,000 dpm/100 cm^2 and maximum values from 6000 to 500,000 dpm/100 cm^2.

Unaffected System Vents and Drains (D1800, D2000, D2500, D2700, D2800, D2900, D3000, D3100 and D3300) had only one system (D3100 Service Building HVAC) with significant activity above the MDA which was due to the hot side ventilation sources going to the Service Building ventilation duct work. There were three (3) other systems (High Pressure Drains, Staff Bldg HVAC, and Turbine Bldg Sumps) with positive activity. The High Pressure Drains showed tritium activity at levels just above MDA. Staff Building ventilation (up to 1300 dpm/100cm^2 total beta activity)showed no evidence of plant derived radionuclides and the elevated readings were likely due to radon daughter activity. Two Turbine Building sumps tested positive for plant derived nuclides (up to 1.7 pCi/g) and the sump oil collection tank also tested positive (1pCi/g). Affected and Unaffected Environs (R0300) had two areas in the vicinity of drain discharge openings showing positive results for Co-60 of 0.04 to 11.0 pCi/g.

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

2.4.7 Embedded Piping

MY has a limited amount of piping embedded in concrete (less than 1000 linear feet). Embedded piping includes approximately 800 feet of primary and secondary component cooling water pipes. Component cooling piping showed maximum activity up to 22,000 dpm/100 cm^2. Small segments of refueling cavity and spent fuel pool skimmer piping (approximately 175 feet) are embedded within the walls of the two pools. The skimmer piping is known to be contaminated and activity levels could be as high as 20,000 to 180,000 dpm/100 cm^2 removable beta contamination based on data obtained from spent fuel pool cooling (C0700) and RHR (C0500) survey packages. Two valves from the containment spray system are encased in concrete. Levels up to 40,000 dpm/100 cm^2 were detected in the spray system (C0300).

Circulating water and service water pipes are buried cast concrete pipes rather than embedded pipes. One direct measurement above MDC was identified in the circulating water pipes but, no plant related nuclides were detected. Service water receives the liquid effluent overboard pipe with approximately a 3 foot embedded section and showed maximum activity levels of 3100 dpm dpm/100 cm^2. Mean values were less than MDA.

2.4.8 Rubble

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

While rubble was not a significant item during site characterization, it is expected to become a significant dose contributor following structure demolition. Because of the impact of rubblized concrete on the exposure pathway, concrete core samples were collected and analyzed during initial characterization (A9900, A9901, A9902) and continuing characterization. Concrete activity was found to be due to penetration of surface contamination as well as activation of concrete constituents in areas exposed to neutron flux. (Activated concrete was approximately 5% of the concrete in Containment.) Surface contamination penetration was limited to the top 0.1 inch while activation activity was fairly uniformly distributed to a depth of 3 to 4 inches.

Table 4
Initial Contaminated Concrete Characterization

Contaminated Concrete		
Radionuclides	**Activity Fraction**	**Depth of Activity**
Cs-134	0.008	99% of activity is contained in the first one inch of concrete.
Cs-137	0.821	
Co-60	0.170	
Eu-152	<0.001	
Mn-54	<0.001	
Sb-125	<0.001	

Table 4 Initial Activated Concrete Characterization		
Activated Concrete		
Radionuclides	**Activity Fraction**	**Depth of Activity**
Cs-134	0.028	Activity is approximately evenly distributed throughout the material to a depth of about three inches.
Cs-137	0.046	
Co-60	0.266	
Eu-152	0.600	
Eu-154	0.060	

2.4.9 Paved Areas

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

2.4.10 Components

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

In Group D there were 5 systems (water treatment, compressed air, auxiliary steam, component cooling and hydrogen/nitrogen) with one or more components requiring further evaluation as part of continuing characterization. The water treatment plant had chemical batching tanks with Naturally Occurring Radioactive Material (NORM) present up to 26,000 dpm/100 cm^2 beta activity. The hydrogen/nitrogen and compressed air piping was discovered to have contamination in one or more sections of pipe running through the PAB. The component cooling and auxiliary boiler/steam system have been described in sections 2.4.2 and 2.4.4 above.

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

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

There were no hazardous material health and safety considerations identified as a result of these surveys.

2.4.12 Surface and Ground Water

Surface water from a small drainage pond at the north end of the site and estuary water in the area of the circulating water discharge area were sampled. Ground water samples were collected from foundation sumps and monitoring wells located on site. Sample results for surface and ground water were reported within the individual survey area packages (R0100, R0200, R0300, R1100, R2200 and R2400).

The overall range of the tritium analyses was 441pCi/L to 6812 pCi/L and the mean was 1467 pCi/L. The highest value was from the Containment foundation sump and the remaining measurements were well below both a background tritium measurement of 6073 pCi/L obtained from a lake in northern Maine in 1984 (1984 MY REMP Report) and the EPA Drinking Water MCL of 20,000 pCi/L.

2.4.13 Background

Measurements were made of several types of construction materials from offsite locations which maybe used as background samples. Soil samples from remote locations were also taken and analyzed to be used as background soils This information was collected even though background subtraction is not anticipated to be required as discussed in Section 5 of the LTP subsection 5.4.3.

2.4.14 Waste Volumes and Activities

The Decommissioning Contract proposal provided estimated waste volumes and curie amounts as shown below.

Projected Activities and Volumes		
Reactor Vessel and Internals	2,600,000 Ci	11,527 cu.ft
Contaminated Components	1,600 Ci	27,000 cu.ft
Activated Concrete	388 Ci	23,000 cu.ft
Contaminated Debris	0.10 Ci	163,000 cu.ft
Radioactive Water (contained in systems)	0.03 Ci	850,000 gal
Contaminated Soil	0.01 Ci	25,000 cu.ft

2.5 Continuing Characterization

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

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

In addition, the plan to rubblize concrete and leave it on site created a need for more nuclide profile data, especially concrete nuclide profiles, in order to support site specific dose pathway analysis. The Continued Characterization Plan and survey schedule were developed in order to obtain the data needed for both purposes consistent with the decommissioning schedule.

Furthermore, the new Spent Fuel Pool Decay Heat Removal System is contaminated. Remediation plans called for the system components to be removed and disposed of as radwaste. Once fuel has been transferred to the ISFSI, the area occupied by the SFP cooling system will be surveyed.

2.5.1 Methods

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

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

2.5.2 Results

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

The range of residual radioactivity existing on surfaces, within soils and systems, as well as background areas and materials targeted for sampling during Continuing Characterization are summarized below. Detailed data including mean, maximum, and standard deviation are presented by survey package in the attached tables.

a. Risk Areas

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

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

soil contamination were determined by gamma spectroscopy of soil samples. In addition, soil was sent for radiochemical analyses in order to confirm the ratio of radionuclides including the hard-to-detect nuclides.

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

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

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

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

and aux boilers were contaminated. The system was labeled to show the extent of contamination.

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

In addition, some unresolved data from the GTS Report was investigated. The activity in the water treatment plant (CD0200) was determined to be NORM.

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

b. Soils

Surface soil was sampled and analyzed for radionuclides during site characterization. The radionuclides were detected in the top 15 cm of on-site soil in the survey areas encompassing the backyard. Additional data were collected during continued characterization to better establish nuclide profiles. The predominant plant-related, beta-gamma emitting radionuclides detected were H-3, Co-60, Ni-63 and Cs-137, making up 91 percent of total activity. TRUs were not detected in CCS soil samples so two sets of higher activity soil samples taken by GTS were composited and subjected to radiochemical analyses for the hard-to-detect nuclides. No TRUs were detected in the composites when analyzed with techniques giving MDAs of 0.01 pCi/g to 0.005 pCi/g.

Sub-surface soil has been sampled and characterized in areas in which there was knowledge or indication of contamination below 15 cm. The nuclide ratios were consistent with surface ratios. In addition, building sub-slab soil samples will be taken during remediation to determine the presence and extent of any sub-slab contamination.

c. Systems and Components

Residual contamination on or in plant piping was the result of the deposition of both fission and activation products. Prior to and during characterization surveys, samples of process piping were obtained to determine which systems

were contaminated and the current radionuclide profiles including the hard-to-detect nuclides. The bounds of the contaminated piping were not established initially so systems were opened and surveyed to define the bounds of contamination. Decommissioning plans state that contaminated system components and piping will be removed and disposed of as radioactive waste.

Fe-55, Ni-63, Co-60 and Cs-137 made up 97 percent of the system activities determined during initial characterization. TRUs contributed less than 1 percent of the total activity. The major beta-gamma emitter detected in system materials was Co-60 with a range of activity of 1 to 715 pCi (MDAs were 0.03 to 5 pCi/g). No Cs-137 was detected on system components during CCS due to the high solubility of the Cs-137 in waterfilled systems. No additional quantitative gamma analyses for systems or components were conducted during CCS.

d. Embedded Piping

A review to determine the extent of embedded piping was undertaken during continuing characterization. A limited amount (approximately 1000 lineal feet) of embedded piping was identified. Embedded piping, including any in-slab drains, remaining after demolition will receive special surveys during the FSS. The nuclides and ratios in embedded piping are consistent with those listed in the "Systems and Components" section described above since the systems with embedded sections of contaminated pipe were the systems sampled during initial characterization as reported in subsection 2.4.3.

e. Structures-Surface Activity

Residual contamination on building surfaces was the result of system leaks and maintenance activities which deposited fission and activation products on these surfaces. Prior to and during the initial characterization surveys, samples of building surfaces were obtained to determine current radionuclide profiles including the hard-to-detect nuclides.

Four radionuclides, Fe-55, Ni-63, Co-60 and Cs-137 comprise approximately 97 percent of the radioactivity on the structures, with transuranics (TRUs) contributing less than 1 percent of the total activity. The predominant beta-gamma emitters in these areas are Co-60 and Cs-137.

f. Structures-Concrete

Concrete structures 3 foot above grade will be demolished. Surfaces will be decontaminated to unrestricted use criteria prior to demolition. This rubble will be used as fill within the lower elevations of the buildings and/or used as fill on other parts of the site. The radionuclide distribution is described below.

Radioactivity found in the concrete shielding materials in containment was the result of both contamination and activation. Concrete plugs were removed and analyzed in order to estimate the radioactivity levels and nuclide distributions of shielding materials. The predominant radionuclides present in the structural concrete are Co-60, Ni-63 and Cs-137 which comprise approximately 96 percent of the beta-gamma activity. Eu-152 and Eu 154 are present at approximately 71% in the neutron activated containment concrete. No TRUs were detected in concrete samples analyzed with MDAs of 0.05 to 0.1 pCi/g.

Table 5
Contaminated Concrete Mixture

Contaminated Concrete		
Radionuclides	**Activity Fraction**	**Depth of Activity**
H-3	0.002	99% of activity is contained in the first one mm of concrete.
Fe-55	0.009	
Co-60	0.071	
Ni-63	0.601	
Sr-90	0.0007	
Cs-134	0.007	
Cs-137	0.310	
Np-237	<0.00008	

Table 6 Activated Concrete Mixture		
Activated Concrete		
Radionuclides	**Activity Fraction**	**Depth of Activity**
Co-60	0.27	Activity is approximately evenly distributed throughout the material to a depth of about three inches.
Cs-134	0.02	
Eu-152	0.65	
Eu-154	0.06	
Cs-137	None from Activation	

Concrete cores were counted using both hand-held instruments and gamma spectrometers. This information coupled with the radiochemical analytical data were used to determine instrument E_s values.

2.5.3 Background Determination

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

a. Soils

Soil samples were taken from the non-impacted areas and analyzed in order to establish soil background levels. The samples showed two distinct populations with respect to background concentrations of Cs-137 using the Kruskal-Wallis test; samples from undisturbed soil had mean Cs-137 levels of 0.3 pCi/g while disturbed soil had mean Cs-137 levels of 0.2 pCi/g. The naturally-occurring uraniums (U-234, U-235, and U-238) were present in expected amounts (approximately 0.01 pCi/g of U-234 and U-238 with about 0.0002 pCi/g of U-234). Sr-90 was not detected at or above a MDA of 0.4 pCi/g.

Table 7 Soil Mixture		
Contaminated Soil		
Radionuclides	**Activity Fraction**	**Depth of Activity**
H-3	0.026	Soil samples from 0 to 15 cm. Depth.
Co-60	0.012	
Ni-63	0.040	
Cs-137	0.902	
U234	0.009	
U-235	0.0002	
U-238	0.009	

b. Structures

Background measurements were taken on structural materials during initial characterization in order to verify the contribution of background activity to the total measurement value. The same types of detectors are being used for FSS as were used during characterization.

Structural background values may be subtracted from structure survey measurements in order to determine the level of residual, plant-derived radioactivity when total activity levels are close to the MDCs. Background values for structural materials are shown below.

Table 8 Structural Material Backgrounds		
Background Counts per Minute*		
Materials	**43-68 Proportional Detector 126 cm²**	**SHP-360 G-M Pancake Detector 15.5 cm²**
Painted Cinder Block	296	70
Wood	301	57
Ambient	319	65
Steel	321	61
Carpet	339	68
Floor Tile	371	74
Ceiling Tile	386	78
Bare Cinder Block	394	79
Painted Concrete	411	83
Bare Concrete	415	88
Asphalt	525	99
Granite	566	128
Porcelain	607	116
Brick	632	126

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

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

2.5.4 Impact of Repowering

If the site is repowered, those systems or components remaining which were anticipated to be removed during decommissioning will have to be surveyed to determine if they meet the dose criteria for unrestricted release and use. Such items may include Turbine Hall drains, sumps, small cooling lines, and sub-slab soils.

2.6 Summary

2.6.1 Impact Of Characterization Data On Decontamination And Decommissioning

Characterization data confirmed what was known about the MY site in terms of the level and extent of radioactive contamination. A major portion (700 acres) of the site met the classification of non-impacted. Primary systems and structures were found to be contaminated to expected levels. Non-RA systems and structures were found to be free of contamination except as previously stated (PCC/SCC, Service Water, Condensate/Aux Condensate). Two small areas of surface contamination (Contractor's parking lot, warehouse parking lot) were discovered and one suspect area (Bailey Point) was confirmed.

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

The data compiled by GTS Duratek as supplemented by the continued characterization measurements are sufficient to project schedules and waste volumes, evaluate decontamination techniques, perform dose assessments and evaluate any safety or health issues affecting workers on site.

The HSA and characterization measurement results are sufficient to meet the objectives listed in subsection 2.1 and demonstrate compliance with the guidance contained in Regulatory Guide 1.179 and NUREG-1700. The more than 19,000 measurements taken provide the data to determine the radiological status of the site and facility as well as identify the location and extent of contamination outside the RCA. The radionuclide analyses performed were sufficient to estimate the source term and isotopic mixture (based on the achieved standard deviation of the data). The analysis results also provide sufficient information to support dismantlement, radioactive waste disposal, decommissioning cost estimates and remediation decision making processes. The source term information was also suitable for instrument selection. The radiological data were acceptable to develop the necessary quality assurance methods for sample collection and analysis. The data obtained during characterization support dose assessment and FSS design. None of the characterization results identified any

underlying or unique health and safety issues associated with decommissioning.

2.7 References

- NUREG-1575, Multi-Agency Radiation Survey and Site Investigation Manual, (MARSSIM), 1997.
- 10CFR.50.75, Reporting and Recordkeeping for Decommissioning Planning.
- Continuing Characterization Plan.
- CCS Quality Control.
- Corrective Action Program Interface.
- QA Records Management System.
- Radiation Protection Program Assessment.
- Selection, Training and Qualification of Radiation Protection Personnel.
- MY RCRA Closure Plan.
- NUREG-1507, Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions.
- NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans. (Draft)
- Regulatory Guide 1.179, Standard Format and Content of License Termination Plans for Nuclear Reactors.

ATTACHMENT A

Non Impacted Area Assessment

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

Introduction

One aspect of the FSS Plan is the proper classification of areas within the site. Areas must be classified as either: Impacted, Class1, Class 2, or Class 3; or Non-impacted. Non-impacted areas are defined in NUREG-1575 (MARSSIM) as areas that "have no reasonable potential for residual contamination, no radiological impact from site operations and are typically identified during the Historical Site Assessment". The MY Historical Site Assessment did not classify any areas within the site but it did provide data which could be used in conjunction with other information to classify areas.

Area Description

Approximately 700 acres of the MY site are found to the West of Bailey Cove, North of the access road (Ferry Road) and bounded by Back River to the east. The land is generally located beyond the 2000 foot exclusion zone established under the requirements of 10CFR100. As such, the area has been open and accessible to the general public and is bounded by residential land owners.

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

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

Historical Site Assessment

The land areas under consideration are approximately 0.25 miles or more from the Reactor Building and process buildings. No radioactive material was used or stored beyond the peninsula of Bailey Point. License restrictions and administrative controls have been in place since power operations began in 1972 to prevent unauthorized removal of radioactive material from the owner controlled area. Planned offsite releases of radioactive material were limited to the permitted effluent releases (which were kept ALARA by process controls) and radioactive solid waste which was shipped to licensed burial sites. The Historical Site Assessment (HSA) documented 118 actual or potential events involving unplanned releases of radioactive material or hazardous material during the 25 year operating history of the plant. Of the 118 total events, 79 were events potentially involving radioactive material. Based on

a review of the documentation assembled in the HSA, none of these events would have resulted in residual contamination of the area under consideration. Therefore, there is no reasonable potential for residual contamination in the area.

Radiological Environmental Monitoring Program

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

TLD measurements have shown no difference in dose rates between the area under discussion and the control areas further from the site. Bailey Farm well water had slightly lower H-3 levels on average than the water supplies in the Wiscasset area. Precipitation H-3 levels at local sampling stations (Eaton and Bailey Farms) were similar to the control station levels. Fruits and vegetables sampled at the Bailey Farm showed the presence of only K-40 and fallout-produced Cs-137. Grasses sampled at the Eaton and Bailey Farms showed only natural K-40 and fallout-produced nuclides during periods of atmospheric testing. Soil samples were taken at several locations within the area of concern starting in 1970 and ending in 1997. Initial soil samples had Cs-137 at levels consistent with published values for fallout activity. Samples taken during the intervening period had Cs-137 levels consistent with that which should have resulted from the decay of the initial 1970 sample activity. No radionuclides of plant origin were detected in these areas.

Special Surveys And Reports

The HSA and other sources document samples of radiation and radioactive materials taken in the area in question. Pressurized ion chamber readings, TLD measurements, soil samples and even a "fly over" dose rate survey have documented radiation levels in the area similar to, or slightly less than, those measured pre-operational surveys. The slight decline in levels is likely due to decreased levels of fallout-produced Cs-137 (Aerial Radiation Measurement Study, 1974 and University of Maine, 1974 and 1997). Shoreline sediment levels of radioactivity in Bailey Cove were measured in 1994 prior to the State of Maine reopening the area to clamming. This study examined the radionuclides profiles of sediment from three areas in the original liquid effluent discharge pathway. Only Cs-137 was detected in the sediment and the 1994 levels were consistent with levels produced by fallout.

Based on NUREG-1575 requirements, classification of an area as "not impacted" can be made solely on the Historical Site Assessment. Rather than rely solely on the HSA, the area in question was subjected to site characterization surveys. During 1997 and 1998, GTS performed site characterization measurements in the area which included gamma dose rates determined by pressurized ion chamber and micro R meter, soil samples and "drive around" surveys using a vehicle-mounted 1.5"x 3"x 33"

scintillation detector. The characterization surveys (PIC and "drive around") in the area produced 22 areas of elevated radiation levels. Upon investigation, 22 of the elevated readings in the areas in question were found to be due to local increases of naturally occurring radiation (granite outcroppings). The 180 soil samples taken throughout the area showed only background levels of radioactive material in quantities slightly less than those reported in the 1972 pre-operational studies in this area which is consistent with the decay of the fallout-produced activity.

Conclusion

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

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

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

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

Soil Cs-137 (pCi/g)					
Sample Location	**1970 MY**	**1972 MY**	**1974 MY**	**1996 MY**	**1997 GTS Characterization**
Bailey House	0.64	1.67	1.8	0.4	0.21; n=30
Bath	0.66				
Dresden	0.58				
Eaton Farm	0.53	0.87	2.5	0.09	0.46; n=60
Edgecomb	0.48				
Foxbird		0.35		0.48	
Knight Cemetery		4.96		2.42	
Long Ledge		0.80		0.38	
Harrison's	0.52				
Mason Station	0.68				

Soil Cs-137 (pCi/g)					
Sample Location	1970 MY	1972 MY	1974 MY	1996 MY	1997 GTS Characterization
Montsweag Dam	0.42				
Westport	0.56	1.11		1.03	
North of Ferry Road					0.39; n=60
Merrymeeting Airfield					0.35; n=60
Shoreline					0.20; n=30
Mean Value	0.56	1.63	2.15	0.80	0.32

Surface & Well Water Data	(Mean H-3 pCi/L) 1977-1984
Bailey House	235
Montsweag Dam	276
Morse Well	187
Biscay Pond	297
Wiscasset Reservoir	278
Moxie Lake	6073

Precipitation Data	(Mean H-3 pCi/L) 1977-1982
Bailey House	416
Eaton Farm	417
Westport	422
Dresden	397

Air Particulate Data (Mean Gross Beta Activity, pCi/m^3)			
MY Pre-Operational Data			
1970		0.12	
1971		0.12	
1972 Jan-Jun		Zone I=0.06, Zone II=0.07	
Univ. of Maine 1981-1997		MY 1988-1998	
Wiscasset	0.02*	Montsweag	0.021
Augusta	0.02*	Bailey House	0.020
		Mason Station	0.020
		Westport	0.021
		Dresden	0.022

* Values estimated by graph. Individual data not available.

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

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

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

ATTACHMENT B

Characterization Data

Group A
Radiological Characterization Results For Structures and Surfaces

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate MicroR/hr		
Package	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**
A0100 Cont.El -2 Ft	81,976 (30,453)	1,970,974	259,134.5	296 (33)	4,282	598.7	0.0 (8.4)	2.4	0.5	2,375 (816)	4,065	816
A0200 Cont. El 20 Ft	62,970 (16,277)	2,238,614	247,399.2	2,388 (35)	128,734	13,577.2	0.7 (9.7)	7.3	1.6	887 (463)	1,961	463
A0300 Cont. El 46 Ft	38,444 (16,058)	345,960	55,889.2	1,469 (33)	31,054	3245.7	0.2 (8.7)	5.8	1.1	499.5 (388)	2,408	387.5
A0400 Fuel Bldg El 21 Ft	6,815 (12,436)	312,939	32,365.4	38.4 (32)	879	106.2	-0.1 (8.5)	1.8	0.6	706.6 (650)	2,901	649.7
A0500 DWST	438 (2,322)	2,659	792.6	4.9 (32)	20.3	7.0	0.1 (8.4)	3.9	1.0	14.0 (0.9)	14.6	0.9
A0600 PAB El 11 Ft	1,106 (13,168)	32,328	7513.5	5.2 (32)	32.3	8.0	-0.1 (8.5)	3.9	0.7	1,100 (827)	3,477	827
A0700 PAB El 21 Ft	460 (15,837)	25,000	4655.1	5.9 (32)	51.5	9.7	-0.2 (7.7)	1.8	0.3	581 (950)	4,068	950
A0800 PAB El 36 Ft	508 (18,042)	14,073	2166.5	5.9 (34)	94.2	11.0	0.1 (7.0)	2.0	0.6	187 (182)	769	182
A0900 RA Svc Bld	699 (1,970)	18,955	2927.8	9.2 (34)	251	26.6	-0.6 (8.2)	3.9	0.6	42 (78)	501	78
A1100 LLWSB	852 (17,886)	74,216	6023.3	0.3 (38)	35.8	7.0	0.1 (8.1)	4.1	0.8	334 (752)	3,563	752

Group A
Radiological Characterization Results For Structures and Surfaces

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate MicroR/hr		
Package	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**
A1200 RCA Storage	73,939 (26,286)	2,233,580	379,578.7	128.7 (37)	2,073	323.1	-0.1 (8.6)	1.8	0.6	2,162 (2864)	12,389	2,864
A1300 Equip Hatch	27.5 (600)	720.5	255.1	4.9 (35)	19.8	7.6	-0.1 (7.8)	1.9	0.5	27.1 (34)	122.7	33.7
A1400 Pers Hatch	350.2 (2198)	6,758	1379.9	47.1 (35)	657.5	126.8	-0.2 (7.8)	1.9	0.3	47.5 (41)	180.2	41.2
A1500 Mech Pen	214.9 (661)	3,678	734.3	4.4 (38)	23.5	7.7	-0.2 (8.4)	3.9	0.6	9.4 (2.6)	14.0	2.6
A1600 Elec Pen	-138.0 (654)	557.1	269.7	1.9 (37)	18.2	6.9	0.0 (7.7)	1.8	0.6	12.7 (1.0)	14.0	1.2
A1700 Spray Bld	83,249 (24,797)	4,968,088	431,253.4	177.5 (37)	19,727	1445.2	0.0 (7.2)	2.0	0.4	1,598 (2124)	9,041	2,124
A1800 Aux Feed Pump	147.5 (2,019)	1,278	422.4	2.3 (37)	36.6	11.3	-0.1 (7.7)	1.8	0.5	18.9 (7.0)	34.9	7.1
A1900 HV-9	130.6 (6318)	2,563	725.3	0.6 (36)	24.6	7.0	-0.1 (8.2)	1.8	0.6	90.6 (46)	182.9	45.9
A2100 RWST	3,602 (21,587)	54,719	13,158.9	2.7 (38)	72.4	13.5	0.0 (8.4)	1.8	0.7	687.5 (374)	1,078.4	374.0
A2200 BWST	7,269 (21,255)	43,189	10,833.4	7.1 (36)	73.2	16.9	-0.1 (8.2)	1.8	0.6	667.6 (247)	1,197	246.6
A2300 PWST	668 (2,780)	3,258	942.1	5.8 (32)	27.4	7.1	0.1 (8.4)	1.8	0.8	N/A	N/A	N/A

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate MicroR/hr		
Package	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**	**Mean (MDC)**	**Maximum**	**Std. Dev.**
A2400 Test Tks	955.5 (1438)	4,300	1062.8	3.5 (36)	30.7	7.3	0.4 (8.2)	5.8	1.3	N/A	N/A	N/A

Group B
Unaffected Structures and Surfaces, Including Structural Background Survey

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate microR/hr			Plant Derived Nuclides ?
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
B0100 Turb El 61Ft	26.7 (636)	653.7	246.9	3.5 (17)	19.1	4.8	-0.3 (7.6)	4.8	0.9	9.0 (2.0)	15.2	1.9	N/A
B0200 Control Rm (Old)	215.8 (616)	1054.2	384.1	4.1 (16)	25.8	5.4	-0.5 (7.6)	2.0	0.7	10.2 (1.0)	12.5	1.1	N/A
B0300 MCC	-91.0 (701)	552.5	299.7	1.9 (17)	11.7	4.8	-0.2 (7.3)	2.1	0.9	12.2 (2.0)	14.9	2.0	N/A
B0400 Fire Pmp	10.1 (610)	840.1	351.2	2.6 (32)	18.4	5.3	-0.6 (8.2)	0.7	0.4	11.2 (2.0)	12.8	1.6	No
B0500 Turb El 21Ft	62.1 (649)	8613.8	752.2	2.8 (17)	203.4	15.8	-0.4 (7.3)	2.1	0.7	8.6 (3.0)	17.3	2.8	N/A
B0600 Turb El 39 Ft	48.2 (603)	2031.4	332.9	2.9 (17)	30.0	6.1	-0.1 (7.3)	3.5	0.9	6.3 (3.0)	13.7	2.9	N/A
B0700 Svc. Bld. Non-RCA	80.0 (821)	1621.5	411.1	2.8 (32)	19.9	5.0	-0.1 (8.4)	2.4	0.7	12.5 (4)	26.0	3.5	N/A
B0800 FOSB	-82.7 (587)	451.4	286.0	5.5 (16)	19.9	6.1	-0.2 (6.7)	0.9	0.5	8.4 (1.0)	9.9	0.8	N/A
B0900 EDGs	-176.9 (683)	411.9	209.8	4.3 (16)	19.9	5.6	-0.1 (6.7)	0.9	0.6	10.8 (2.0)	13.1	1.6	N/A

Group B
Unaffected Structures and Surfaces, Including Structural Background Survey

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate microR/hr			Plant Derived Nuclides ?
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
B1000 Aux Boiler	183.4 (679)	1309.7	492.6	3.4 (16)	16.5	5.9	-0.2 (6.7)	2.4	0.7	9.2 (1.0)	10.5	0.9	N/A
B1100 Circ Water	-333.9 (699)	672.7	300.5	1.8 (16)	11.4	4.1	0.0 (6.7)	2.4	0.9	8.5 (1.0)	10.8	1.3	No
B1200 Admin Bld	293.1 (686)	1628.2	431.9	4.3 (16)	14.8	5.1	0.0 (6.7)	2.4	0.9	13.3 (2.0)	15.2	1.5	N/A
B1300 WART	-146.3 (666)	1163.8	542.5	2.6 (16)	13.1	4.5	0.1 (6.7)	2.4	0.9	11.1 (1.0)	12.9	1.2	N/A
B1400 Info Ctr	295.3 (678)	1928.8	325.6	2.1 (16)	21.5	5.0	0.1 (6.7)	3.8	1.0	13.4 (1.0)	16.8	1.3	N/A
B1500 Warehse 2	96.1 (566)	539.0	212.4	0.6 (18)	19.4	5.2	-0.3 (7.3)	2.1	0.8	10.3 (1.0)	15.1	1.4	N/A
B1600 Trng Annex	-13.5 (657)	708.2	256.1	1.6 (18)	17.7	4.8	-0.2 (7.3)	2.1	0.8	17.8 (4.0)	23.8	3.5	N/A
B1700 Staff Bld	129.4 (727)	952.9	279.5	-1.0 (18)	14.4	4.5	-0.4 (7.3)	3.5	0.7	14.2 (3.0)	23.2	3.3	No
B1800 Spare Gen Bld	-39.8 (548)	341.9	176.6	0.1 (18)	9.3	4.6	-0.5 (7.3)	0.7	0.5	N/A	N/A	N/A	No

Group B Unaffected Structures and Surfaces, Including Structural Background Survey													
	Direct Beta DPM/100 cm2			**Removable Beta DPM/100 cm2**			**Removable Alpha DPM/100 cm2**			**Exposure Rate microR/hr**			**Plant Derived Nuclides ?**
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
B1900 Bailey House	612.3 (682)	6523.7	1595.1	0.3 (18)	11.0	6.1	-0.4 (7.3)	0.7	0.6	9.4 (4.0)	16.1	3.6	No
B2000 Bailey Barn	-96.6 (592)	306.5	187.3	1.1 (18)	9.3	4.6	-0.4 (7.3)	0.7	0.6	9.2 (1.0)	10.6	0.8	N/A
B2100 Lube Oil Storage	8.7 (630)	610.4	240.7	0.2 (18)	7.6	4.3	-0.5 (7.3)	0.7	0.6	8.8 (2.0)	10.9	1.8	N/A
B2200 Cold Shop	139.4 (604)	762.3	317.9	0.6 (18)	7.6	4.0	-0.5 (7.3)	0.7	0.5	8.0 (1.0)	9.0	0.9	N/A
B2300 Cable Vault	-23.4 (632)	275.3	195.1	0.5 (18)	21.3	5.0	-0.3 (6.9)	2.3	0.6	13.8 (2.0)	17.1	1.9	N/A
B2400 Staff Tunnel	19.2 (779)	575.6	359.6	3.8 (18)	18.0	6.7	-0.1 (6.9)	3.7	0.9	20.3 (2.0)	24.2	2.3	N/A

Group C
Radiological Characterization Results For Affected Systems

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate microR/hr			Tritium DPM/100 cm2
Package	**Mean (MDC)**	**Max**	**Std. Dev**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean MDC)**	**Max**	**Std. Dev.**	**Mean**
C0100 PASS	N/A	N/A	N/A	77,858 (5000)	300,000	126,236	1.5 (8.4)	8.0	3.7	1386 (423)	4161	1422.9	61.1 (39)
C0200 Waste Solid.	N/A	N/A	N/A	2344 (34)	4073	2069.9	-0.3 (8.4)	-0.3	0.0	23,333 (53,199)	219,340	53,199	399.9 (39)
C0300 Contain. Spray	N/A	N/A	N/A	25,185 (34)	39,530	14,366.8	11.5 (8.4)	24.7	11.5	2593 (4192)	22,862	4192	18.4 (39)
C0400 ECCS	N/A	N/A	N/A	70,933 (5000)	200,000	111,776	3.3 (8.4)	5.9	3.0	4416	34,960	6025	1377.8 (139)
C0500 RHR	N/A	N/A	N/A	76,000 (5000)	180,000	91,476.8	N/A	N/A	N/A	4882 (4112)	15,772	4112	23,617 (139)
C0600 Pri. Vent & Drains	N/A	N/A	N/A	50,585 (5000)	140,000	77,438	-0.2 (8.4)	0.0	0.2	165,583 (325,892)	1,326,311	325,892	548 (39)
C0700 SFP Cooling	N/A	N/A	N/A	13,693 (5000)	20,000	6466.2	3.4 (8.4)	10.1	5.8	829,672 (2,924,669)	16,945,540	2,924,669	31.0 (39)
C0800 Waste Gas	N/A	N/A	N/A	3251 (34)	6470	2854.0	-0.3 (8.4)	-0.3	0.0	3295 (5000)	23,554	4,999.5	5825 (39)
C0900 Pzr.	N/A	N/A	N/A	213,333 (5000)	360,000	128,582	N/A	N/A	N/A	41,636 (59,187)	376,269	59,187	82,468 (139)
C1100 RCS	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	53,580 (34,275)	181,323	34,275	N/A

Group C
Radiological Characterization Results For Affected Systems

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

Group D
Unaffected Systems

	Direct Beta DPM/100 cm2			Removable Beta DPM/100 cm2			Removable Alpha DPM/100 cm2			Exposure Rate microR/hr			Plant Derived Nuclides ?
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
D0100 Condens.	66.7	2184.5	425.2	-0.5	14.6	5.1	-0.3	2.3	0.7	1.9	2.1	0.1	No
D0200 Water Treat.	1250.8 (1937)	26,046.3	4898.1	38.1 (16)	945.1	162.8	13.6 (7.6)	362.2	61.9	12.6 (18)	44.2	17.7	No
D0300 Potable Water	526.2 (1089)	2638.6	767.7	6.7 (16)	29.2	6.9	0.4 (7.6)	9.1	2.3	4.5 (2.0)	7.1	1.6	No
D0400 Sewer	384.8 (1088)	5657.1	1051.5	3.2 (36)	32.2	8.9	0.0 (8.2)	1.9	0.6	11.3 (4.0)	16.2	4.3	No
D0500 Circ Water	162.0 (587)	811.8	295.1	3.1 (15)	14.7	4.2	-0.1 (6.9)	5.1	0.9	3.7 (5.0)	17.2	5.1	No
D0600 Svc Water	38.0 (1687)	1013.9	347.9	197.5 (37)	3133.7	658.5	-0.2 (8.6)	1.8	0.5	N/A	N/A	N/A	Yes
D0700 Fire Prot.	-35.6 (1257)	1114.7	240.2	2.4 (17)	20.6	5.2	0.2 (6)	2.5	0.9	N/A	N/A	N/A	No
D0800 Lube Oil	66.0 (1681)	723.4	253.6	2.5 (17)	22.3	6.1	0.1 (6)	2.5	0.7	6.0 (6.0)	12.3	5.5	Yes
D0900 Comp. Air	3677.5 (6324)	104,589	14,456.3	27.0 (17)	685.2	95.1	0.4 (6)	6.8	1.4	N/A	N/A	N/A	No
D1000 Aux Boiler	446.0 (2606)	2723.9	730.5	12.3 (17)	114.8	21.8	0.0 (6)	2.5	0.8	7.1 (5.0)	20.1	5.3	No
D1100 S/G	270.8 (1347)	2664.1	1067.4	9.2 (17)	47.5	11.1	0.3 (6)	2.5	1.0	35.0 (45)	66.8	44.9	No

Group D Unaffected Systems													
	Direct Beta DPM/100 cm2			**Removable Beta DPM/100 cm2**			**Removable Alpha DPM/100 cm2**			**Exposure Rate microR/hr**			**Plant Derived Nuclides ?**
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
D1200 Main Steam	-9.2 (1002)	4598.7	649.0	0.8 (36)	59.6	9.3	-0.3 (8.2)	2.2	0.7	N/A	N/A	N/A	No
D1300 Aux Steam	667.3 (2382)	11,786.6	1963.4	1.9 (36)	19.4	6.5	0.0 (8.2)	2.0	0.5	162.8 (219)	435.1	218.9	N/A
D1400 Turb Control	-38.3 (839)	416.5	189.7	-0.9 (19)	20.9	6.3	-0.4 (7.1)	0.8	0.5	0.8 (0.4)	1.6	0.4	No
D1500 Steam Dump	-216.5 (677)	64.1	139.9	-0.8 (19)	10.8	4.1	-0.5 (7.1)	0.8	0.5	N/A	N/A	N/A	N/A
D1600 Main Feed	-0.3 (640)	453.9	160.8	-1.2 (19)	24.2	6.3	-0.4 (7.1)	2.2	0.6	2.0 (2)	5.4	2.2	No
D1700 EFW	-136.5 (2414)	851.3	347.6	0.9 (18)	21.0	5.3	-0.3 (7.1)	3.6	0.8	N/A	N/A	N/A	No
D1800 Htr. Drain, Extract	42.4 (1182)	1864.3	323.3	-2.7 (19)	9.1	3.8	-0.4 (7.1)	2.2	0.6	0.9 (0.4)	1.3	0.4	No
D1900 Comp Cooling	1168.0 (4385)	21,644.3	6616.3	5.2 (36)	38.0	10.7	-0.1 (7.2)	2.0	0.3	10.1 (2)	12.8	2.0	No
D2000 Vac Prim	24.8 (1256)	672.1	257.8	1.6 (18)	14.2	4.8	-0.3 (7.1)	2.2	0.8	N/A	N/A	N/A	No
D2100 Amertap	107.5 (1200)	1880.2	507.5	2.2 (18)	15.9	5.4	0.1 (7.1)	3.6	1.1	N/A	N/A	N/A	No

Group D Unaffected Systems													
	Direct Beta DPM/100 cm2			**Removable Beta DPM/100 cm2**			**Removable Alpha DPM/100 cm2**			**Exposure Rate microR/hr**			**Plant Derived Nuclides ?**
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
D2200 Sealing Steam	23.3 (1067)	582.0	237.8	0.2 (18)	10.9	4.2	-0.5 (7.1)	0.8	0.5	N/A	N/A	N/A	No
D2300 Aux DG	31.7 (645)	535.3	210.8	3.1 (36)	31.8	9.3	0.1 (7.2)	2.0	0.7	N/A	N/A	N/A	No
D2400 Chem Sample	35.2 (1617)	645.5	251.2	307.2 (35)	4861.3	995.8	0.3 (7.8)	6.0	1.4	N/A	N/A	N/A	N/A
D2500 HP Drain	132.2 (1048)	594.8	260.3	-0.1 (18)	7.5	4.7	-0.4 (7.1)	0.8	0.6	N/A	N/A	N/A	N/A
D2600 Envir	336.6 (535)	1257.1	400.1	3.7 (14)	12.9	3.9	0.6 (6.9)	3.9	1.3	N/A	N/A	N/A	No
D2700 Admin HVAC	74.3 (789)	643.3	276.3	5.2 (18)	32.8	8.5	0.3 (7.1)	2.2	1.1	8.0 (0)	8.0	0.0	No
D2800 Info Ctr Hvac	156.2 (702)	627.8	256.9	0.6 (18)	10.9	4.7	-0.5 (7.1)	0.8	0.5	N/A	N/A	N/A	No
D2900 Turb HVAC	142.4 (577)	445.4	161.5	4.6 (14)	33.1	5.9	0.3 (6.9)	3.9	0.9	N/A	N/A	N/A	No
D3000 Staff HVAC	262.9 (779)	1286.3	366.0	2.2 (18)	15.9	6.0	-0.1 (7.1)	2.2	0.9	N/A	N/A	N/A	No
D3100 Svc HVAC	5346.8 (1082)	87,565.8	19,067.0	80.0 (14)	1445.0	247.1	0.6 (8.5)	5.9	1.3	22.4 (17)	51.4	17.4	Yes

Group D Unaffected Systems													
	Direct Beta DPM/100 cm2			**Removable Beta DPM/100 cm2**			**Removable Alpha DPM/100 cm2**			**Exposure Rate microR/hr**			**Plant Derived Nuclides ?**
Package	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	**Mean (MDC)**	**Max**	**Std. Dev.**	
D3200 H2/N2	12,037.3 (3059)	125,317	36,307.5	104.5 (14)	828.9	245.4	0.6 (8.5)	9.9	2.3	N/A	N/A	N/A	No
D3300 Turb Sumps	433.1 (1091)	5800.9	1166.9	8.1 (32)	33.6	9.0	0.0 (8.4)	1.8	0.8	10.8 (5.0)	15.9	4.9	Yes
D3400 LLWSB	457.0 (992)	3099.3	1300.0	7.1 (32)	27.4	8.7	0.1 (8.4)	6.0	1.3	N/A	N/A	N/A	No

Group R
Radiological Characterization Results For Affected and Unaffected Environs

										Exposure Rate microR/hr		
Package	**# Samples**	**# Positive Co-60**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**			**Mean**	**Maximum**	**Std. Dev.**
R0100 RA Yard West	58	23	0.62	3.29	55	10.99	156.0	N/A	N/A	N/A	N/A	N/A
R0200 Yard East	35	12	0.28	1.94	33	4.88	133.0	N/A	N/A	N/A	N/A	N/A
R0300 Roof Drains	7	4	4.09	11.2	6	0.33	0.53	N/A	N/A	N/A	N/A	N/A
R0400 Shoreline	27	1	0.08	0.08	27	0.34	0.98	N/A	N/A	N/A	N/A	N/A
R0500 Bailey Pt.	45	0	0	0	44	0.38	1.09	N/A	N/A	13.27	19.83	1.49
R0600 Ball Field	32	0	0	0	3	0.04	0.06	N/A	N/A	11.92	13.68	0.63
R0700 Constr. Debris	31	0	0	0	2	0.05	0.06	N/A	N/A	11.99	14.52	1.05
R0800 Admin. Parking	30	0	0	0	26	0.26	0.83	N/A	N/A	17.9	33.87	4.2
R0900 BOP	36	6	1.22	5.11	24	11.06	85.6	N/A	N/A	25.85	77.71	16.8
R1000 Foxbird Is	73	3	0.22	0.38	43	0.43	1.63	N/A	N/A	11.48	42.76	4.97
R1100 Roof Drains	15	0	0	0	3	0.07	0.09	N/A	N/A	N/A	N/A	N/A

Group R
Radiological Characterization Results For Affected and Unaffected Environs

										Exposure Rate microR/hr		
Package	**# Samples**	**# Positive Co-60**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**			**Mean**	**Maximum**	**Std. Dev.**
R1200 LLWSB Yard	30	0	0	0	5	0.10	0.13	N/A	N/A	N/A	N/A	N/A
R1300 ISFSI	30	0	0	0	5	0.12	0.28	N/A	N/A	12.92	31.2	3.68
R1400 Shorelines	30	0	0	0	30	0.20	0.35	N/A	N/A	N/A	N/A	N/A
R1500 Ash Pit Rubble	30	0	0	0	9	0.08	0.21	N/A	N/A	11.34	12.63	0.63
R1600 Eaton Farm Land	60	0	0	0	59	0.46	1.43	N/A	N/A	12.07	17.8	2.06
R1700 Land North of Ferry Rd	60	0	0	0	50	0.47	1.55	N/A	N/A	9.65	13.74	1.56
R1800 Bailey Farm Land	31	0	0	0	22	0.27	0.76	N/A	N/A	10.63	14.57	1.31
R1900 Bailey Cove	14	0	0	0	14	0.27	0.37	N/A	N/A	N/A	N/A	N/A
R2000 Diffuser	5	2	0.1	0.12	4	0.10	0.13	N/A	N/A	N/A	N/A	N/A

Group R
Radiological Characterization Results For Affected and Unaffected Environs

										Exposure Rate microR/hr		
Package	**# Samples**	**# Positive Co-60**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**			**Mean**	**Maximum**	**Std. Dev.**
R2100 Warehse Yard	30	0	0	0	4	0.13	0.33	N/A	N/A	8.41	10.62	1.33
R2200 Backgrnd	62	0	0	0	62	0.35	1.4	N/A	N/A	11.37	13.59	1.26
R2300 SFP Substation	16	1	0.14	0.14	15	0.35	0.81	N/A	N/A	26.14	29.4	1.46
R2400 IT Duplicates	44	0	0	0	9	0.48	1.62	N/A	N/A	N/A	N/A	N/A

R2500 Investigation Package							
Package	**# Samples**	**# Positive Co-60**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**
R0500 Bailey Pt	8	3	11,218.5	33,600.0	7	0.13	0.21
R0600 Ball Field	15	0	0	0	5	0.16	0.29
R0700 Construction Debris	40	0	0	0	3	0.04	0.06
R0800 Admin Parking Lot	15	0	0	0	14	0.17	0.33
R1000 Foxbird Is	10	0	0	0	7	0.13	0.21
R1300 ISFSI	10	2	0.43	0.45	4	0.07	0.12
R1600 Eaton Farm Land	5	0	0	0	2	0.27	0.29
R1800 Bailey Farm Land	20	0	0	0	13	0.10	0.15

R2501 Investigation Package							
Package	**# Samples**	**# Positive**	**Mean Co-60 pCi/g**	**Max Co-60 pCi/g**	**# Positive Cs-137**	**Mean Cs-137 pCi/g**	**Max Cs-137 pCi/g**
R0900 BOP	41	15	0.09	0.13	41	13.59	66.4
R1000 Foxbird Is.	26	2	0.08	0.11	24	2.53	10.0
R2500 Contractors Parking	27	0	0	0	4	0.20	0.31

MAINE YANKEE

LTP SECTION 3

IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

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ATTACHMENT A **Drawings Associated with Specific Decommissioning Tasks**

3.0 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Summary and Introduction

3.1.1 Summary

This section of the LTP describes the remaining dismantlement activities at MY. The remaining dismantlement activities will be performed in accordance with the regulations in 10 CFR 50, will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment.

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

The remaining systems, and components will be decontaminated or, removed, packaged, and either transported to an offsite processing facility, transported directly to a low-level radioactive waste disposal facility, or handled by other alternatives in accordance with applicable regulations. In most cases decontamination of plant structures will be completed sequentially with equipment removal. Decontamination of structures will include a variety of techniques ranging from water washing to surface material removal. structural material may be packaged and either transported to an offsite processing facility, or transported directly to a LLRW disposal facility. Alternative disposal methods, in accordance with applicable State and Federal regulations, will be used when available.

Following the removal or decontamination of systems, components, and structures, a comprehensive final status survey (FSS) will be completed as described in section 5 of this LTP. This survey will verify that radioactivity has been reduced to sufficiently low levels, as stipulated in 10 CFR 20.1402 to allow the unrestricted release of the site. Upon completion of the FSS, MY will generate a FSS report to support license termination or release of site areas, as appropriate.

3.1.2 Phased Release and License Termination

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

a. Following the completion of LTP activities in a given area, MYAPC will provide to the NRC a report covering the area which it seeks to release from the Part 50 license. This report will contain the information which the NRC needs to make its 10 CFR 50. 82(a)(11) determination and will include:

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

   2. A certification that the remaining dismantlement activities described in the license termination plan have been performed.

   3. FSS results for the area.

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

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

   6. An evaluation of the impact on the following license programs for the site lands remaining within the domain of the Part 50 license: ODCM, Emergency Plan, Security Plan, Fire Protection Plan, QA Plan, Training Plan, and Post Shutdown Decommissioning Activities Report.

   Upon satisfactory NRC review, the NRC will provide a duly authorized certification or license amendment to MYAPC that the NRC has made the required 10 CFR 50.82(a)(11) determination regarding the area to be released from the Part 50 license and that the area is henceforth released from the Part 50 license. This certification or license amendment will carry the same authority as that associated with terminating a license under 10CFR50.82(a)(11).

Once an area is so released, it is understood that the NRC will not require any additional surveys or decontamination of these areas by MYAPC in response to future NRC criteria or standards, or third party survey results, unless, pursuant to 10 CFR 20.1401(c), the NRC determines that the criteria of this subpart were not met and residual activity remaining at the site could result in significant threat to public health and safety

3.1.3 Waste Disposal Strategies

The strategies for waste disposal are included in the MY Decommissioning Waste Management Plan which addresses issues such as; estimates of the quantity of radioactive material to be released, control mechanisms, and radioactive waste characterization.

3.1.4 Introduction

The goal of the D & D of MY is to return the property to greenfield, by demolishing property structures and facilities to a level of 3 feet below present grade, with the exception of Central Maine Power's (CMP's) transmission facilities, the Eaton Farm House and Barn, the Bailey Farm House and Barn, the Ball Field, and the Montsweag Pump House. Structure and facility foundations 3 feet below present grade will be backfilled with concrete demolition debris. However, since repowering is still an option at MY, some structures, facilities and commodities may be released for unrestricted use intact, and in place.

In general, the dismantlement activities will be carried out in the following four phases:

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

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

Phase 1: Prepare Site & Release Non-Impacted Areas

The preparations period began with permanent plant closure on August 7, 1997. This phase involves the demolition of miscellaneous tanks, buildings, fences and vehicle barriers, etc. to allow ease of access to the site. Additionally, as demonstrated by this LTP, no radiological or non-radiological contaminants were found North of Old Ferry road, or West of Bailey Cove, and these areas are therefore designated NON-IMPACTED. These areas are expected to be released on an early basis in accordance with 10 CFR Part 20 Subpart E and 10CFR50.82 (a)(11)(i) and (ii).

Additional activities that have been completed or are in process during the preparations period include but are not limited to the following:

a. Assessment of Systems, Structures, and Components (SSCs)

b. Deactivation/removal of plant SSCs

c. Plant systems and components are decontaminated and/or removed, as they are no longer required to support the decommissioning activities or impact worker health and safety.

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

e. The FSAR was revised to reflect the permanently defueled plant condition and was re-titled "Defueled Safety Analysis Report" (DSAR). The DSAR was submitted to the NRC on February 6, 1998. Additional licensing basis documents were also revised and submitted to reflect the plant's defueled condition (defueled security plan and emergency plan).

f. Submission of the PSDAR including environmental considerations

g. Site characterization

h. Section 2 of this LTP presents a summary of the site characterization and continued characterization at MY. This summary was used at MY in development of decommissioning work packages (including asbestos abatement and source term reduction), schedule, and this LTP.

Phase 2: Dismantle Commodities & Decontaminate Structures

Commodities and structures are designated for reuse/repower, recycling, onsite demolition, and offsite disposal. In accordance with pathway analyses, recyclable commodities are dismantled, and structures and commodities designated for reuse/repower, onsite demolition or offsite disposal, are decontaminated accordingly.

Phase 3: Demolish Buildings & Restore Site

Activities will be reviewed during planning to ensure no adverse effect on the "Spent Fuel Pool Island (SFPI)" (i.e. walls of adjacent buildings that have a support function of the SFP will remain intact). Concrete buildings at levels 3 foot above grade will be demolished. concrete rubble (meeting the unrestricted use criteria) will be placed as fill in remaining foundations located 3 feet below present grade, and/or used as fill on other parts of the site. Other buildings are designated for either reuse/repower, recycling, or onsite / offsite disposal; and are dispositioned accordingly.

Phase 4: Establish ISFSI

The ISFSI is designed and built during this Phase.

Remaining dismantlement activities will consist primarily of phase 2 & 3 activities. However, some systems extend through out many areas of the plant. D&D will primarily be by area and not necessarily by system. Each D&D activity is evaluated by appropriate survey.

3.2 Overview

Activities are ongoing for removal of the Reactor Coolant System (RCS). MYAPC's decommissioning schedule details among other services, the removal of the reactor coolant pumps, steam generators, pressurizer, reactor vessel internals, reactor vessel head, and reactor vessel, and large bore piping.

A SFPI has been created, with the spent fuel pool and other systems associated with spent fuel storage electrically and mechanically isolated. Electrical Power is fed from the 115KV incoming line with a back up diesel generator specifically for security, but available for the SFPI. An

industrial water-to-air cooling system has replaced the primary component cooling /service water systems that serviced the spent fuel pool cooling and clean up system. Makeup water is supplied from the PWST with back up from the Wiscasset water supply and the fire protection service system. A portable mix tank and pump batches borated water when required in the make up for the spent fuel pool.

Mechanical SSCs related to nuclear safety remain safety related, and maintained under nuclear grade controls. They are as follows:

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

QA related items:

a. Fuel assemblies

a. Boric acid

b. Weld rod

c. Reagents for SFP purification and liquid radwaste processing

d. Packaging of radioactive material for transport (as required by 10CFR71)

e. Structures (seismic only)*

1. Primary Auxiliary Building (only as it affects the fuel building and SFPI)

f. Special nuclear material

g. Calibration services for controlled measuring and test equipment

h. Liquid penetrant materials

* Limited to the portion of each structure designed to safely resist hypothetical earthquake loads.

3.2.1 Waste Characterization

The MY Decommissioning Project Waste Management Plan includes waste disposal strategies, and addresses issues such as: estimates of the quantity of radioactive material to be released, control mechanisms, and radioactive waste characterization. Radioactive waste has been characterized by sending representative samples for Part 61 analysis. Table 1A list the nuclides for which the samples were analyzed. Table 1B displays typical sample Part 61 analysis results.

TABLE 1A Nuclides checked for by 10CFR61 analysis					
Nuclide	**Principal Emission★**	**Nuclide**	**Principal Emission★**	**Nuclide**	**Principal Emission★**
Ag-110m	Gamma	Zr-93	beta	*Nb-94 (in activated metal - so C-14, Ni-59, Ni-63)	Gamma
Am-241	alpha	Sn-126	beta	+Kr-85	Gamma
C-14	beta	Iso-U	alpha	#Cr-51	Gamma
Cm-242	alpha	K-40	Gamma	#Fe-59	Gamma
Cm-243/244	alpha	Zn-65	Gamma	#Nb-95	Gamma
Co-57	Gamma	Eu-154	Gamma	#Zr-95	Gamma
–	--	Eu-155	Gamma	#Mo-99	Gamma
Co-60	Gamma	Eu-152	Gamma	#I-131	Gamma
Cs-134	Gamma	Tl-208	Gamma	#Xe-133	Gamma
Cs-137	Gamma	Bi-212	Gamma	#Ba-140	Gamma
Fe-55	ec	Pb-212	Gamma	#La-140	Gamma
H-3	beta	Bi-214	Gamma	#Ce-141	Gamma
Mn-54	Gamma	Pb-214	Gamma	#Sn-113	Gamma
Ni-59	ec	Ra-226	Gamma	#Sb-124	Gamma
Ni-63	beta	Ac-228	Gamma	#Ru-103	Gamma
Pu-238	alpha	Pa-234m	Gamma	#Co-58	Gamma
Pu-239/240	alpha	Th-234	Gamma		
Pu-241	beta	U-235	Gamma		
Sb-125	Gamma	Be-7	Gamma		
Sr-90	beta	Ce-144	Gamma		
*Tc-99	beta	Sb-126	Gamma		
* +I-129	beta	Sn-126	Gamma		

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

Table 1B - 10CFR61 Sample Analysis Results (Typical)

NUCLIDE	1 RESIN SAMPLE 8/21/96	2 LIQUID FILTERS 9/4/96	3 SMEAR ACTIVITY 6/18/97	4 CAVITY DRAIN DOWN FILTER ACTIVITY 7/23/96	5 UPENDER SMEAR ACTIVITY 6/10/98	6 NORTH OF FOREBAY SOIL ACTIVITY 3/28/98	7 PWST 1 SOIL ACTIVITY 3/28/98	8 PWST 2 SOIL ACTIVITY 3/28/98	9 #3 SG BOWL SMEAR ACTIVITY 3/28/98	10 O/A TRENCH CONCRETE CORE ACTIVITY 3/27/98
H-3	2.00E-01		1.86E-02	1.40E-01		1.7	2.6	3.2	15500	1600
C-14	8.51E-02							0.2	4270	843
Mn-54	8.01E-01		1.63E-03	1.00E-02	1.00E-03				1260	
Fe-55	9.81E+00	7.49E-01	2.46E-01	2.58E-01	4.33E-02				160000	16200
Co-57	2.07E-02		2.37E-04	4.79E-04						
Co-58	7.70E-02		4.15E-02						674	
Co-60	9.68E+00	1.64E+00	4.48E-01	3.61E-01	1.14E-01	1		1	147000	4450
Ni-59	1.04E-01									
Ni-63	1.42E+01	1.40E+00	3.34E-01	8.97E-02	3.86E-02	7.9			18700	35600
Zn-65										
Sr-90	2.38E-01			2.74E-02		1.1	1.9	1.6	370	51.3
Zr-93										
Nb-94										
Tc-99									6920	2310
Ag-110m			1.37E-03							
Sb-125	2.72E-01	2.62E-03	5.81E-03	1.61E-03	2.95E-03				2110	173
Sn-126										
I-129										
Cs-134	2.00E+01		5.54E-03			0.3	0.3			32.9
Cs-137	3.72E+01	3.35E-03	8.06E-02	1.03E-02		132	6.5	82.5		849
Ce-144				2.45E-03						
Eu-152										
Eu-154										
Eu-155										
U-234						0.4	0.3	0.6		
U-235						0.1		0.1		
U-238						0.5	0.4	0.6		
Pu-238	6.67E-04	1.53E-04	1.45E-05	1.83E-04	1.20E-05	0.7	1.3	0.7	6.9	30.1
Pu-239/240	2.79E-04	1.91E-04	2.24E-05	6.02E-04	1.53E-05		0.4		5.3	11.9
Pu-241	2.05E-02	1.65E-02	1.98E-03	2.32E-02	9.60E-04				315	96.6
Am-241	3.56E-04	2.71E-04	3.00E-05	2.77E-04	1.55E-05	0.2		0.2	3.4	10.7
Cm-242	1.64E-04			3.21E-06	6.10E-06					
Cm-243/244	4.53E-04	2.34E-04	1.19E-05		8.50E-06				1.8	

3.2.2 Radioactive Waste Projections

Any data provided herein are estimated values and may or may not represent actual final volumes. Alternate means and methods may be utilized when appropriated to reduce these volumes. The projected activities and volumes of radioactive material generated are summarized in the Table 2.

Table 2
Projected Activities and Volumes

Activity	Curies	Volume
Reactor vessel and internals	2,600,000. Ci	11,527 Cu.Ft.
components	1,600. Ci	27,000 Cu.Ft.
Activated concrete	388. Ci	23,000 Cu. Ft.
debris	0.10 Ci	163,000 Cu. Ft.
Radioactive water	0.03 Ci	850,000 Gallons
soil	0.01 Ci	25,000 Cu. Ft.

The estimated radioactive waste volumes are published in Table 3 (MY's Decommissioning Project Waste Management Plan - Table 1). They are less than the estimated volumes used in NUREG/CR-0130. Each specific waste category is discussed in section 6.3 of the MY Decommissioning Project Waste Management Plan

Table 3
Waste Volume Estimates

Waste	Source	Amount	Disposition Path
High level waste (HLW)	Spent fuel	1434 fuel assemblies	ISFSI) for interim storage US DOE/disposal
Greater than class C (GTCC)	Reactor vessel internals Segmentation filters Pre-existing filter	227 ft^3 TBD TBD	ISFSI/USDOE
Large nuclear steam supply system (NSSS) components	Reactor pressure vessel (RPV) Non-GTCC RPV hardware	9,500 ft^3 1,500 ft^3	Direct burial offsite

Table 3 Waste Volume Estimates			
Waste	**Source**	**Amount**	**Disposition Path**
Large nuclear steam supply system (NSSS) components: (continued)	RPV head	300 ft^3	Approved processor
	Pressurizer	2,200 ft^3	
	Reactor coolant pumps, motors & assemblies	4,800 ft^3	
	Steam generators (3)	20,000 ft^3	
Scrap metal (not including above categories)	Radioactive-contaminated metal	3,100 tons (150,000 ft^3)	Approved processor offsite disposal
	Non-radioactive-contaminated metal	6,200 tons (300,000 ft^3)	
Dry active waste (DAW)	Non-metallic trash	13,000 ft^3 (100 tons)	Approved processor
Resin	Liquid radioactive waste processing	400 ft^3	Approved processor for volume reduction and disposal
	Spent fuel pool purification	150 ft^3	
	Pre-existing	200 ft^3	
soil	Radioactive areas	25,000 ft^3	Direct burial offsite
Concrete	scabble residue, activated concrete (some activated concrete will remain)	11,400 ft^3	Direct burial offsite
	Concrete below the unrestricted use criteria	1,200,000 ft^3 (44,000 yd^3)	Onsite fill
Mixed waste	None identified (see also PCBs below)	TBD	Approved processor for treatment and/or disposal
Radioactive water	Aqueous waste from decontamination, washing, cutting, cooling, segmentation, showers, laboratory operations, spent fuel pool	850,000 gallons	Onsite and/or offsite treatment and discharge
Hazardous waste	Possibly used/unused chemical materials Mercury-containing wastes (switches, manometers)	220 ft^3	Approved processor for treatment and/or disposal

3.2.3 System Deactivation

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

Table 4
Status of Major MY Systems, Structures, and Components
As of 1 November, 1999

System/Component/Structure	Required for SFP	Status
Reactor coolant system	NO	Preparations for removal are ongoing
Reactor vessel internals	NO	Preparations for removal are ongoing
Reactor vessel	NO	Preparations for removal are ongoing
Steam generators	NO	Preparations for removal are ongoing
Reactor coolant pumps	NO	Preparations for removal are ongoing
Pressurizer and pressurizer quench tank	NO	Preparations for removal are ongoing
Chemical & volume control system	NO	Preparations for removal are ongoing, portable system provides support of borated make-up to SFP
HP/LP safety injection system	NO	Preparations for removal are ongoing
Containment spray system	NO	Preparations for removal are ongoing
Primary component cooling water system	NO	Preparations for removal are ongoing
Secondary component cooling water system	NO	Preparations for removal are ongoing
Service water system	NO	Preparations for removal are ongoing

Table 4 Status of Major MY Systems, Structures, and Components As of 1 November, 1999		
System/Component/Structure	**Required for SFP**	**Status**
Potable water system Wiscasset water system	YES	Preparations for partial removal are ongoing, portion remaining in service to support SFP System and site needs
Spent fuel pool and fuel handling equipment	YES	Preparations for partial removal are ongoing, portion remaining in service to support SFP System
Spent fuel pool cooling system	YES	Will remain in service as long as materials are stored in the SFP
Spent fuel pool cooling and demineralizer system	YES	Will remain in service as long as materials are stored in the SFP
Steam generator blowdown system	NO	Preparations for removal are ongoing
Primary water make-up system	NO	Preparations for removal are ongoing
Refueling water storage tank	NO	Preparations for removal are ongoing
Plant effluent system	YES	Preparations for partial removal of components not required for support of spent fuel pool are ongoing
Containment ventilation systems	YES	Preparations for removal are ongoing as portions of system are no longer required to support decommissioning or SFP
Fuel building ventilation systems	YES	Will remain in service as long as materials are stored in the SFP
Auxiliary building ventilation systems	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing.
Instrument and service air systems	NO	Preparations for removal of Components not required for support of spent fuel pool are ongoing.

Table 4 Status of Major MY Systems, Structures, and Components As of 1 November, 1999		
System/Component/Structure	**Required for SFP**	**Status**
Waste gas system	NO	Preparations for removal are ongoing
Solid radioactive waste system	YES	Will remain in service as long as materials are stored in the SFP, and support decommissioning activities after removal of SFP contents.
Liquid radioactive waste system	YES	Will remain in service as long as materials are stored in the containment Building, Auxiliary Building, or SFP, and support decommissioning activities after removal of SFP contents.
Radiation monitoring system	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing
Process sampling system	NO	Preparations for removal are ongoing
Electrical systems	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing in Accordance with decommissioning
Fire protection systems	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing (Back Up make-up water supply to Spent Fuel Pool)
Containment building	NO	Preparations for removal are ongoing
Auxiliary building (has a wall that supports the SFPI systems)	YES	Preparations for partial removal of Components not required for support of spent fuel pool are ongoing in Accordance with decommissioning schedule

Table 4 Status of Major MY Systems, Structures, and Components As of 1 November, 1999		
System/Component/Structure	**Required for SFP**	**Status**
Turbine/generator building (has a wall that supports the SFPI systems)	NO	Preparations for removal are ongoing
Low level waste storage building	YES	Will remain in service as long as materials are stored in the SFP, and support decommissioning activities after removal of SFP contents
Warehouse #5	NO	Preparations for removal are ongoing

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

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

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

3.2.4 Occupational Exposure

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

3.2.5 Public Exposure

Measures will be employed to control and monitor potential effluents.

3.2.6 Detailed Planning and Engineering

Detailed planning precedes initiation of each specific activity, and includes engineering design, ALARA planning, and refinement of cost, schedule, and required resources estimates.

Table 6 provides estimated exposure per area of activity and a schedule for the decommissioning activities.

Components and non-concrete building materials will be radiologically evaluated. Those components and non-concrete material will be shipped offsite to a vendor for processing or disposal. Components/materials determined to meet the operational free release criteria will be removed and scrapped. This process will leave empty structures that are essentially concrete walls, floors and ceilings.

Characteration has determined that concrete within the Radiologically Controlled Area (RCA) of the site shows the following:

a. Painted concrete has surface contamination up to 1 million dpm/100 cm2 (worst case) which is amenable to surface remediation techniques such as wiping, washing, power washing or abrasive surface removal.

b. Bare concrete has surface contamination, absorbed contamination and activation products within the concrete matrix. Surface contamination levels are similar to those for painted concrete. Absorbed activity has been found to penetrate to a depth of approximately 0.5 inches.

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

d. Surface abrasive or surface removal remediation techniques may generate airborne radioactivity. Airborne activity will be controlled within the requirements of 10 CFR Part 20 and measured using standard processes and procedures existing within in the radiation protection program. These processes and procedures have proven successful for controlling decontamination and demolition activities in the past while protecting the health and safety of the workers and the public.

3.2.7 Contamination Control

When applicable, during demolition engineering controls such as misting will be applied to concrete surfaces. The proposed control mechanisms to ensure areas are not re- are summarized here. A detailed description of the control mechanisms to ensure areas are not re-contaminated is given in LTP section 5 - FSS.

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

a. Personnel training

b. Installation of barriers to control access to surveyed areas

c. Installation of barriers to prevent the migration of contamination from adjacent overhead areas

d. Installation of postings requiring personnel to perform contamination monitoring prior to surveyed area access

e. Locking entrances to surveyed areas of the facility

f. Installation of tamper-evident labels

3.3 Systems and Structures Review and Discussions

Systems, components, and structures outside the RCA that have slight contamination will be

decontaminated (if necessary), released, or disposed of as radioactive waste.

The D&D status of these systems, structures, and components, as of November 1,1999, is summarized in Table 4. Table 5 contains a list of major components to be removed during each year of the project. As indicated by Table 4, the majority of radiologically systems and components not required to support the storage of spent fuel have been abandoned and accepted for decommissioning in accordance with the plant program (SERT file 98-136). SERT designates them as ready to be "accepted for decommissioning". These SSCs will be deactivated, dismantled, and disposed of in accordance with the MYAP PSDAR and published schedule (partially reconstructed chronological order in Table 6).

3.4 D&D Methods

3.4.1 Decontamination

systems and components will typically be; dismantled and sent to an offsite processing facility or sent to a low level radioactive waste disposal facility. Potentially or slightly components (i.e. lighting ballast, mercury switches, etc) will be decontaminated onsite for release. MY previously performed a chemical decontamination of the RCS system. This reduced radiation dose rates prior to dismantlement and general area decontamination.

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

Application of coatings and hand wiping may be used to stabilize or remove loose surface contamination. Airborne contamination control and waste processing systems are used as necessary to control and monitor releases. If structural surfaces are washed to remove contamination, controls are implemented in accordance with approved plant procedures to ensure that wastewater is collected for processing by liquid waste processing systems.

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

a. Precautions are taken to ensure that in the unlikely event liquid inadvertently is discharged from the tank it will be captured for processing by a liquid waste processing system;

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

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

Concrete that is or activated may be removed and sent to a low level radioactive waste disposal facility, or remediated to below unrestricted use criteria, or handled by other methods in accordance with applicable regulations. Removal of concrete will be performed using methods that control the removal depth to minimize the waste volume produced. Appropriate engineering controls for control of dust and debris will be used to minimize the spread of contamination and reliance on respiratory protection measures.

3.4.2 Dismantlement

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

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

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

3.4.3 General D&D Considerations

The following general decontamination and dismantlement considerations, as applicable, will continue to be incorporated into packages during the decommissioning period.

Dismantlement activities are reviewed to ensure that they do not impact safe storage of fuel in the SFP. During the decommissioning period, dismantlement activities will be reviewed to ensure that they do not impact safe storage of fuel and GTTC in the ISFSI licensed under 10 CFR 72 or 10 CFR 50. Work packages are implemented in accordance with administrative controls. When required some or all of the following are considered for evaluation;

a. Involve an unreviewed safety question or changes to the technical specifications as specified in 10 CFR 50.59; or

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

c. Increase the specified derived concentration guideline levels (DCGL's) or area factors

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

e. Increase the action thresholds associated with survey unit classification investigations. (Investigation Levels)

Where practical for ALARA purposes, temporary shielding is used during decommissioning activities. Some dismantlement activities may be performed under water for shielding purposes as well as contamination control.

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

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

a. Operation of the appropriate portions of the containment ventilation and purge system, or an alternate system, during decontamination and dismantlement activities in the containment building;

b. Operation of the appropriate portion of the auxiliary building ventilation system, as required.

c. Operation of the appropriate portion of the fuel building ventilation

system to support the fuel building.

NOTE: The auxiliary building roof physically supports the fuel building ventilation ducts.

d. Use of local HEPA filtration systems for activities expected to result in the generation of airborne radioactive particulate (e.g. grinding, chemical decontamination, or thermal cutting of components)

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

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

b. D&D of systems, structures, and components by decontamination in place, removal and decontamination, or removal and disposal;

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

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

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

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

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

in preparation for disposal or release)

3.4.4 Temporary Systems to Support D&D

Temporary liquid and solid waste processing systems may be used during decommissioning for processing plant waste. These systems may include filters and/or demineralizers and may be used at one or more locations in the waste-processing path.

Localized temporary ventilation equipment and HEPA filtration may be used to supplement building ventilation and minimize the spread of radioactive particulate contamination.

3.5 Coordination with Other Regulatory Agencies

The activities of packaging and shipping large radioactive components (i.e. reactor vessel, pressurizer, steam generators, reactor coolant pumps, and large bore piping) requires permitting by or notification to; the State of Maine Department of Environmental Protection (MDEP), United States Nuclear Regulatory Commission, United States Department of Transportation (USDOT), United States Coast Guard (USCG), South Carolina Department of Health and Environmental Control (SCDHEC), and other States requiring transportation permits, notifications, and/or permits for use of State Licensed Facilities.

3.6 Major Decommissioning Activities

10CFR50.2 defines "major decommissioning activity" as any activity that results in permanent removal of major radioactive components, permanently modifies the structure of the containment, or results in dismantling components (seperating and packaging GTCC waste) for shipment in accordance with 10CFR 61.55.

Those activities are summarized as follows:

a. Removal of the steam generators and the pressurizer. The external surfaces will be decontaminated as required, and all openings will be sealed-welded. These components will serve as their own disposal containers.

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

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

disposal site, and

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

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

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

g. Segment the RCS and other large-bore piping, decontaminate to acceptable limits for offsite direct disposal, size reduction/disposal, or offsite recycle as appropriate considering the residual activity level.

h. The containment equipment access will be modified (with closure capability) to facilitate moving a multi-wheeled transporter into containment for loading/ removal of large components. Interior surfaces may be damaged during decontamination activities (which require removal of concrete to a depth of several inches).

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

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

During 1999, 2000, and 2001, Maine Yankee intends to remove containment piping and components. Reactor vessel internals will be partially removed in 2000 and the reactor vessel itself (with some internals) will be removed in 2001, from the 10CFR 50 licensed area of the MY site. The reactor vessel will be loaded into a transport/disposal container (awaiting NRC approval). The vessel and its container will be moved onto a sea-going barge and transported via the Atlantic Ocean, Intercoastal Waterway, then up the Savannah River where it will be

offloaded at the Savannah River Project (SRP). After barge offloading, the vessel package will be transported overland for disposal at the Chem-Nuclear low level radioactive waste disposal facility, near Barnwell, South Carolina. These activities will be coordinated with the State of Maine, US DOT, US Coast Guard, USNRC, South Carolina Department of Health and Environmental Control (SCDHEC), and the other States requiring notification of the shipment.

3.7 Materials Removal and Sequence

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

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

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

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

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

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

b. Fuel Building Overhead Crane;

c. Equipment Room Monorails; and

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

e. Turbine Hall Overhead Crane

f. Plant Equipment Monorail Systems

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

The containment building polar crane is capable of reaching most locations inside the containment building and can handle large, heavy loads. The fuel building yard crane has access into the fuel building via a roll up door for movement of heavy components. The fuel building crane is used to some extent for movement of components in the spent fuel pool.

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

3.8 Remaining Decommissioning Activities

Initiate dismantlement activities throughout the plant. In relation to plant commodities and internal structures, the project developed the "Rip and Ship" philosophy approach to dismantlement. It provides a safe, productive, and cost-effective means to commodity removal and accelerates access to the building surfaces for decontamination efforts.

NSSS components removal should be completed approximately three and one half years following cessation of operations. It is expected that the majority of plant structures and facilities will be decontaminated and dismantled within seven years of cessation of operations as listed below in Table 5.

Table 5
Major MYAPC Area/Systems, Structures, and Components Removed (By Year)
1999
Backlands
Reactor coolant pumps & motors (HA)
Rx missile shield (control rod drive shield)
Rx cavity fuel upender & carriage
Boric acid mix tank & pump
Safety injection tank #1 & #3
Iodine filter
Aux charging pump
Charging pump (5) "also next year"
Letdown system piping
Water treatment
SCC/PCC heat exchangers
SCC/PCC pumps
Sewage treatment
2000
Pressurizer
PZR quench tank
Steam generators
Rx vessel head
CEDM
Rx vessel internals
Regenerative heat exchanger

Table 5
Major MYAPC Area/Systems, Structures, and Components Removed (By Year)
SI tank #2
HPSI/LPSI
Warehouse #5
Charging pumps
Evaporator
PLAD pumps
Turbine steam system
Circulating water system
RWST/SCAT
Primary drain tank
2001
Neutron shield tank
Rx pressure vessel
Letdown heat exchanger
Primary water pump
Boric acid storage tank(BAST)
Blowdown tank (BDT)
VCT
Waste gas
Turbine building* (Turbine building feedwater heaters)
2002
Emergency feedwater

Table 5
Major MYAPC Area/Systems, Structures, and Components Removed (By Year)
Spent fuel pool * (limited)
Boron waste storage tank (BWST)
Primary water system
115 KV yard*limited
2003
OFFGAS - Off gas stack demolition
Fire protection system * limited
2004
Spent fuel pool
Spent fuel building
Fire protection system
Aux boiler and stack
PW - potable water connection
2026 or after DOE takes possession and removes the stored Materials
ISFSI site D&D with remediation as required

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

Table 6 list estimated exposure/area of activity and decommissioning activities chronologically listed (by Acronym as shown on the current decommissioning schedule). Attachment A at the end of this document provides reference to the area of decommissioning activity and its Acronym, including drawings which show the components and areas being D&D (with same acronyms as shown on its scheduled time line charts). The decommissioning schedule may be revised during the project. However, the LTP does not require revision to describe the schedule changes since this section is a general description of D&D activities and options. Existing lines of communication (ie. weekly telecom) will be utilized to inform the NRC of any

significant changes to major milestones schedule.

Equipment and materials will be removed from areas unless the radiation surveys indicate that the structures can be released for unrestricted access and conventional demolition. On or before August 20, 2002, after the fuel is transferred to the ISFSI, the SFP and its supporting systems are scheduled for D&D.

Table 6	
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled	
Area/Activity	**Exposure**
(Bolded Acronyms are for affected areas of decommissioning activities)	
DC.2 Period 2 (Decommissioning)	
DC.2.01 NSSS Removal	
DC.2.01.01 **Reactor coolant piping**	
DC.2.01.02 **Pressurizer relief tank**	
DC.2.01.03 **Reactor coolant pumps and motors**	
DC.2.01.04 **Pressurizer**	
DC.2.01.05 **Steam generators**	
DC.2.01.06 **CRDMs & service structure removal**	
DC.2.01.07 **Reactor vessel internals**	
DC.2.01.08 **Reactor vessel**	93.951 REM
DC.2.03 System removal	
DC.2.03.01 Containment	
DC.2.03.01.01 **Cbl-1**	97.114 REM
DC.2.03.01.02 **Cbl-2**	65.745 REM
DC.2.03.01.03 **Cbl-3**	63.171 REM
DC.2.03.01.04 **Cbl-4**	11.592 REM
DC.2.03.01.05 **Cbl-5**	25.411 REM
DC.2.03.01.06 **Cbl-6**	22.608 REM
DC.2.03.01.07 **Cbl-7**	6.485 REM
DC.2.03.01.08 **Cbl-8**	43.334 REM
DC.2.03.01.09 **CB2-1**	19.313 REM
DC.2.03.01.10 **CB3-1**	19.615 REM
DC.2.03.01.11 **CB3-2**	26.683 REM
DC.2.03.01.12 **CB3-3**	6.078 REM
DC.2.03.01.13 **CB3-4**	4.042 REM
DC.2.03.01.14 **CCG**	3.105 REM
DC.2.03.01.15 **CEHO**	3.871 REM
DC.2.03.01.16 **CICI L DC.**	6.533 REM
DC2.03.01.17 **CPHO**	.728 REM
DC.2.03.01.18 **CPLE**	.173 REM

Table 6	
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled	
Area/Activity	**Exposure**
(Bolded Acronyms are for affected areas of decommissioning activities)	
DC.2.03.02 Primary Auxiliary Bldg	
DC.2.03.02.01 **P21A**	.742 REM
DC.2.03.02.02 **P21B**	6.387 REM
DC.2.03.02.03 **P21C**	22.718 REM
LDC.2.03.02.04 **P21D**	9.160 REM
DC.2.03.02.05 **P21E**	39.169 REM
DC.2.03.02.06 **P21H**	16.495 REM
DC.2.03.02.07 **P21L**	1.418 REM
DC.2.03.02.08 **P21.s**	2.799 REM
DC.2.03.02.09 **P21V**	.956 REM
DC.2.03.02.10 **PLAD**	22.184 REM
DC.2.03.02.11 **PLBA**	13.790 REM
DC.2.03.02.12 **PLCP**	5.054 REM
DC.2.03.02.13 **PLDC**	1.551 REM
DC.2.03.02.14 **PLEC**	13.751 REM
DC.2.03.02.15 **PLLA**	38.761 REM
DC.2.03.02.16 **PLPA**	28.907 REM
DC.2.03.02.17 **PLPD**	11.122 REM
DC.2.03.02.18 **PLPT**	30.815 REM
DC.2.03.02.19 **PLPW**	.289 REM
DC.2.03.02.20 **PU48**	.485 REM
DC.2.03.02.21 **PUDD**	.512 REM
DC.2.03.02.22 **PUEC**	5.921 REM
DC.2.03.02.23 **PUFN**	.506 REM
DC.2.03.02.24 **PUHV**	.383 REM
DC.2.03.02.25 **PUL**	1.741 REM
DC.2.03.02.26 **PUSA**	.316 REM
DC.2.03.02.27 **PUTC**	.529 REM
DC.2.03.02.28 **PUWG**	.279 REM

Table 6	
Estimated Exposure/Area of Activity & Decommissioning Activities Scheduled	
Area/Activity	**Exposure**
(Bolded Acronyms are for affected areas of decommissioning activities)	
DC.2.03.04 **Service/fuel building/SVH/SPRB**	
DC.2.03.04.01 **DWST**	.103 REM
DC.2.03.04.02 **EFPR**	.159 REM
DC.2.03.04.03 **FBP**	
DC.2.03.04.04 **LSAB**	
DC.2.03.04.05 **NFLA**	.628 REM
DC.2.03.04.06 **RCAD**	1.622 REM
DC.2.03.04.07 **RCAW**	8.772 REM
DC.2.03.04.08 **RMCC**	.046 REM
DC.2.03.04.09 **SBDR**	.314 REM
DC.2.03.04.10 **SBHP**	.044 REM
DC.2.03.04.11 **SBMS**	.293 REM
DC.2.03.04.12 **SBP**	
DC.2.03.04.13 **SBSR**	.111 REM
DC.2.03.04.14 **SBTT**	
DC.2.03.04.15 **SBVH**	
DC.2.03.04.16 **SFP**	32.159 REM
DC.2.03.04.17 **SFPH**	9.120 REM
DC.2.03.04.18 **SFPV**	.287 REM
DC.2.03.04.19 **SPRB**	78.093 REM
DC.2.03.04.20 **SVH**	.054 REM
DC.2.03.05 Miscellaneous	
DC.2.03.05.01 **BWST**	.162 REM
DC.2.03.05.02 **CST**	.003 REM
DC.2.03.05.03 **CWI**	
DC.2.03.05.04 **FI**	
DC.2.03.05.05 **FOB**	
DC.2.03.05.06 **FPH**	
DC.2.03.05.07 **GH L**	
DC.2.03.05.08 **HRB**	.528 REM
DC.2.03.05.09 **PWST**	.068 REM
DC.2.03.05.10 **RWST/SCAT**	1.549 REM
DC.2.03.05.11 **STFB**	
DC.2.03.05.12 **STPI**	
DC.2.03.05.13 **West - RCA**	7.136 REM
Total Estimated exposure for the project	**937.543 REM**

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
Miscellaneous yard structures	
Z-0327 Remove hut - 8 side office complex	21JAN99
Period 1 -Preparation and Planning	
Site Characterization	
CG Fabricate and Deliver Containment Crane	10FEB99
2123 S&W Acceptance of Cranes	04MAR99
Period 2 - Decommissioning	
PLAD El 11' thru 21' Aerated Drain Tank	
0210STRPAB El 11' ADT- Source Term Reduction-Make Cuts	31MAR99
PLCP El 11' Aux Chrg Pump Cubicle	
0200STRPAB El 11' Aux Chrg Pump- Src Trm Red - Make Cuts	31MAR99
PLBA El 11' Boric Acid Mix Tank	
0206STRPAB El 11' BAMT- Source Term Reduction - Make Cuts	01APR99
PLPT El 11' Pipe Tunnel	
0205STRPAB El 11' Pipe Tun- Src Trm Red - Remove Pipe & HS	07APR99
PLLA El 11' Letdown Area	
0203STRPAB El 11' Letdown- Src Trm Red-Remove Pipe & HS	08APR99
P21A El 21' Valve Alley	
A011 PLBA - PAB El 11'TK-3 Remove Asbestos	13APR99
PLPA El 11' Ctmt Penetration Area	
0204STRPAB El 11' Ctmt Pen- Src Trm Red - Remove Pipe & HS	14APR99
PLEC El 11' Evap Cubicle	
0201STRPAB El 11' Evap Cub- Src Trm Red -	26APR99
PLPD El 11' Primary Drain Tank	
0208STRPAB El 11' PD Tnk- Src Trm Red - Unbolt & Make Cuts	27APR99
PLDC El 11' Degas Cubicle	
0202STRPAB El 11' Degas Cub- Src Trm Red - Make Cuts	27APR99
CB1-8 Outside Vault	
CB18ST Loop 1 Lwr Levl O/A - Source Term Reduction -Make Cuts	27APR99
CB19ST Loop 2/3 Lwr Levl O/A - Srce Trm Red -Make Cuts	28APR99
Period 1 -Preparation and Planning	
Site Characterization	
CCS18 Soil Samples for Part 61	03MAY99
Period 2 - Decommissioning	
P21D El 21' Degas Cubicle	
0195STRPAB 21' Degas - Source Term Reduction Make Cuts	03MAY99
CB1-1 SG#1 Vault	
CB11ST Loop 1 Source Term Reduction - Make Cuts	05MAY99
PLBA El 11' Boric Acid Mix Tank	
0206 PLBA -PAB 11' Boric Acid Mix Tank Commodities	06May99
CB1-3 SG#3 Vault	
CB13ST Loop 3 Source Term Reduction - Make Cuts	10MAY99
CB1-5 PZR & Quench Tk	
CB15ST CTMT Quench Tank Area - Source Term Red. Make Cuts	12MAY99

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
P21C El 21' Chrg Pump Cubicles	
0193STRPAB 21' CH Pump - Source Term Reduc. Make Cuts	13 MAY99
Period 1 -Preparation and Planning	
Site Characterization	
CCS23 Concrete Samples for Part 61 and Groundwater	13 MAY99
Period 2 - Decommissioning	
GH Gas House	
0025 Gas House Remove Commodities & Demolition (GHS)	17MAY99
PLCP El 11' Aux Chrg Pump Cubicle	
0200 PLCP -PAB 11' Aux Chrg Pump Cubicle Commodities	18May99
P21B El 21' BA Pump Area/Workshop	
0191STRPAB 21' BA Pump - Source Term Reduc. Make Cuts	19MAY99
P21S El 21' Sample Sink Area	
0189STRPAB 21' Sample Sink - Source Term Reduction Make Cuts	19MAY99
Miscellaneous Yard Structures	
0028 Remove North Interior PA Fence	20MAY99
PLAD El 11' thru 21' Aerated Drain Tank	
0199 PLAD - Phases I (Pumps & Small Bore - 349 hrs)	24MAY99
PUTC El 36' VCT Cubicle	
0213STRPAB El 36' VCT Cubicle- Src Trm Red - Make Cuts	03JUN99
Demolition of Remaining Site Buildings	
A020 Eaton Farm House/Barn - Remove Asbestos	14JUN99
SPRB Spray Buildiing	
SPRBST Spray Bldg - Source Term Reduction - Make Cuts	17JUN99
ABR Aux Boiler Room	
A001 ABS -Piping & Tank by Aux Boilers - Asbestos Removal	17JUN99
FWH Turbine Feedwater Heaters	
A002 Turbine Building -FWH Asbestos Removal	21JUN99
P21A El 21' Valve Alley	
A012 P21A-PAB El 21' Valve Hall- Remove Asbestos	22JUN99
Period 1 -Preparation and Planning	
Site Characterization	
0760A RP Perform Soil Sample Acquisition for Characterization	29JUN99
2411T0 Perform Supplemental Site Characterization	29JUN99
0769 RP Perform Characterization of Structures and Systems	30JUN99
0699 Complete Site Characterization & Bounding	01JUL99*
Period 2 - Decommissioning	
HRB High Radiation Bunker	
0514 HRB-High Radiation Bunker Commodities	07JUL99
RWST/SCAT Refueling H2O Storage (TK-4)	
0023 RWST/SCAT - Commodities	07JUL99
HRB High Radiation Bunker	
0221 HRB - High Rad Bunker - Decontamination	12JUL99

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
Period 2 - Decommissioning	
CB1-1SG#1 Vault	
0071 RCP#1 Pipe Cuts	12JUL99
Period 1 - Preparation and Planning	
Site Characterization	
0770 RP Develop/Submit Soils and Structures Report	15JUL99
RCAW RCA West	
0022 RCA West (ILRT, etc) Commodities	15JUL99
CB1-2 SG#2 Vault	
CB12ST Loop 2 Source Term Reduction - Make Cuts	16JUL99
Period 1 - Preparation and Planning	
Site Characterization	
0776 RP Submit Characterization Survey Report to S&W	19JUL99
TMDV El 39' Main Steam Dump Valve Area	
A003 Turbine Hall 39' level - Asbestos Removal (3Areas)	19JUL99
CB1-8 El -2' thru 20' Outside Vault	
A007 CB1-8 El -2' thru 20' Remove Asbestos	21JUL99
Period 2 - Decommissioning	
CB1-1 SG#1 Vault	
0279 SG#1 Cut at Hot Leg	28JUL99
0278 SG#1 Cut Hot Leg Spool	29JUL99
CB2-1 El 20' thru 46'	
A008 CB2-1 El 20' thru 46' Remove Asbestos	29JUL99
CB1-1 SG#1 Vault	
0281 RCP#1 Cut Suction Leg	03AUG99
Miscellaneous Clean Structures	
A021 Bailey Farm House/Barn- Remove Asbestos (2 Areas)	12AUG99
Demolition of Remaining Site Buildings	
Z-0467 PFB - Pre -Fab Bldgs. Demo (Gen, Maint Yd, Stl Bx)	12AUG99
CB1-3 SG#3 Vault	
0071C1 RCP#3 Pipe Cuts	18AUG99
Demolition of Remaining Site Buildings	
Z-0479 Eaton Farm House/Barn Demo & Site Restoration	19AUG99
CB3-3 El46' Outside Vault Areas	
A009 CB3-3 El46' - Remove Asbestos	23AUG99
PLLA El 11' Letdown Area	
0203 PL-- -PAB 11' ---- Letdown Area Commodities	24AUG99
Reactor Internals & Removal Prerequisites	
0064 Remove Upender & Carriage	25AUG99
CB1-3 SG#3 Vault	
0278C1 G#3 Cut Hot Leg Spool	26AUG99
THEA El 21' SCC/PCC HX Area	
Z-0123 THEA- TB El 36' SCC/PCC HX Area Commodities	31AUG99
CPHO Personnel Hatch Outer(PE-4)	
A026 CPHO- Personnel Hatch - Remove Asbestos	31AUG99

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY		COMPLETION DATE
CB1-3 SG#3 Vault		
0279C1	SG#3 Cut at Hot Leg	01SEP99
Demolition of Remaining Site Buildings		
Z-0407	RRT- Railroad Cars Demolition	02SEP99
CB1-3 SG#3 Vault		
0281C1	RCP#3 Cut Suction Leg	08SEP99
Administration Bldg. (Front Office) and Additions		
A027	Admin/Sec Bldg.- Remove Asbestos	09SEP99
Sewage Treatment Plant Pit		
0714	Sewage Treatment Plant Pit	13SEP99
MCR Main Control Room		
A022	Old Computer Room - Remove Asbestos (2 Area s)	16SEP99
STP1 Sewage Treatment Plant		
0225	STP1 - Sewage Treatment Plant Commodities	14SEP99
Service/Fuel/Building/SVH/SPRB		
A023	Old Control Room - Remove Asbestos (3 Areas)	14SEP99
DWST Demin H2O Storage Tank		
0024	DWST Demin Water Storage Tank Commodities	15SEP99
DWST Demin H2O Storage Tank		
0024	DWST Demin Water Storage Tank Commodities	15SEP99
Tank Foundations - Yard		
0027	CST (Condensate Storage Tk) Foundation Demolition	15SEP99
DWST Demin H2O Storage Tank		
0024	DWST Demin Water Storage Tank Commodities	15SEP99
CB1-5 PZR & Quench Tk		
0048X	Cut Pipe from Pressurizer	20SEP99
CB1-1 SG#1 Vault		
0676	Remove CEA Change Racks	23SEP99
CB1-2 SG#2 Vault		
0071B1	RCP#2 Pipe Cuts	23SEP99
Miscellaneous Yard Structures		
0242	LLRW Bldg. Commodity Removal	23SEP99
CB1-3 SG#3 Vault		
0530X	S/G#3 Cut FW&MS Lines	28SEP99
TSRP El21' Switchgear Room Protected		
A030	Protected Switch Gear Room - Remove Asbestos	28SEP99
0082X	SIT #3 Cut Piping	30SEP99
Transformer Foundations		
0245	Remove/Rig out Main Transformers (X1A, X1B, XS)	30SEP99
SVH Steam Valve House		
A016	SVH - Valve House- Remove Asbestos (3 Areas)	30SEP99
Transformer Foundations		
0735	X-24 Transformer Demolition	04OCT99
0736	X-26 Transformer Demolition	05OCT99
Administration Bldg. (Front Office) and Additions		
A031	Gate House - Remove Asbestos	06OCT99

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY			COMPLETION DATE
CB1-1 SG#1 Vault			
	0278B1	SG#2 Cut Hot Leg Spool	06OCT99
Transformer Foundations			
	0737	X-28 Transformer Demolition	11OCT99
CB1-1 SG#1 Vault			
	0279B1	SG#2 Cut at Hot Leg	12OCT99
PRIMARY AUXILIARY BLDG.			
	0210	Cut concrete opening -PLAD	13OCT99
CB1-1 SG#1 Vault			
	0789	Rx Coolant Pumps and Motors(HA)	14OCT99
TSPP	El 21' SCC/PCC Pump Area		
	Z-0118	TSPP- TB El 21' SCC/PCC Pump Area Commodities	18OCT99
CB1-1 SG#1 Vault			
	0281B1	RCP#2 Cut Suction Leg	18OCT99
Fuel Oil Bunker			
	0730	Fuel Oil Bunker Demolition	20OCT99
Wart Building			
	A032	Wart Bldg. (HP Checkpoint - Remove Asbestos	20OCT99
SBTT Service Bldg. Test Tank Area			
	0026	SBTT Service Building Tests Tank Area Commodities	21OCT99
SBTT Service Bldg. Test Tank Area			
	0026	SBTT Service Building Tests Tank Area Commodities	21OCT99
SBVH Service Bldg. Steam & Valve House Alley			
	Z-0055	SBVH Service Bldg. Stm& Vlv House Alley commodities	21OCT99
Turbine Building			
	A034	Turbine Bldg. (Elevator Shaft) - Remove Asbestos	24NOV99
RMCC Reactor MCC Room			
	0127	Remove Commodities (Rx MCC Room)	27OCT99
CB1-1 SG#1 Vault			
	0520X	S/G#2 Cut FW&MS Lines	04NOV99
CB1-1 SG#1 Vault			
	0157X	S/G#1 Cut FW&MS Lines	04NOV99
CB1-3 SG#3 Vault			
	0083X	SIT #3 Cut-Up	04NOV99
Containment			
	A033	CTMT Bldg. (Elevator Shaft) - Remove Asbestos	08NOV99
CB1-1 SG#1 Vault			
	0065X	SIT #1 Cut Piping	09NOV99
PLPT El 11' Pipe Tunnel			
	0205	PLPT -PAB 11' Pipe Tunnel Commodities	10NOV99
Miscellaneous Yard Structures			
	2035	Remove Guard Towers	16NOV99
CB1-4 Regen HX Vault			
	0143X	SIT #2 Cut Piping	18NOV99
CEHO Equipment Hatch Outer			
	0139	Remove Commodities	29NOV99

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY		COMPLETION DATE
PRIMARY AUXILIARY BLDG.		
0210	PLAD PAB 11' ADT - Phase 2 Remove Commodities	30NOV99
PLDC El 11' Degas Cubicle (continued)		
0202	PLDC -PAB 11' Degas Cubicle Commodities	30NOV99
Tank Foundations - Yard		
0726	DWST Demolition	01DEC99
CB1-7 Iodine Filter Area		
0069	Iodine Filter Area - Remove Commodities	02DEC99
Containment Decontamination		
0509	CB1-7 Rad Decon, CTMT Iodine Filter Area	09DEC99
SVH Steam Valve House		
0133	SVH - Remove Commodities	13DEC99
CB1-1 SG#1 Vault		
0066X	SIT #1 Cut-Up	15DEC99
PRIMARY AUXILIARY BLDG.		
0171	Rework Roof Interface - SFPI HVAC	22DEC99
CB1-4 Regen HX Vault		
0141X	SIT #2: Regen HX Commodity Removal	22DEC99
0144X	SIT #2 Cut-Up	22DEC99
TBWT El 21' Water Treatment		
Z-0108	TBWT- TB El 21' Water Treatment Commodities	28DEC99
Tank Foundations - Yard		
0727	RWST Demolition	29DEC99
PLEC El 11' Evap Cubicle		
0201	PLEC -PAB 11' Evap Cubicle Commodities	03JAN00
PAB Decontamination		
PLBA03	PLBA- Rad Decon PAB El-11' Boric Acid Mix Tank	10JAN00
PU48 El 36' FN-48 Area		
0184	PU48 - PAB 36' Fan FN-48 Area Commodities	10JAN00
Service Building		
0713	Service Bldg. Test Tank (SBTT) Demolition	12JAN00
SVH Decontamination		
0134	Steam Valve House Decontamination	17JAN00
PAB Decontamination		
PLCP03A	PLCP- Rad Decon PAB El-11' Aux Chrg Pmp Cubicle	17JAN00
PU48 El 36' FN-48 Area		
0183	Cut Concrete Opening - PU48 (Roof)	13JAN00
SVH Steam Valve House		
0132	SVH - Steam Valve House (HA)	17JAN00
Turbine Building Decontamination		
Z-0126	Turbine Bldg. Decontamination	18JAN00

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY		COMPLETION DATE
PAB Decontamination		
PLAD03	PLAD- Rad Decon PAB El-11' ADT - Phase 2	24JAN00
High Radiation Bunker		
0222	HRB- High Radiation Bunker Demolition	24JAN00
PAB Decontamination		
PLLA03APLLA-	Rad Decon PAB El-11' ADT - Letdown Area	31JAN00
PLPA El 11' Ctmt Penetration Area		
0204	PLPA -PAB 11' Ctmt Pen. Area Commodities	07FEB00
P21H El 21' Letdown HX Room		
0192	P21H - PAB 21' Ltdn HX Room Commodities	07FEB00
Miscellaneous Yard Structures		
0733	RCA West Demolition	09FEB00
PAB Decontamination		
PLPA03APLPA-	Rad Decon PAB El-11' CTMT Penetration Area	14FEB00
PAB Decontamination		
P21H03A	P21H- Rad Decon PAB El-21' Letdown HX Room	16FEB00
PAB Decontamination		
PLPT03APLPT-	Rad Decon PAB El-11' Pipe Tunnel	21FEB00
CB1-5 PZR & Quench Tk		
0049X	Rig out Pressurizer	23FEB00
P21V HPSI/LPSI Valve Rm		
0188	P21V - PAB 21' HPSI/LPSI Room Commodities	01MAR00
CB1-5 PZR & Quench Tk		
0051Z	Cut up Quench Tank	02MAR00
Reactor MCC Room		
Z-0157	Reactor MCC Room Demolition	06MAR00
PAB Decontamination		
0212	PAB Decontamination - Roof	09MAR00
P21E El 21' Evap Cubicle		
0194	P21E - PAB 21' Evap Cubicle Commodities	13MAR00
PLPD El 11' Primary Drain Tank		
0209	PLPD -PAB 11' PD Tnk Commodities	20MAR00
PAB Decontamination		
P21E03A	P21E - Rad Decon PAB El-21' Evap Cubicle	23MAR00
PAB Decontamination		
PLPD03a	PLPD - Rad Decon PAB El 11' Primary Drain Tank	27MAR00
CB1-1 SG#1 Vault		
0576	SG#1 Commodity Removal (HA)	28MAR00
PLPW El 11' Primary H2O Pump Area		
0198	PLPW-PAB 11' Primary Water Pump Area Commodities	28 MAR00
Steam Valve House		
Z-0156	Steam Valve House Demolition	29MAR00
PAB Decontamination		
PLPW03	PLPW - Rad Decon PAB El 11' Primary H20 Pump Area	04APR00
PAB Decontamination		
PLDC03	PLDC - Rad Decon PAB El 11' Degas Cubicle	11APR00

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
PAB Decontamination	
PLEC03APLEC - Rad Decon PAB El 11' Evap Cubicle	18APR00
P21D El 21' Degas Cubicle	
0196 P21D - PAB 21' Degas Cubicle Commodities	18APR00
0195 Cut Concrete Opening - P21D	24APR00
CB3-3 El46' Outside Vault Areas	
0118 CB3-3 Remove Commodities	25APR00
Reactor Internals & Removal Prerequisites	
0527 Upper Guide Support Stand Removal	26 APR00
PAB Decontamination	
P21D03AP21D - Rad Decon PAB El 21'Degas Cubicle	27APR00
CICI Incore Instrument Sump	
0053 Remove Commodities	27APR00
Circ Water Pumphouse	
0732 Circ Water Pump House Demo (Incl TK-33)	03MAY00
Containment Decontamination	
0542 CICI - Rad Decon, CTMT Incore Instrument Sump	16MAY00
Demolition of Remaining Site Buildings	
Z–0347 WHSE-Demolition Warehouses #5	18MAY00
CB1-3 SG#3 Vault	
0562 SG#3 Commodity Removal (HA)	18MAY00
P21L El 21' (General)	
0187 P21L - PAB 21' General Commodities	24MAY00
PAB Decontamination	
P21C03AP21C - Rad Decon PAB El 21'Chrg Pump	31MAY00
CB1-2 SG#2 Vault	
0564 SG#2 Commodity Removal (HA)	05JUN00
P21B El 21' BA Pump Area/Workshop	
0191 PAB 21' Boric Acid Pump & Workshop	05JUN00
P21C El 21' Chrg Pump Cubicles	
0193 PAB 21' CH Pump Cubicles Commodities	05JUN00
P21A El 21' Valve Alley	
0186 P21A PAB 21' Valve Hall Commodities	13JUN00
PAB Decontamination	
P21B03AP21B - Rad Decon PAB El 21' Boric Acid Pump	14JUN00
P21S El 21' Sample Sink Area	
0189 Cut Concrete Opening - P21S Sample Sink Wall 21'	19JUN00
PUEC El 36' Decay Drum Cubicle	
0178 PUEC - PAB 36' Upper Lvl Evap Cubicle Commodities	27JUN00
CB1-1 SG#1 Vault	
0351X SG#1 Load on Barge #1	03JUL00
PUFN El 36' FN-1A/B Area	
0179 Cut Concrete Opening - PUFN (Roof)	03JUL00
PUEC El 36' Decay Drum Cubicle	
0181 Cut Concrete Opening - PUEC (Roof)	03JUL00

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY		COMPLETION DATE
Containment Decontamination		
0507	CB1-5 Rad Decon, CTMT	03JUL00
P21S El 21' Sample Sink Area		
0190	P21D - PAB 21' Commodities	05JUL00
PAB Decontamination		
P21L03A	P21L - Rad Decon PAB El 21' General	06JUN00
Repowering		
0790	PZR (HA)	17JUL00
CB1-5 PZR & Quench Tk		
0375x	Pressurizer Load on Barge #1	17JUL00
PAB Decontamination		
P21V03AP21V	Rad Decon PAB El 21' HPSI/LPSI Room	17JUL00
PUWG El 36' Waste Gas Cubicle		
0175	PUWG - PAB El 36' Waste Gas Cubicle Commodities	18JUL00
Reactor Internals & Removal Prerequisites		
0103	Remove Vessel Head	20JUL00
0125	CEDM Removal(Special LSA Boxes Req'd)	27JUL00
PUSA El 36' Radioactive Storage Area		
0173	PUSA - PAB 36' Radioactive Waste Storage Area	31JUL00
PUSA El 36' Radioactive Storage Area		
0173	PUSA - PAB 36' Radioactive Waste Storage Area	31JUL00
PUHV El 36' HVAC Room		
0176	PUHV - PAB 36' HVAC Room Commodities	16AUG00
Repowering		
0791	Sgs (HA)	16AUG00
CB1-2 SG#2 Vault		
0358X	SG#2 Load on Barge #2	16AUG00
CB1-3 SG#3 Vault		
0368X	SG#3 Load on Barge #2	16AUG00
Reactor Internals & Removal Prerequisites		
1503	ICI Removal	21AUG00
PLPD El 11' Primary Drain Tank		
0208	Cut Concrete Opening -PLPD	22AUG00
Reactor Internals & Removal Prerequisites		
0140	Asbestos Abatement Reactor Vessel Head	06SEP00
PUFN El 36' FN-1A/B Area		
0180	PUFN - PAB 36' Fan FN-1A/B Area Commodities	12SEP00
Reactor Internals & Removal Prerequisites		
0505	Segment RV Head	14SEP00
Repowering		
2435	Decision to Repower	18SEP00
PUL El 36' (General), BAST, BDT		
0174	PUL - PAB 36' General , BAST & BDT Commodities	04OCT00
PUDD El 36' Decay Drum Cubicle		
0177	Cut Concrete Opening - PUDD (Block Wall)	10OCT00
Reactor Internals & Removal Prerequisites		
0107	Vessel Head Packaging and Ship	30OCT00

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
PUDD El 36' Decay Drum Cubicle	
0178 PUDD - PAB 36' Decay Drum Cubicle Commodities	02NOV00
PUTC El 36' VCT Cubicle	
0213 Cut Concrete Opening - PAB Upper Lvl VCT Cub(Roof)	08NOV00
PUTC El 36' VCT Cubicle	
0214 PUTC -PAB 36' VCT- Cubicle Commodities	21NOV00
Reactor Vessel Internals	
0106 Rx Vessel Internals Segmentation & =Packaging (FTI)	04DEC00
Reactor Vessel Internals	
0745 Store GTCC Material	19DEC00
0746 Ship Class C Material to Barnwell	19DEC00
CB2-1 El 20' thru 46'	
0121 Annulus El 20' thru 46'- Remove Commodities	27DEC00
PAB Decontamination	
PUEC03 PUEC - Rad Decon PAB Upper Level Evap Cubicle	28DEC00
Turbine Building	
0247 Turbine building Demolition	16JAN01
Reactor Spray Building	
0130 Remove Commodities Spray Building	25JAN01
SWEC Loads RPV/Prep Container	
0512 Cut off RV Nozzles	27JAN01
SWEC Loads RPV/Prep Container	
CN5200 SWEC Grout Vessel Cavity	27JAN01
PAB Decontamination	
PUDD03 PUDD - Rad Decon PAB El 36' Decay Drum Cubicle	05FEB01
Containment Decontamination	
0521 CB2-1 Rad Decon, CTMT Personnel Hatch Inner	06FEB01
SWEC Loads RPV/Prep Container	
CN5100 Lift RPV form the cavity and place inside Cask	17FEB01
SWEC Loads RPV/Prep Container	
CN5300 SWEC Grout Annular Space	21FEB01
CPHO Personnel Hatch Outer(PE-4)	
0136 Remove Commodities	27FEB01
SPRB Spray Building	
0129 SPRB - Spray Building (HA)	01MAR01
SWEC Loads RPV/Prep Container	
CN5000 SWEC Loads RPV/Prep Container (HA)	02MAR01
Containment Decontamination	
0544 C PHO-Rad Decon, CTMT Personnel Hatch Outer	06MAR01
RPV Mobilize, Transport and Dispose	
CN6030ZLoad RPV and Transport to Barge One	26MAR01
CN6040ZBarge Transport of RPV to SRS	27MAR01/ 08APR01
CN6000 RPV Mobilize, Transport and Dispose	16APR01
Containment Decontamination	
0394 CB1-1 Rad Decon, Ctmt Chrg Flr. S/G1 Sec Man way	17APR01

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
Spray Building	
0569 Spray Building Demolition	13MAY01
Primary Aux Building and Pipe Tunnel	
Z-0157 PAB - Primary Aux Building Demolition- Phase I	14MAY01
Miscellaneous Yard Structures	
0731X Demo Gas House Foundation	16MAY01
Demolition of Remaining Site Buildings	
Z0247 ABS- Aux Boiler Stack Demolition	16MAY01
Containment Decontamination	
0398 CB1-3 Rad Decon, Ctmt Chrg Flr S/G3 Sec Man way	17MAY01
Containment Decontamination	
0397 CB1-2 Rad Decon, Ctmt Chrg Flr S/G2 Sec Man way	31MAY01
Reactor Vessel Internals	
0114 Segment Neutron Shield Tank	07JUN01
Service Building	
Z-0152X Service Building Demolition (Phase 1)	11JUN01
Period 2 - Decommissioning	
0115 Package & Ship Neutron Shield Tank	06SEP01
Reactor Internals & Removal Prerequisites	
0042 Remove RX missile Shield (CRD Shield)	06SEP01
CB3-1 Cavity Area	
0147 CB3-1 Remove Commodities/Cut Liners	25SEP01
Containment Decontamination	
0535 CB3-1 Rad Decon, CTMT Cavity Head Area	02OCT01
PAB Decontamination	
0154 Decontamination (HA)	09OCT01
Containment Decontamination	
0508 CB1-6 Rad Decon, CTMT-2 Lvl Sump Pump Area	28NOV01
CB1-8 El -2' thru 20' Outside Vault	
0124 Annulus, Elev. -2 thru 20' Remove Commodities	31OCT01
Containment Decontamination	
0511 CB1-8 Rad Decon, CTMT-2 Lvl Outside Loop 1	13DEC01
CB3-2 Refueling Area	
0150 CB3-2 Remove Commodities/Cut Liners	17DEC01
Containment Decontamination	
0537 CB3-2 Rad Decon, CTMT Cavity Upender Pit	24DEC01
CCG El 46' (General)	
0152/153 Remove Commodities (Charging Floor)	22JAN02
Containment Decontamination	
0539 CCG Rad Decon, CTMT Chrg Floor (General)	28JAN02
CPLE- Ctmt Pzr Elevator Rm	
0156 Remove Commodities	29JAN02
CB3-4 Polar Crane (CR-1)	
0159 CB3-4 Remove Commodities	21MAR02
Containment Decontamination	
0538 CB3-4 Rad Decon, CTMT Polar Crane (CR-1)	17APR02

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
Demolition of Remaining Site Buildings	
Z-0347X WHSE- Warehouses #4 (Annex) Demolition	25APR02
Demolition of Remaining Site Buildings	
Z-0347X1WHSE- Warehouses #2/3 Demolition	11JUN02
Miscellaneous Bldgs Decontamination	
0232 PWST Decontamination	15AUG02
Fuel and RCA Building	
A036 Fuel Bldg.-Remove Asbestos Above Spent Fuel Pool	21AUG02
0257 RCA/LSA/Fuel Building Demolition	25AUG03
0230 XFMRS (X-14, X16, Z1) Comm Removal	28AUG02
Equipment Hatch	
0711 Containment Equipment Hatch Demolition	09SEP02
Staff Building and Tunnel	
Z-0069 STFB-Staff Building Demolition	16SEP02
YARD TRANSFORMERS (X-1A&B, X-14 & X-16)	
0231 SFPI XFMR/Pagado Comm Removal	17SEP02
EFPR Emergency Feedpump Room	
Z-0057 EFPR Emergency Feedpump Room Commodities	18SEP02
Miscellaneous Clean Structures	
0719 Barge Slip/Road Demolition	23SEP02
Miscellaneous Bldgs Decontamination	
0234 BWST/Berm Decontamination	23SEP02
SBP Service Building Proper	
Z-0141 SBP- Service Bldg. Proper Commodities	24SEP02
SBSR Service Bldg. El 21' Seal Rebuild Room	
Z-0058 SBSR - Service Bldg. El 21' Seal Rebuild Room	24SEP02
Miscellaneous Clean Structures	
0725 Emergency Feed Pump Room Demolition	30SEP02
0724 Sanitary Lines Demolition	30SEP02
PWST Primary H20 Storage (Tk-16)	
0513 PWST - Primary Water Storage Tank Commodities(HA)	01OCT02
Tank Foundations - Yard	
0233 PWST Demolition	01OCT02
SBDR Service Bldg. El 21' Decon Room	
Z-0059 SBDR - Service Bldg. El 21' Decon Room	02OCT02
SBHP Service Bldg. El 21' HP checkpoint	
Z-0139 SBHP - Service Bldg. El 21' Checkpoint Commodities	07OCT02
SFPH Spent Fuel Pool HX	
Z-0131 SFPH- Spent Fuel Pool Heat Exchanger Commodities	09OCT02
Personnel Hatch	
0712 Containment Personnel Hatch Demo	20OCT02
SFPV Spent Fuel Pool Ventilation Area	
Z-0137 SFPV- Spent Fuel Pool Ventilation Area Commodities	22OCT02
Demolition of Remaining Site Buildings	
Z-0307 ULP-Utilities Light Poles Demolition	23OCT02
Fuel Loading & Transfer to ISFSI	
0551 Decommissioning of SFP,&SFPI SSFSI	27OCT02

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
Demolition of Remaining Site Buildings	
Z-0317 PW-Potable Water Connection Demolition	31OCT02
SFP Spent Fuel Pool	
Z-0133 SFP - Spent Fuel Pool Commodities	04NOV02
BWST Boron Waste Storage (TK - 13A&B)	
0515 BWST - Boron Waste Storage Tank Commodities	05NOV02
Tank Foundations - Yard	
0235 BWST/Berm Demolition	05NOV02
CW1 Circ Water Pumphouse	
0226 CW1 - Circ Water Pump House Commodities	05NOV02
Demolition of Remaining Site Buildings	
Z-0357 WWH-Well Water House Demolition	05NOV02
Service/Fuel/Building/SVH/SPRB	
0254 Cut Liner/Fuel Racks - Ship	11NOV02
FBP Fuel Building Proper	
Z-0135 FBP- Fuel Bldg. Proper Commodities	14NOV02
Wart Building	
0728 Wart Bldg. / F&C Demolition	19NOV02
RCAW RCA Storage Bldg. El 21' Waste Solid	
Z-0140 RCAW RCA Stor Bldg. El21' Waste Solidification	11DEC02
YARD TRANSFORMERS (X-1A&B, X-14 & X-16)	
0219 Yard Transformers (HA)	11DEC02
0236 115KV Yard Dismantlement	11DEC02
Soil Removal	
0237 Soil Removal	11DEC02
SBMS Service Bldg. El 21' Machine Shop (HOT)	
Z-0138 SBMS - Service Bldg El 21' Machine Shop (Hot)	19DEC02
115KV Switchyard Foundations	
0716 115KV Transformer Yard Demolition	26DEC02
Service/Fuel/Building/SVH/SPRB	
0253 Remove Commodities - Phase	07JAN 03
NFLA New Fuel Laydown Area	
0252 Remove Commodities- Phase 1 (NFLA)	07JAN03
RCAD RCA Drumming Room	
Z-0132 RCAD - RCA Drumming Room	07JAN03
Containment Decontamination	
0163 Containment Gross Decontamination - All Areas	14JAN03
Demolition of Remaining Site Buildings	
Z-0207 OFF GAS-Off gas Stack Demolition	22JAN03
Administration Bldg. (Front Office) and Additions	
Z-0197 ADMN-Admin/Sec Bldg. Incl Gatehouse Demolition	12FEB03
Transformer Foundations	
0738 X-14, 16, 16 A&B Transformer Demolition	13FEB03
Demolition of Remaining Site Buildings	
Z-0257 TEST-Test Pit Demolition	17FEB03
Z-0417 TPS-Temp Power Shack Demolition	18FEB03
Z-0227 GPT-Temp Generator/Pagoda/Transformer	19FEB03

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY	COMPLETION DATE
Demolition of Remaining Site Buildings (Continued)	
Z-0447 Demolition (Fences, meter house, conc pads)	20FEB03
Miscellaneous Clean Structures	
0720 Electric House Demolition	25FEB03
Demolition of Remaining Site Buildings	
Z-0437 SEAL-Seal Pit Outfall Demolition	11MAR03
Miscellaneous Clean Structures	
Z-0482 Bailey Farm House/Barn Demo. & Site Restoration	18MAR03
Fuel and RCA Bldg. Decontamination	
Z-0136 Fuel and RCA Bldg. Decontamination	31MAR03
ABR Aux Boiler Room	
A037 ABS - Aux Boilers - Asbestos Removal	03APR03
Service Building Decontamination	
Z-0142 Service Building Decontamination	05MAY03
Service Building	
Z-0152 Service Building Demolition (Phase 2)	27MAY03
Demolition of Remaining Site Buildings	
Z-0237 MOD-Modular Offices (2) Demolition	29MAY03
FBI Fox Bird Island	
0227 FI - Foxbird Island/Seal Pit Demolition	16JUN03
Miscellaneous Clean Structures	
0722 Remove Outside Utilities (3ft. Below) Demo.	17JUN03
Fuel Loading 7 Transfer to ISFSI	
0571 Demo Remaining Sec PAB, CTMT, Fuel Bldg. Phase II	25AUG03
Service/Fuel/Building/SVH/SPRB	
0250 Fuel Building/RCA Commodities (HA)	25AUG03
Containment Demolition	
Z-0148 Containment Building Demolition	25AUG03
Diffuser Area	
0229 Engineering Evaluation of Diffuser Piping	03SEP03
Demolition of Remaining Site Buildings	
Z-0367 INFO- Info. Center Bldg. Demolition	09SEP03
Miscellaneous Bldg Decontamination	
0661 Misc Building Decontamination (HA)	11SEP03
Miscellaneous Bldg Decontamination	
0238 LSA Bldg. Decontamination	11SEP03
Demolition of Remaining Site Buildings	
Z-0377 COLL-Collection Site Demolition	15SEP03
Z-0387 LIFT- Lift Station Demolition	17SEP03
Diffuser Area	
0228 Diffuser Piping Demolition (If Required)	22SEP03
LSA Storage Building	
0239 LSA Bldg. Demolition	25SEP03
Demolition of Remaining Site Buildings	
Z-0397 ET-Elect. Towers Demolition	25SEP03
Z-0267 SECF- Security Fence Demolition	09OCT03
Z-0407RTRRT-Railroad Tracks Demolition	16OCT03

TABLE 6 (Continued)

Area of Activity & Decommissioning Activities Scheduled (Arranged Chronologically)

ACTIVITY		COMPLETION DATE	
Demolition of Remaining Site Buildings (continued)			
Z-0297	FPHS-Fire Protection Hose/STA/	12NOV03	
FPH Fire Pumphouse			
0227	FPH - Fire Pump House Commodities		08DEC03
Demolition of Remaining Site Buildings			
Z-0337	MPH-Montsweag Pump House Demolition	10DEC03	
System Removal			
2570	Dismantlement of all Site Area (HA)	13MAY99/ 08DEC03	
Fire Pump House & Pond			
0717	Fire Pump House & Pond Demolition		30DEC03
Decommissioning Project Completion			
0591	Final Doc. Submittal to Maine Yankee		29APR04
0287	Decommissioning Complete/Part 50 Lic. Terminated	**28SEP04**	
NOTE:	If the ISFSI is licensed under Maine Yankee Atomic Power Company's Part 50, that license will not be terminated until the Department of Energy (DOE) takes possession and removes the stored material (projected date 2026). After removal of all the stored material the Independent Spent Fuel Storage Installation (ISFSI) will be dismantled, decommissioned, remediated and final survey performed for unrestricted release.		
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ISFSI Dismantlement, Decommissioning and Remediation (After removal of all Stored Materials)			
AAA	Dismantlement of structures, support buildings, fences, lighting and utilities poles		2026*
BBB	Site Remediation, planting of grass, trees, etc as required.		2026*
BBB	Final Facility Site Survey (There should be no area due to the design of the ISFSI components and system)	2026*	
CCC	Release of the Facility Site for unrestricted use		2026*
DDD	Termination of Maine Yankee Atomic Power Company's Part 50 License		2026*

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

Note in the projected Year 2026, after DOE has taken possession and removed the stored materials

Drawing #1 presents a time line of Activities scheduled the preparation , removal, shipment and disposal of the Reactor Vessel from 1999 through 2001

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Drawing #2 presents a time line of Activities scheduled for the Containment Building in support of Major Components removal from 1999 through 2000.

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Drawing #3 presents a time line of Activities scheduled for the Primary Auxiliary Building in support of Site Decommissioning Activities.

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3.8 Remaining Decommissioning Activities (continued)

3.8.1 Decontamination of Structures

a. In-situ Concrete Decontamination by Bulk Removal

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

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

b. In-situ Surface Decontamination of Concrete

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

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

Perform decontamination to the extent required to achieve the unrestricted use criteria for demolished concrete. This rubblized concrete material will be deposited as fill within the lower elevations of the buildings, or used as fill on other parts of the site. Therefore, for concrete walls, floors, etc. that are scheduled for demolition, a decontamination of the surface will be performed to meet the unrestricted use criteria. Concrete decontamination will be performed to the depths required to remove the surface contamination to these levels using the most cost-effective technique, i.e., scabbling, high pressure water robotics, etc. In evaluating the best approach to the decontamination of larger concrete surfaces, we have determined that more aggressive technology is needed. To that end, the DOC is pursuing the development of remote operated scabblers attached to small excavator

machines. This removes the worker from the immediate area, thus reducing potential radiological exposure, increasing personnel safety, and what we estimate will be a significant increase in productivity.

d. Concrete Surfaces Located Greater than Three Feet below Grade

These surfaces will be decontaminated to the unrestricted use criteria established for demolished concrete, left in place, and filled concrete debris. Remediation of areas will occur, as deemed necessary to meet the unrestricted use criteria for demolished concrete.

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

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

Internal building steel within the RCA will be dismantled, packaged, and shipped for processing, unless it can be easily determined that the steel can be operationally free-released and therefore released to our demolition contractor.

Steel located within non-RCA buildings, i.e., not considered to have been exposed to radiological contamination, will be surveyed and released for demolition, i.e., Turbine Bldg., Circ Water Pump Structure, etc. The external structural steel of plant buildings has been assessed during our walkdowns and, depending upon the area, will either be surveyed and released for demolition or dismantled for packaging and shipment to a waste process (this case is very rare).

f. Embedded and Buried Piping Survey and Decontamination

Embedded piping is piping that is encased in concrete, whereas buried piping is as stated buried in a trench and surrounded by soil.

The radiological pipe crawler allows in-situ survey, characterization, and decontamination of underground and embedded piping versus demolition. By

using this technology safety risks, demolition costs, and secondary waste are reduced. (This technology has been used by other licensees to unconditionally release over 18,000 linear feet of piping with verification by the NRC). Based on survey results a decision considering the best engineering practice, will determine whether the remaining buried or embedded piping will be; left as is, capped, or grouted. The majority of embedded piping is not expected to be present at MY after building demolition.

3.8.2 Building Demolition and Site Restoration

Plant structures and facilities will be demolished to a level 3 ft below grade. The concrete will be reduced in size to the extent required to facilitate use as fill.

Table 7
Approach to Handling of Building Materials for Regulatory Release

No.	Type of building material	Approach
1	Areas with low contamination potential	Operationally free-release
2	Concrete with medium to high surface contamination potential	Remediate to unrestricted use criteria levels fill or disposal at Envirocare
3	metals removed	Ship to processor or for disposal at Envirocare or Barnwell
	Non-contaminated metals removed	Ship to processor for scrap or disposal
4	Built-up tar roofing, inner layer of siding (with actual or potential contamination)	Process at Hake or directly dispose at Envirocare
	"Clean" tar roofing, siding	Ship to a processor or disposal
5	Outer layer of siding (Galbestos)	Surface release survey; sent to asbestos landfill
6	Refueling cavity and spent fuel pool liners	Process at Hake

Demolition of property structures and facilities to a level of 3 feet below present grade, with the exception of the Eaton Farm House and Barn, the Bailey Farm House and Barn, the Ball Field, and the Montsweag Pump House.

Table 8 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
Containment building	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Steam and valve house	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Spray building	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Containment equipment hatch outer	1	Demo.
Containment personal hatch outer	1	Demo.
LSA building	1	Demo. backfill
Reactor MCC room	1	Demo. backfill
Emergency feed pump room	1	Demo. backfill
Primary auxiliary building	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Fuel building	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
RCA building	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Service building - hot side	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Service building test tanks (TK-14 A/B)	1	Complete demo.; remove pads; backfill
Demin. water storage tank (TK-21)	1	Complete demo.; remove pads; backfill
Primary water storage tank (TK-16)	1	Complete demo.; remove pads; backfill
Boron water storage tank (TK-13 A/B)	1	Complete demo.; remove pads; backfill

Table 8 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
Refueling water storage tank (TK-4 & TK-54) and greenhouse	1	Complete demo.; remove pads; backfill
Offgas stack	1	Complete demolition
Aux boiler stack	1	Complete demolition
High radiation bunker	1	Demo. 3ft below grade; fill w/concrete rubble; backfill
Turbine building	2	Demo. 3ft below grade; fill w/concrete rubble; backfill
WART/I&C building	2	Complete demolition; backfill
LLWB - Low level waste building	2	Complete demolition; backfill
FI - Foxbird island	2	Demo. equip to 3ft below grade; backfill
Security/Gatehouse building	3	Complete demolition; backfill
Administration building	3	Complete demolition; backfill
Fuel oil bunker	3	Demo. 3ft below grade; fill w/concrete rubble; backfill
Gas house	3	Complete demolition; backfill
Circ water pump house	3	Demo. 3ft below grade; fill w/concrete rubble; backfill
Temporary power shack - West of fuel building	3	Complete demolition; backfill
Modular office buildings (2)	3	Complete demolition; backfill
8 Sided storage building	3	Complete demolition; backfill
Test pit	3	Complete demolition; backfill
West - area West of the containment Building including (3) guard towers	3	Complete demolition; backfill

Table 8 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
Condensate storage tank (concrete pad only)	3	Complete demolition; backfill
Transformer including elect tower and concrete structures/pad	see below	see below
1. X-IA	3	Remove transformer for disposition; demo. pads; backfill
2. X-IB	3	Remove transformer for disposition; demo. pads; backfill
3. X-24	3	Remove transformer for disposition; demo. pads; backfill
4. X-26	3	Remove transformer for disposition; demo. pads; backfill
5. X-28	3	Remove transformer for disposition; demo. pads; backfill
6. X-IS	3	Remove transformer for disposition; demo. pads; backfill
7. X- 14	3	Remove transformer for disposition; demo. pads; backfill
8. X-16	3	Remove transformer for disposition; demo. pads; backfill
9. X- I 6A	3	Remove transformer for disposition; demo. pads; backfill
10. X- 16B	3	Remove transformer for disposition; demo. pads; backfill
Temp generator/transformer/pagoda (cold & dark equipment)	3	Remove transformer for disposition; demo. pagoda; backfill

Table 8 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
Outside protected area		
STPI - Sewage treatment plant	3	Demo. 3ft below grade; fill w/concrete rubble; backfill
Lift station	3	Demo. 3ft below grade; backfill
Information center bldg (including concrete pad and blocks 4ft x 4ft x4ft)	3	Complete demolition; backfill
Collection site	3	Complete demolition; backfill
Staff building/staff tunnel	3	Demo. 3ft below grade; fill w/concrete rubble; backfill
Annex - WHSE 4	3	Complete demolition; backfill
WHSE 2 & 3 - warehouse	3	Complete demolition; backfill
WHSE 5 - warehouse	3	Complete demolition; backfill
115KV transformer yard	3	Remove transformer for disposition; demo. pads; backfill
FPH - fire pump house/retention pond	3	Demo.; level; backfill
Seal pit (outfall)	3	Demo. to 3ft below grade; backfill
Barge slip area	3	grade area adjacent to slip
Pre-fab building (generator) and yard	3	Complete demolition; backfill
ELECT towers - from and to 115KV yard	3	Demo. towers and pads; backfill
Railroad cars (4) - includes track spur from fuel bldg to the property line	3	Demo. rail cars; remove rails to location designated; grade
Well water house	3	Demo. to 3ft below grade; backfill
MET equip. inside fenced area	3	Complete demolition; backfill

Table 8 Structures and Facilities Within the Scope of Work for Demolition		
Building or area description	**Condition of release**	**Final configuration of structure**
Electrical fence (around radio tower area)	3	Complete demolition; backfill
Radio tower	3	Complete demolition; backfill
Enclosed fences (around radio tower ties)	3	Complete demolition- backfill
Concrete pad & fence	3	Complete demolition; backfill
Concrete blocks - 4ft x 4ft x4ft Vehicle barriers	3	Remove initially and store for use; then demo.
Remove outside utilities - 3 feet below grade		
1. Potable water	3	Demo. 3ft below grade; backfill
2. Utility light poles	3	Demo. 3ft below grade; backfill
3. Fire hydrant hose stations	3	Demo. 3ft below grade; backfill
4. Sanitary lines	3	Demo. 3ft below grade; backfill
Microwave tower	3	Demo tower, ties, and pad; backfill

Notes:

a. Number 1 denotes that the bldg/area will undergo gross decontamination and the commodities will be removed prior to building demolition.

b. Number 2 denotes that the bldg/area will have commodities removed and rough decontamination will be done as required. Remaining commodities will remain as is for building demolition.

c. Number 3 denotes that the bldg/area is clean and the commodities will remain as is for building demolition.

As a result of this approach, the following sequence of dismantlement and demolition will occur for buildings within Area #1:

a. Strip, package, ship commodities from the buildings (piping, steel, components, etc.) Commodities, including building steel determined to be clean may be released to the demolition contractor.

b. Perform decontamination of the building concrete surfaces to levels below the unrestricted use criteria. Package the debris from decon. and ship for LLW processing and/or disposal.

c. Perform a final survey.

d. Release for demolition.

e. Demolish the building structure to 3ft below grade. Separate the clean rebar from the concrete.

f. Rubblize the concrete as necessary, and use as fill 3 feet below grade and onsite.

g. Release rebar using operational release procedures and ship rebar to demolition contractor.

The structures located within Area #2, are those that are on the cold side of the plant and have been maintained as radiologically "clean," with the exception of some systems and equipment that may have internal contamination. Within these areas, the process for demolition will follow this process:

a. Remediate, package, and ship of systems, components, commodities, identified within the site characterization report and assessed and bounded by our team. Structural steel of plant buildings will either be surveyed and released for demolition or dismantled for packaging and shipment as material..

b. Decontaminate, if required, to achieve the release criteria established within our FSS plan.

c. Perform radiation surveys to allow material release to the demolition contractor.

d. Release for demolition to the contractor.

e. Demolish structures and foundations to depth specified.

f. Subsurface piping to be handled in accordance with approved environmental plan

g. Perform final grade.

The buildings, structures, and facilities located within, what we have identified as Area # 3, are those that do not have a history of contamination and are therefore classified as "probably clean". In certain cases there are minor exceptions to this generalization, based upon the information we have received in the site characterization report, such as a small area within the information center and the staff building, that appear to have been remediated. Also, the site characterization report identifies higher activity levels within the basement of the environmental lab (Bailey House), that we believe may be attributed to background from the granite. However, we will evaluate and release these individual areas in accordance with our FSS plan to allow for demolition.

Therefore, Area #3 buildings, structures, and facilities will be processed as follows:

a. Remove ancillary equipment required for asset recovery (furniture, etc.) - (It is assumed that MY will remove equipment designated for asset recovery, prior to the scheduled remediation/ demolition of the structure).

b. Perform survey in accordance with established procedures and criteria.

c. Release for demolition to the contractor.

3.8.3 Dismantle Commodities

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

a. Areas within the RCA of the plant - commodities will be removed from each area of the building/structure/yard, packaged, and transported for disposal as radiological waste (Class A, B, or C).

b. Areas within the non-RCA of the plant that may have some internal system contamination - systems identified during the site characterization and subsequently verified and bounded by our RP team, will be remediated, and the balance of the building/structure surveyed and released to our demolition contractor. Remediation (leaving as is, removal, capping, or grouting) will depend on the level of radioactive contamination found (if any).

c. Areas outside of the Restricted Area (RA) that have never been exposed to radiological contamination - Commodities will be removed with the building/structure by our demolition contractor, upon completion of appropriate survey.

3.8.4 Material Disposition

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

Concrete materials generated from the Hot Side of the plant, that meet the volumetric DCGL value following rough decontamination and rubblization, will be deposited within the lower levels of the plant structures and used as fill. Disposal of building reinforcing steel and structural steel, which has been released, will be performed by the demolition contractor to scrap and/or landfill areas.

3.9 ISFSI D&D/remediation

Upon DOE taking possession and the shipment of all the storage containers offsite, MYAPC will decommission the ISFSI in accordance this LTP. MYAPC anticipates that there will be no radioactive contamination leakage detected from leaking spent fuel casks during the life of the ISFSI, and therefore intends to use the December 1997 MARSSIM Appendix B protocol to release the ISFSI after the spent fuel has been removed. Our basis for using, and application of, the December 1997 MARSSIM Appendix B protocols are as follows:

A large number of users of radioactive materials may use a simplified procedure to demonstrate regulatory compliance for decommissioning, avoiding complex final status surveys. Sites that qualify for simplified decommissioning procedures are those where radioactive materials have been used or stored only in the form of non-leaking, sealed sources.

As such, MYAPC will qualify for implementation of a simplified procedure and will provide the NRC with: (1) a certification that, on the concrete pad used to store spent fuel casks, no

residual radioactive contamination attributable to the user's activities is detectable above the building surface DCGLs listed in Section 6; and (2) documentation on the removal of the spent fuel from the ISFSI. This information will be used by the NRC to document protection of both the public health and safety and the environment, based on the transfer of the spent fuel.

The absence of radioactive contamination will be demonstrated by: (1) documenting the amounts, kinds and uses of radio nuclides as well as the processes involved; (2) conducting a final status radiation survey of the concrete pad used to store spent fuel casks; and (3) submitting a report on this survey.

A FSS will be conducted to demonstrate compliance with the release criterion. The following information will be included in the FSS report:

a. Basis for selecting the instrumentation used for the survey

b. Nature of the radio nuclides surveyed

c. Measurement techniques and instruments used, including references for procedures and protocols used to perform the measurements

d. Minimum detectable concentrations (MDCs) of the instruments and measurement systems used to perform the measurements

e. Calibration, field testing, and maintenance of the instrumentation

f. Qualifications of the personnel using the instrumentation

g. Methods used to interpret the survey measurements

h. Qualifications of the personnel interpreting the survey measurements

i. Measurement results and measurement locations including the operator's name, instrument model and serial number, date the measurement was performed, and traceability of the measurement location

The number of measurements that will be required on the concrete pad used to store spent fuel casks has been determined using the following reasonably conservative values: $\alpha = 0.05$, $\beta = 0.05$, and $\Delta/\sigma = 1$. Therefore, at least 30 measurements will be made for the final status radiation survey on the concrete pad used to store spent fuel casks.

The results of the survey will be compared to the building surface derived concentration

guideline levels (DCGLs) given in Section 6, using an appropriate statistical test, such as the Sign or Wilcoxon test. If all measurements are less than the $DCGL_W$, then the statistics do not need to be addressed because the conclusions are obvious. If the mean of the measurements exceeds the $DCGL_W$, the survey unit obviously fails to demonstrate compliance and the statistics do not need to be addressed.

Radiation levels and concentrations will be reported as follows:

a. For external dose rates, units of:
   - micro-rem per hour at one meter from surfaces;

b. or levels of radioactive materials, including alpha and beta measurements, units of:
   - dpm/100 cm^2 (removable and fixed) for surfaces.

ATTACHMENT A

Drawings Associated with Specific Decommissioning Tasks

MAINE YANKEE

LTP SECTION 4

SITE REMEDIATION PLAN

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4.0 SITE REMEDIATION PLAN

4.1 Remediation Actions and ALARA Evaluations

This section of the LTP describes various remediation actions which may be used during the decommissioning of MY. In addition, the methods used to reduce residual contamination to levels that are ALARA as per 10CFR20 Subpart E are described. Finally, the Radiation Protection Program requirements for the remediation are described.

4.2 Remediation Actions

To achieve a DCGL, remediation actions are performed throughout the decommissioning process based on local area survey data. The remediation action taken is dependent on the type of area contaminated. The principal areas that may be subjected to remediation activities are structures and soils.

4.2.1 Structures

Following the removal of equipment and components, structures will be surveyed and contaminated building materials will be decontaminated or removed and disposed of as radioactive waste. With the exception of components that may remain under a repower option; the primary structure surfaces which would be present are concrete. Building materials that meet the site requirements for unconditional release will be released as scrap or be disposed of in a landfill. Contaminated structure concrete will be remediated to a level which will meet the radiological criteria for rubblization as presented in Section 6.0. The remediated and rubblized concrete will be placed into building foundations and backfilled with soil to grade level.

Remediation techniques that will be used for the remaining structure surfaces include washing, wiping, pressure washing, vacuuming, scabbling and chipping. Washing, wiping, vacuuming and pressure washing techniques may be used for both metal and concrete surfaces. Scabbling and chipping are mechanical surface removal methods that are intended for concrete structures. The principle remediation method that will be used is scabbling.

Pressure washing requires the use of vacuum units to remove the liquids generated. Dry vacuuming is an action expected to be performed during the normal remediation activities and is not evaluated as a separate ALARA action.

Washing and wiping techniques are actions that are normally performed during the course of remediation activities and will not always be evaluated as a separate ALARA action. When washing and wiping techniques are used as the sole means to reduce residual contamination below DCGL levels, ALARA evaluations are performed. Washing and wiping techniques used as a house keeping or good practice measure will not be evaluated. Examples of washing and wiping activities for which ALARA evaluations would be performed include:

a. Decontamination of stairs and rails.

b. Decontamination of structural materials, metals or media for which decontamination reagents may be required.

c. Structure areas that do not provide sufficient access for utilization of other decontamination equipment.

Needle gunning is a technique where pneumatically operated, thin hardened steel rods are used to abrade concrete surfaces. The pistol grip housing includes a vacuum attachment that captures the abraded media for subsequent disposal. Needle guns are typically used in regions where conventional scabbling equipment cannot reach. Examples include cracks and inside corners. This activity is used in conjunction with surface scabbling . Needle gunning techniques can also be applied to painted and oxidized surfaces. It is important to note that needle gunning is a form of scabbling.

Chipping includes the use of pneumatically operated chisels and similar tools coupled to vacuum assisted collection devices. Chipping activities are usually reserved for cracks and crevices but may also be used in lieu of concrete saws to remove pedestal bases or similar equipment platforms. This action is also a form of scabbling.

As the structures are rubblized, contaminated embedded piping will be removed and disposed as radioactive waste. Any remaining contaminated piping in the below grade concrete would be remediated using decontamination such as grit blasting techniques. Under a repower option, significantly contaminated embedded piping will be remediated. Grit blasting utilizes grit media such as garnet or sand under intermediate air pressure directed through a nozzle that is pulled through the closed piping at a fixed rate. The grit blasting action removes the interior surface media layer of the piping. A HEPA vacuum system maintains the sections being cleaned under negative pressure and collects the media for reuse or disposal. The final system pass is performed with clean grit to remove any residual contamination.

4.2.2 Soils

Soil contamination above the site specific DCGL will be removed and disposed of as radioactive waste. Clean fill will be used for the excavated areas. Clean fill is defined as fill which does not contain site derived radionuclides that exceed the site specific DCGL. The site characterization process established the location, depth and extent of soil contamination. As needed, additional investigations will be performed to insure that any changing soil contamination profile during the remediation actions is adequately identified and addressed. Soil remediation will be initially addressed by the removal of media for any identified areas which are outside the RCAs. A majority of site soil contamination is associated with three distinct areas (the PWST, RWST and the Shielded Radioactive Waste Storage Area) within the RA. The remediation of these areas will be performed following the removal of associated or adjacent tanks, components and pad interferences.

The contaminants within the RWST area are primarily due to past spill and heater leak incidents associated with the tank. Soil remediation will require removal of media to an averaged depth of approximately 1 meter immediately adjacent to the tank area. Additional remediation activities are expected to encompass a depth of 30 to 60 centimeters in the area down gradient from the tank and bounded east and west by local surface contour and the forebay berm. As required, additional soil profiling will be conducted to confirm the final survey criteria are met.

Soil contamination near the PWST is due to the past storage of radioactively contaminated components and waste storage containers in the area immediately east and north of the PWST area. Local terrain features were such that associated contaminants subjected to weathering conditions would be transported toward the PWST area. The averaged soil remediation depth in this region is less than 60 centimeters.

Contaminated soils associated with the Shielded Radiological Waste Storage area originated in part, from seasonal weathering conditions and specific tasks associated with components and stored containers. This area was evaluated in the past. A new bed of asphalt was placed over the region to mitigate the migration of any residual contaminants. The averaged soil contamination depth in this region is less than 60 centimeters.

Soil remediation equipment will include, but is not limited to, back and track hoe excavators. As practical, when the remediation depth approaches the soil interface region for unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique will be used. This simple methodology minimizes the

mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. For excavated regions, clean fill will be used to replace the excavated media which do not meet the DCGL criteria.

4.3 Remediation Activities Impact on the Radiation Protection Program

The Radiation Protection Program approved for decommissioning is similar to the Program in place during 25 years of commercial power operation. During power operations, contaminated structures, systems and components were decontaminated in order to perform maintenance or repair actions. The techniques used were the same as those projected for decommissioning. Many components were removed and replaced during operation. The techniques used for component removal were the same as those planned for use during decommissioning.

The MY Radiation Protection Program adequately controlled radiation and radioactive contamination during decontamination and equipment removal processes. The same controls will be used during decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas. Decommissioning does not present any new challenge to the Radiation Protection Program above those encountered during normal plant operation and refueling. Decommissioning allows radiation protection personnel to focus on each area of the site and plan each activity well before execution of the remediation technique.

Low levels of surface contamination are expected to be remediated by washing and wiping. These techniques have been used over the operational history of the facility. Water washing with detergent has been the method of choice for large area decontamination. Wiping with detergent soaked or oil-impregnated media has been used on small items, overhead spaces and small hand tools to remove surface contaminants. These same techniques will be applied to remediation of lightly contaminated structure surfaces during remediation actions.

Intermediate levels of contamination and contamination on the internal surfaces of piping or components has been subjected to high-pressure washing, hydrolazing or grit blasting in the past. The refueling cavity has been decontaminated by both pressure washing and hydrolazing. Pipes, surfaces and drain lines have been cleaned and hot spots removed using hydrolazing or grit blasting. Small tools, hoses and cables have been pressure washed in a self-contained glove box to remove surface contamination. These methods will be used to reduce contamination on moderately contaminated exterior surfaces as well as internal surfaces of pipes or components during decommissioning.

Scabbling or other surface removal techniques will reduce high levels of contamination, including contaminated concrete. Concrete cutting or surface scabbling has been used at MY in the past during or prior to installation of new equipment or structures both outside and inside

the RCA.

Abrasive water jet and mechanical cutting of components will be used to reduce the volume of reactor internals. Mechanical cutting was used at this facility during past operations. Abrasive water jet cutting utilizes similar actions as hydrolazing and grit blasting which have been used at the site in the past. The current radiation protection program provides adequate controls for these actions.

The decommissioning organization is experienced in and capable of applying these remediation techniques on contaminated systems, structures or components during decommissioning. The Radiation Protection Program is adequate to safely control the radiological aspects of this work and no changes to the Program are necessary in order to ensure the health and safety of the workers or the public.

4.4 ALARA Evaluations

As described in Section 6, dose assessment scenarios were evaluated for the residual contamination on building surfaces; the building occupancy scenario and the demolition scenario. The ALARA analysis is dependent on the assumed scenario because the surface area representing 25 mrem/y differs for each scenario. In addition, the assumed population density differs in the two scenarios. These issues were evaluated to determine whether the building occupancy scenario or the demolition scenario would be the most conservative for the ALARA evaluation. It was concluded that the building occupancy scenario was the most conservative to evaluate since the area corresponding to 25 mrem/y is smaller i.e. 200 m^2 vs. 2700 m^2, and the population density is greater, i.e. 0.09 person/m^2. Therefore the parameters for the building occupancy scenario were used in the ALARA evaluation and assumed to conservatively represent the demolition scenario as well.

4.4.1 ALARA Evaluation Structure Model

The costs are based on the floor of a 8 by 8 (64 m^2) by 3 meter structure or room (192 m^3). The evaluation dimensions used are found in NUREG 5512 Volume 2, Section 6.2.1. The room size used in NUREG 5512 (floor area, 64 m^2) provides conservatism to the receptor for exposure dose resulting from surface contamination. Investigations at MY indicate that wall contamination is spatially present but not to the extent as the model provided in NUREG 5512 (50% of the floor contamination level). The range of structure wall contamination at MY is estimated at 4 to 10 percent of the contaminated surface total. The ALARA evaluations used conservatively do not include wall surface area.

4.4.2 ALARA Evaluations and Generic Screening Levels

The Draft NMSS Decommissioning Standard Review Plan, Section 7.0, ALARA Analysis, states, "Licensees or responsible parties that remediate building surfaces or soil to the generic screening levels established by the NRC staff do not need to demonstrate that these levels are ALARA. However, licensees or responsible parties should remediate their facility below these levels through practices such as good housekeeping. In addition, Licensees or responsible parties should provide a description in the final status survey report of how these practices were employed to achieve the final activity levels." While structure and soil remediation will be performed to generic screening levels, MY is providing ALARA evaluations of the remediation actions.

The ALARA evaluations are performed in accordance with the guidance in Regulatory Guidance DG-4006. A spreadsheet format is used to account for the dose contribution of each radionuclide present in the source term. The principle equations used for the calculation are presented in Appendix A. The evaluation determines if the desired beneficial effects from the remediation action are greater or less than the costs of the action. When the benefits are greater the remediation action should be performed and conversely when the benefits are less, the remediation actions are not required. The evaluation requires that a reasonable determination of the remediation action costs be performed. The cost emphasizes the remediation methods expected to be used and is weighed against the dose that is averted while taking into account the changing monetary worth over the radionuclide decay which occurs during the evaluation time period. The evaluation time period used is 1000 years (in lieu of the 70 year factor for structures). The extended period is used for evaluating rubblized structure media. The population density for the scenario is also included in order to evaluate the dose to the critical group.

4.4.3 Remediation Costs

For the MY facility the remediation techniques examined are scabbling, pressure water washing, wet and dry wiping and grit blasting for embedded piping. The principle remediation method will be scabbling. The individual elements and total costs for each of the remediation activities are provided in the tables of Appendix B. The cost elements are defined in Appendix A.2, *Calculation of Total Cost*. The principal surfaces for which remediation activities will be performed will be concrete.

a. Site Structure Contaminants

Current investigation of characterization data relating to structure concrete surfaces at the MY facility indicate that a major fraction (>95%) of the contamination occurs in the top millimeter of the concrete. The investigation

was based on the evaluation of 42 concrete cores acquired from representative areas of the site primary structures as described in Section 2 of the LTP. Scabbling actions result in the removal of the top 0.125 to 0.25 inches (0.318 to 0.635 cm) of concrete. The ALARA evaluation was performed by bounding the costs estimate for a scabbled depth of 0.125 and 0.25 inches. For each evaluation the same manpower and equipment costs are used. The major variable for the bounding conditions are the costs associated with manpower and waste disposal.

b. Structure Activated Concretes

Concrete activation is associated primarily with the containment structure. Characterization of the reactor associated bioshield and loop area concretes have provided specific information regarding the identification, concentration and distribution of the radionuclides. In addition to the observed concrete activation products ^{152}Eu, ^{154}Eu, ^{134}Cs and ^{60}Co, the concrete surfaces in the containment structure are radioactively contaminated by the deposition and transport mechanisms relative to fluids and airborne distribution which occurred during the operational history. To achieve the same dose reduction as that for concrete with only surface contamination the remediation costs for activated concrete will be greater.

4.4.4 Structure Cost Evaluation Details

The principle concrete surface area used for the evaluation is a room with dimensions 8 meters by 8 meters. The lower walls are not included in the evaluation. The table below provides the estimated waste volume for rubble generated during scabbling operations at different depths.

Table 4.1
8 by 8 Meter Room (64 m^2) Concrete Volume Estimate For Scabbled Depth

Scabble Depth inches (cm)	Waste Volume, ft^3 (m^3)
0.125 (0.318)	7.18 (0.203)
0.250 (0.635)	14.35 (0.406)

a. Remediation Action and ALARA Evaluation Considerations

Specific evaluation information is needed to adequately perform the remediation activities and perform unbiased and sound ALARA evaluations prior to remediation activities. Some items that affect these parameters are:

- Depth of contaminants
- Surface area(s) of contamination relative to total
- Types of surfaces: vertical walls, overhead surfaces, media condition
- Consumable items and equipment parts
- Cleaning rate and efficiency (decontamination factor)
- Work crew size
- Support activities such as, waste packaging and transfer, set up time and interfering activities for other tasks
- Waste volume

b. Scabbling

Unit costs were determined by reviewing Walkdown Summary Sheets. DOC Contractor representatives and Radiation Protection personnel performed walkdown inspections of the rooms in each structure. The walkdown evaluation includes a summary of the remediation activities anticipated for each structure and, the estimated surface area in each structure, room or location that will require scabbling (10,276 m^2). The DOC has estimated that scabbling activities utilizing scabbling tools and needle guns can be effectively accomplished on smooth concrete surfaces to a depth of 0.25 to 0.5 inches at a rate of 40 ft^2 per hour. The operation is performed using one equipment operator and one laborer. In addition, costs for radiation protection support activities and supervision are included.. Attachment B provides a breakdown of the cost elements.

The unit cost (Table 4.2) is used to determine the cost for remediating an 8 m by 8 m room (64 m^2). The unit and remediation cost is presented in Table 4.3.

Scabbling the room assumes that 100 % of the room floor surfaces contain contamination above the DCGL of which 95% of the residual activity is removed by the remediation action. The equipment is capable of scabbling 3.72 square meters per hour. The debris is vacuumed into collectors that are transferred into roll-off or transport containers. For the evaluation, 48 cubic foot transport containers are assumed and result in a greater volume shipped per load. The shipped volume used in the estimate is 7.93 m^3 per shipment where highway transport weight is the limiting factor for transport. Contamination reduction rates are very high (>95%). Based on evaluation of concrete core samples, scabbling is expected to be the principle method for the remediation of concrete surfaces. The cost elements used to derive the unit costs for the evaluation are listed in Attachment B. The formula associated with the cost elements are provided in Attachment A.

c. Pressure Water Washing

The unit costs provided in Table 4.2 for water washing was established by assuming10% of the site structures surface area that was derived from the structure walkdown inspections is pressure washed. This information was used to provide a cost per meter square factor. Appendix B provides the cost details. Table 4.3 provides the cost for a 8 meter by 8 meter room. This equipment consists of a hydrolazer and when used, a header assembly. The hydrolazer nozzle directs the jet of pressurized water that removes surficial materials from the concrete. The header when used minimizes over-spray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing. The cleaning speed is approximately 44 square meters per hour and generates about 5.4 liters of liquid per square meter. The contamination reduction rates are dependent on the media in which the contaminants are fixed, the composition of the contaminants, cleaning reagents used and water jet pressure. Mitigation of loose contaminants is high. Reduction of hard-to-remove surface contamination is approximately 25% for the jet pressure and cleaning speed used. The use of reagents and slower speeds can provide better contamination reduction rates but at proportionally higher costs. The operation is performed using one equipment operator and two laborers. In addition, costs for radiation protection support activities and supervision are included. The formula associated with the cost elements are provided in Attachment A and the cost elements are provided in Attachment B

d. Wet and Dry Wiping

The unit costs provided in Table 4.2 for wet and dry wiping assumes 10% of

the site structures surface area that was derived from the structure walkdown inspections is cleaned. The information is used to develop a cost per meter square. Appendix B provides the detail costs. The cost for remediating a 8 meter by 8 meter room is provided in Table 4.3 Wet wiping consist of using a cleaning reagent and wipes to address surfaces that cannot be otherwise cleaned or decontaminated. Dry wiping includes the use of oil-impregnated media to pickup and hold contaminants. The cleaning rate of these actions are estimated at 2.8 square meters per hour (~ two minutes per square foot). This action is labor intensive. The action is effective for the removal of loose contaminants and reduction of surface contaminants especially when cleaning reagents are used. The evaluation assumes the action removes the loose and 20.0 percent of the hard -to-remove contaminants. Waste generation is about 0.005 m^3 per hour. Decontamination factors vary and are dependant on factors such as, the reagents that are used, the level of wiping effort and the chemical and physical composition of the contaminant. The contamination reduction efficiency used for wet and dry wiping is 20 percent. Removal of loose contaminants is very effective (100 percent). The operation is performed using and two laborer. In addition, the cost for radiation protection support activities a operating engineer and supervision are included. The formula associated with the cost elements are provided in Attachment A. Attachment B list the cost elements used for the evaluation.

e. Grit Blasting (Embedded Piping)

Because the ALARA evaluation for embedded piping left in place must consider the external gamma dose from the piping (if significant) be added to the TEDE resulting from residual surface contamination of a room, only the unit costs are provided in Table 4.3. The cost for grit blasting was established by assuming that 1000 linear feet of piping is decontaminated. The entire interior surface is assumed to require decontamination and the internal diameter is assumed at 4 inches (typical drain line dimensions). The grit blasting system is comprised of a hopper assembly that delivers a grit media (garnet or sand) at intermediate air pressures through a nozzle that is pulled at a fixed rate (~1 foot per minute) through the piping. A HEPA vacuum system maintains the piping system under a negative pressure and collects the grit for reuse (cyclone separator) or disposal. Usually several passes are required to effectively clean the piping to acceptable residual radioactive levels. The contamination reduction efficiency used for grit blasting is 80 percent. The final pass is made with clean grit to mitigate the possibility of loose residual contaminants associated with previous cleaning passes. Grit decontamination factors are related to pressure, nozzle size grit media and the number of passes made. A

nominal grit usage rate of one-half pound per linear foot is used in the calculation.. This cost unit information is provided as cost per linear foot factor. Appendix B provides the cost details used to derive unit cost. Mitigation of loose contaminants is high. The formula associated with the cost elements are provided in Attachment A.

ALARA evaluations will be addressed on a case basis for any remaining embedded piping. As stated in DG-4006, normal room surveys will account for direct (external gamma) radiation from the piping. The direct (external gamma) dose from the pipes will as necessary, be evaluated in addition to the TEDE from the residual radioactivity on the surfaces in the room.

f. Soil Excavation

The unit costs provided in Table 4.2 for soil excavation was established by assuming 2.48E+04 ft^3 (701.5 m^3) is excavated from the site. This information was used to provide a cost per cubic meter factor. These results are used to perform an ALARA evaluation where1000 m^2 of soil (0.15 m deep) is excavated (150 m^3). The equipment consists of an excavator that first moves the soil at the contaminated depth interface to a pile that is scooped into a staged shipping container. When filled, the container is moved from the excavation area with a fork lift. Contamination reduction is assumed at 95%. Unit costs are provided in table 4.2 The cost for the action and cost benefit of the averted dose are provided in Table 4.4. The operation is performed using two equipment operators and two laborers. Costs for radiation protection support activities and supervision are also included. The formula associated with the cost elements are provided in Attachment A and the cost elements are provided in Attachment B.

4.5 Unit Cost Estimates

In order to effectively perform ALARA evaluations and remediation actions, unit cost values are required. These values are used to perform the Regulatory Guide DG-4006 cost-benefit analysis previously described. The table below includes the cost for the site-specific concrete scabbling as described. The costs provided are relative to the removal of 0.25 inches of concrete. Values for other activities are also included. Several of the unit cost items provided in the table below present the site specific unit cost relative to the surface area being remediated.

MY has incorporated to the extent practical, spreadsheets and tables that as necessary, will be used to provide the data required for summarizing the costs associated with expected specific

remediation actions. The spreadsheets provide the resultant $DCGL_W$ fraction, the present worth of the future averted dose and the benefit from the collected averted dose. The spreadsheets include the radionuclide mixture and are weighted for the relative dose due to the mixture. Information relative to the cost provided, work rates, waste generation rates and similar data are maintained as a part of the technical basis documents developed for this section.

Table 4.2
Unit Cost Estimate

Remediation Technique	Unit Cost[a]	Remarks
Wiping/Washing[a]	$64.13/m²	Unit cost factors provided in Attachment B
Pressure Water Washing and Vacuuming	$12.27/m²	Unit cost factors provided in Attachment B
Concrete Scabbling (Upper Bounds)	$86.20/m²	Unit cost factors provided in Attachment B. Needle gun activities are included with scabbling
Concrete Scabbling (Lower Bounds)	$53.13/m²	Unit cost factors provided in Attachment B. Needle gun activities are included with scabbling
Grit Blasting Embedded Piping	$65.65/linear ft	Unit cost factors provided in Attachment B
Soil Excavation	$3388/m³	Unit cost factors provided in Attachment B

[a]The high cost for wiping and washing is due both to the labor intensive time required and the costs (49% of the total) of waste processing and disposal associated with the water used. Because radiation protection practices depict wiping as good practice for removing loose contamination, wiping is performed and not always as a function of an ALARA evaluation

[b]A contingency of 25% has been added to the person hour total for the activities

Table 4.3
Cost for 8 by 8 Meter Room (64 m²)

Remediation Action	Unit Cost per m²	Contamination Reduction Factor	Total Cost
Scabbling (Upper Bounds)	$86.20	0.95	$5,517
Scabbling (Lower Bounds)	$53.13	0.95	$3,400
Pressure Washing	$12.27	0.25	$785
Wiping and Washing	$64.13	0.20	$4,104

4.6 Structure ALARA Evaluation Results

For the remediation actions listed in Table 4.3 above, the desired beneficial effects from the remediation actions are less than the cost of the actions. This is determined by adjusting the terms of equations 11, 12 and 18 of Regulatory Guidance DG-4006 (see Appendix A for each equation) and addressing the dose contribution for each individual radionuclide as related to the respective DCGL. The individual radionuclide fraction is divided by its respective DCGL. The product (fraction/DCGL) is divided by the sum of the fractions (relative dose) and the result is the individual radionuclide dose contribution, *k*, which is included in the adjusted equation below. The sum of the results is the benefit of the averted dose (B_{AD}).

$$B_{AD} = \$2000 x P_D x A x 0.025 x k x F \left(\frac{1 - e^{-(r+\lambda)N}}{r + \lambda} \right)$$

The terms for the above equation are found in Appendix A Sections A.1 through A.4, and *k* is defined above. The spreadsheets in Appendix A, Section A.6 illustrate k and more importantly, the evaluation results which are summarized in Table 4.4 below for a 64 m^2 floor.

Table 4.4
Remediation Action Evaluation Results

Remediation Action	Total Cost	Cost Benefit of Averted Dose
Scabbling (Upper Bounds)	$5,517	$2411
Scabbling (Lower Bounds)	$3,400	$2,411
Pressure Washing	$785	$634
Washing and Wiping	$4,104	$508
Soil Excavation	$508,224	$595

The resulting evaluations demonstrate that the residual radioactivity level are already ALARA without employing the above remediation actions.

APPENDIX A

This appendix provides the method for calculating residual radioactivity levels that are ALARA. The term C_T is the same as the term $Cost_T$ described in the same document. The equations and definitions below substitute the term C for Cost.

A.1 Residual Radioactivity Level ALARA Calculation

(DG-4006 eq. 18).

$$\frac{Conc}{DCGL_W} = \frac{Cost_T}{(2000)(P_D)(0.025)(F)(A)} x \left[\frac{r+\lambda}{1-e^{-(r+\lambda)y}} \right]$$

Where:

$\frac{Conc}{DCGL_W}$	=	Residual radioactivity levels that are ALARA of fraction of $DCGL_W$
$Cost_T$	=	Total monetary cost of remediation action in dollars
2000	=	Is the dollar value of a person-rem averted ($/person-rem)
P_D	=	Population density for the critical group scenario (persons per m^2)
0.025	=	Annual dose to an average member of the critical group from residual radioactivity at the $DCGL_W$ concentration (rem/yr)
F	=	Fraction of the residual radioactivity removed by remediation action. F may be considered to be removable fraction for remediation action being evaluated.
A	=	Area being evaluated in m^2 (used with population density)
r	=	Monetary discount rate (yr^{-1})
λ	=	Radiological decay constant for the radionuclide (yr^{-1})
N	=	Number of years over which collective averted dose is calculated (yr)

Values for the equation parameters may be found in Regulatory Guide DG-4006 Table 3.1. The table below presents some of these generic values.

Equation Parameters		
Equation Terms	**DG-4006 Values**	
	Structure	**Land**
P_D	0.09	0.0004
r	0.07	0.03
N	70	1000[a]

[a] For the site specific calculations, structures to be rubblized result in N = 1000 due to the modeling scenario.

A.2 Calculation of Total Cost

(DG-4006 eq. 13)

In order to evaluate the cost of remediation actions Draft Regulatory Guide DG-4006 provides the elements necessary to derive the costs that are compared to the benefits. The total cost is:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + C_{TF} + C_{WDose} + C_{PDose} + C_{other}$$

The terms for "Cost" is abbreviated as "C" below.

C_T = Total costs (all the elements below)

C_R = Monetary cost of the remediation action (may include mobilization costs).

C_{WD} = Cost for generation and disposal of the waste generated by the action:

C_{WD} = V_A x C_V

- V_A Is the volume of waste produced, remediated in units of m^3 and;
- C_V is the cost of waste disposal per unit volume, including transport cost, in units of $\$/m^3$

C_{ACC} = Cost of worker accidents during the remediation action:

C_{ACC} = \$3,000,000 x F_W x T_A

- \$3,000,000 is cost of a fatality equivalent to \$2,000/person-rem;
- FW is the workplace fatality rate in fatalities/hour worked (4.20E-8/h) and;
- T_A is the worker time required for remediation in units of worker-hours.

C_{TF} = Cost of traffic fatalities during transport of the waste:

C_{TF} = \$3,000,000 x $_{VA}$ x $[(F_T \times D_T)/V_{ship}]$

- F_T is the fatality rate per kilometer traveled in units of fatalities/km (3.80E-8), and:
- DT is distance traveled in km;
- V_{SHIP} is volume of truck shipment in m^3(7.93 m^3).

C_{WDose} = $2,000 x D_R x T:

C_{WDose} == is the cost of the remediation worker dose

- $2000 is the cost of dose received by workers performing the remediation and transporting the waste to the disposal facility.
- D_R is total effective dose equivalent rate to remediation workers in units of rem/hr and,
- T is time worked to remediate the area in units of person-hours

C_{PDose} = Cost of the dose to the public from excavation, transport, and disposal of the waste (1.62 person-hours/m^3 of soil).

C_{other} = Other appropriate costs for the particular situation.

A.3 Calculation of Benefits

(DG-4006 eq. 11)

The benefit from collective averted dose is calculated by determining the present worth of the future collective averted dose and multiplying it by a factor to convert the dose to monetary value:

$$B_{AD} = (\$2000)[PW(AD_{COLLECTIVE})]$$

Where:

B_{AD} = benefit from averted dose for a remediation action, in $

$2,000 = value in dollars of a person-rem averted

$PW(AD_{COLLECTIVE})$ = present worth of future collective averted dose

A.4 Present Worth of Future Collective Averted Dose

(DG-4006 eq. 12)

The present worth of the future collective averted dose is estimated by:

$$PW(AD_{Collective}) = (P_D)(A)(0.025)(F)\left[\frac{Conc}{DCGL_W}\right]\left[\frac{1-e^{-(r+\lambda)N}}{r+\lambda}\right]$$

Where:

P_D = population density for the critical group scenario in people per m^2

A = Area being evaluated in m^2 and represents the floor area only for the attached ALARA calculations.

0.025 = Annual dose to an average member of the critical group from residual radioactivity at the $DCGL_W$ (concentration in rems/y

F = Fraction of the residual radioactivity removed by the remediation action, F may be considered to be the removable fraction for the remediation action being evaluated.

Conc = Average concentration of residual radioactivity in the area being evaluated in units of activity per unit area for buildings or activity per unit volume for soil.

$DCGL_W$ = derived concentration guideline level that represents a dose of 25 mrem/yr to the average member of the critical group, in the same units as "Conc"

r = monetary discount rate in units of y^{-1}

λ = radiological decay constant for the radionuclide in units of y^{-1}

N = number of years over which the collective dose will be calculated.

A.5 ALARA Evaluation Spreadsheets and Development

Evaluation spreadsheets which incorporate the final equation found in section 4.6 are included below for Scabbling, Pressure Washing, Washing and Wiping and Soil Excavation. The spreadsheets if necessary may be modified to address changes or additional regulatory guidance. The spreadsheets provide input for fraction of activity removed, total cost and remediation surface area. Other nuclide fractions can be input to address changes in mixtures and the DCGL values can be replaced as necessary.

Scabbling
(Upper Bounds)

Scabbling
(Lower Bounds)

Pressure Washing

Washing and Wiping

Soil Excavation

APPENDIX B

This Appendix provides the unit cost values used to develop the total cost C_T as defined in this section.

Supporting information and documents are as necessary, maintained with the evaluation files.

Significant changes to support ALARA evaluations will be maintained with the evaluation document files and as directed by guidance documents and applicable procedures.

Walkdown Summary Sheets

The DOC Contractor developed Walkdown Summary Sheets for each structure. Principle locations and areas within structures were also noted. The Summary Sheets provided information regarding specific remediation activities to be conducted in each area or structure. As a part of the walkdown, areas requiring scabbling were delineated. The areal summation is 110,606 ft^2 or 10,275.5 m^2. These values are also used to derive (10%) the regions used to estimate pressure washing and washing and wiping activities.

Remediation Activity Rates

Remediation activity rates were provided by the DOC contractor based on previous experience, from published literature, or from groups or vendors currently performing these or similar activities. Past operational experience was also used in developing the rates.

Contingency

A contingency of 1.25 was added to the manpower hours. Scabbling (the primary activity) was bounded using base hours and the volume of concrete (disposal cost) for remediating 0.125 inches versus the manpower contingency cost and the volume of concrete (disposal cost) for remediation 0.25 inches of concrete.

Equipment

Equipment cost were developed based on the cost of buying specific equipment and prorating the cost over the task activities. Rental rates are also included for specific equipment such as fork lifts and excavators. Consumable supplies and parts were included in the cost for equipment. Shipping containers were included with equipment cost.

Mobilization and Demobilization Costs

Cost were included for delivery and pick up of equipment. Anticipated cost to stage and move

equipment from location to location were also included.

Waste Disposal Cost

Disposal costs for generated waste were based on $90 per cubic foot ($3178.40 per cubic meter). Round trip transportation distance to Clive, Utah (Envirocare site) is 9207 km. Waste volume per shipment is dependant primarily on highway hauling weight restrictions and results in the use of a volume of 7.93 m^3. The distance and haul volume are used for determining transport accident cost in accordance with DG-4006 and Attachment A, Section A2.

Worker Accident Costs

To determine worker accident cost in accordance with DG-4006 and Attachment A, Section A2 the same hours input for manpower cost were used for worker accident cost.

Worker Dose

Cost associated with worker dose are a function of the hours worked and the workers radiation exposure for the task. General dose rates for each area of the walkdown summary sheets used were estimated. The results were summed and the average used for all remediation activities (7.3 mrem/h). For soil excavation a value of 4.0 mrem/h was used.

Manpower Costs

Manpower costs were provided by the DOC Contractor. The individual cost for the applicable disciplines, i.e. laborer, equipment operator, health physic technician ; were developed into an hourly crew rate for the task.

Unit Cost

The sum of the cost elements was divided by the applicable unit (m^2, m^3or linear feet) to provide a unit cost for the activity. Other cost units for cost per hour or linear foot were also developed in the same fashion. The tables to follow provide the crew cost per hour but do not provide the individual hourly rates for individual disciplines.

B.1 Pressure Water Washing And Vacuuming

Area Evaluated For Unit Cost Determination: 1028 m^2 = 0.10 x 10,275.5

Primary Crew Size:	2.0,	Operating Engineer, 1; and Laborer, 1
Support Personnel:	4.0,	Superintendent, Resident and Schedule Engineers, HP Technician

Hourly Cost:	$111.00
Cleaning Rate:	44.6 $m^{2/}$/h
Hours:	33.8 [(1028 m^2/44.6 $m^{2/}$/h) + 4 h set up)(1.25 cont)]
Mobilization Costs	$600
Manpower Cost:	$3,753
Equipment Costs:	$3,200
Liquid Processing Costs:	$1,833 [($1.00/gal)(5.4L/m^2)(1028 m^2)(1.25 liquid.. .... contingency)]/(3.78l/gal)
Waste Disposal Cost:	$2,638 Solids estimated at 8.0E-04 m^3/m^2 = 0.83 m^3 ($3178.4)
Worker Accident Cost:	$4.34 Per DG-4006
Transportation Accident Cost:	$110 Per DG-4006
Worker Dose:	$473 Per DG-4006
Total Costs:	$12,612
Cost per m^2:	$12.27

B.2 Washing and Wiping Remediation Actions

Area Evaluated For Unit Cost Determination: 0.10 x 10,275.5 = 1028 m^2

Primary Crew Size:	2.0,	Laborers
Support Personnel:	5.0,	Superintendent, Resident and Schedule Engineers, Operating Engineer and HP Technician
Hourly Cost:	$75.87	
Cleaning Rate:		2.8 $m^{2/}/h$
Hours:	504.8 [(1028 m^2/2.8 $m^{2/}$/h) + 4h/40h set up)(1.25 cont)]	
Mobilization Costs	$600	
Manpower Cost:	$38,301	
Equipment Costs:	$2,878	
Waste Generation:	5.14 m^3 (.005 m^3/m^2)	
Waste Disposal Cost:	$16,337 ($3178.40/m^3)	
Worker Accident Cost:	$64.82 Per DG-4006	
Transportation Accident Cost:	$680 Per DG-4006	
Worker Dose:	$7068 Per DG-4006	
Total Costs:	$65,929	
Cost per m^2:	$64.13	

B.3 Scabbling Remediation Action (Upper Bounding Condition)

Area Evaluated For Unit Cost Determination: 10,275.5 m^2

Primary Crew Size:	2.0, Operating Engineer, Laborer
Support Personnel:	4.0, Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	$86.26
Cleaning Rate:	3.72 $m^{2/}/h$
Hours:	3,802.2 [(10,275.5 m^2/3.72 $m^{2/}$/h) + 4h/40h set up) (1.25 cont)]

Mobilization Costs	$12,000
Manpower Cost:	$327,976
Equipment Costs:	$201,515 ($53./hr)
Waste Generation:	65.2 m^3 = (10,275.5 m^2)(6.35E-3 m)
Waste Disposal Cost:	$207,232 ($3178.40/m^3)
Worker Accident Cost:	$488 Per DG-4006
Transportation Accident Cost:	$8630 Per DG-4006
Worker Dose:	$55,710 Per DG-4006
Total Costs:	$813,550
Cost per m^2:	$79.17

B.3a Scabbling Remediation Action (Lower Bounding Condition)

Area Evaluated For Unit Cost Determination: 10,275.5 m^2

Primary Crew Size:	2.0,	Operating Engineer, Laborer
Support Personnel:	4.0,	Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	$86.26	
Cleaning Rate:	3.72 $m^{2/}$/h	
Hours:	2,765.2 [(10,275.5 m^2/3.72 $m^{2/}$/h)	
Mobilization Costs	$12,000	
Manpower Cost:	$238,528	
Equipment Costs:	$146,557 ($53./hr)	
Waste Generation:	32.6 m^3 = (10,275.5 m^2)(3.18E-3 m)	
Waste Disposal Cost:	$103,695 ($3178.40/m^3)	
Worker Accident Cost:	$355 Per DG-4006	
Transportation Accident Cost:	$4,318 Per DG-4006	
Worker Dose:	$40,516 Per DG-4006	
Total Costs:	$545,969	
Cost per m^2:	$53.13	

B.4 Grit Blasting Remediation Action

Area Evaluated For Unit Cost Determination: 1,000 linear feet

Primary Crew Size:	3.0,	Operating Engineer, 1; Laborers, 2
Support Personnel:	4.0,	Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	$139.17	
Cleaning Rate:	1 linear foot per minute (3 passes plus 1 clean grit pass =8,000 linear feet)	
Hours:	166.6 [(8,000 linear ft/m = (133.3 h)(1.25)]	
Mobilization Costs	$600	
Manpower Cost:	$23,189	
Equipment Costs:	$36,875	
Waste Generation:	0.8 m^3 = (8,000 linear feet x$10^{-4}m^3$/linear waste media generated)	
Waste Disposal Cost:	$2,524 ($3178.40/m^3)	
Worker Accident Cost:	$21 Per DG-4006	
Transportation Accident Cost:	$105 Per DG-4006	
Worker Dose:	$2,333 Per DG-4006	
Total Costs:	$65,647	
Cost per linear foot:	$65.65	

B.5 Soil Excavation Remediation Action

Area Evaluated For Unit Cost Determination: 701.5 m^3 = 24,775 ft^3

Primary Crew Size:	4.0,	Operating Engineers,2; Laborers, 2
Support Personnel:	4.0,	Superintendent, Resident and Schedule Engineers, and HP Technician
Hourly Cost:	$183.84	
Cleaning Rate:	7.65 m^3/h	
Hours:	201.9 [(701.5 m^3/7.65 m^3/h)+(91.8*4h/40h)(2.0 cont)] A contingency of 2.0 is use due to the loading articulation and continued excavator staging .	

Mobilization Costs	$700
Manpower Cost:	$37,112
Equipment Costs:	$15,670 ($90/h excavator, $45/h fork lift, consumables $2044
Waste Generation:	701.5 m^3 = (24,775 ft^3/35.315 ft^3/m^3)
Waste Disposal Cost:	$2,229,775 ($3178.40/m^3)
Worker Accident Cost:	$13 Per DG-4006
Transportation Accident Cost:	$92,854 Per DG-4006
Worker Dose:	$807 Per DG-4006
Total Costs:	$2,376,842
Cost per m^3:	$3,388

B.5A Soil Excavation Remediation Action

Area Evaluated For Unit Cost Determination: 1000 m^2 0.15 m depth

Primary Crew Size:	4.0,	Operating Engineers,2; Laborers, 2
Support Personnel:	4.0,	Superintendent, Resident and Schedule Engineers, and HP Technician
Cost:	$3,388/$m^3$	
Soil Volume:	150 m^3 (1000 m^2)(0.15 m)	
Cost:	$508,200	

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LTP SECTION 5

FINAL STATUS SURVEY PLAN

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ATTACHMENT A **Survey Unit Breakdown**

5.0 FINAL STATUS SURVEY PLAN

5.1 Introduction

5.1.1 Purpose

The Final Status Survey (FSS) Plan describes the final survey process used to demonstrate that the MY facility and site meet the radiological criteria for unrestricted use as specified in 10 CFR 20.1402.

5.1.2 Overview

The final status survey includes structures, land, and plant systems that are identified as contaminated or potentially contaminated, as a result of licensed activities. The majority of the survey effort will be required in the Containment Building, Fuel Building, Primary Auxiliary Building, Spray Building, RCA Storage Building and the surrounding yard areas. A final status survey of the Independent Spent Fuel Storage Installation (ISFSI) location will be conducted prior to construction of the concrete base.

There are 5 major steps in the final survey process: survey preparation, survey design, data collection, data assessment, and documentation of survey results.

a. Survey Preparation

Survey preparation is the first step in the final survey process and occurs after remediation is complete. In areas where remediation was required, a turnover survey is performed to confirm that remediation was successful. A turnover survey may be performed using the same process and controls as a final survey so that data from a turnover may be used as part of the final survey data. Following the turnover surveys, the final status survey is performed, if required.

The area to be surveyed is isolated and/or controlled to ensure that radioactive material is not reintroduced into the area from ongoing demolition or remediation activities nearby. Tools, equipment, and materials not needed to support survey activities are removed. Routine access, material storage, and worker transit through the area are no longer allowed.

An inspection of the area is conducted by FSS personnel to ensure that work is complete and it is ready for final status survey. Control of activities is transferred from the engineering/construction portion of the Decommissioning Operations Contractor (DOC) to the FSS organization. Approved procedures

provide isolation and control measures that are maintained until demolition or license termination.

b. Survey Design

The survey design process establishes the methods and performance criteria used to conduct the survey. Survey design assumptions are documented in "Survey Packages" in accordance with approved procedures. Areas of the site are assigned to survey units and classified by contamination potential as Class 1, Class 2 or Class 3. Survey unit size is based on factors specified in DG-4006 and other site specific factors related to building demolition. The percent coverage for scan surveys is determined. The number and location for direct and removable surface measurements and soil samples are established. Investigation levels are also established.

Replicate measurements are performed as part of the quality process established to identify, assess, and control errors and uncertainty associated with sampling, survey, or analytical activities. This quality control process provides assurance that the survey data meet the accuracy and reliability requirements necessary to support the decision to release or not release a survey unit.

c. Survey Data Collection

After preparation of a survey package, the final survey data are collected. Trained and qualified personnel perform the necessary measurements using calibrated instruments in accordance with the instructions contained in the survey package.

d. Survey Data Assessment

Survey data assessment is performed to verify that the data are sufficient to demonstrate that the survey unit meets the unrestricted use criterion. Statistical analyses are performed on the data and the data are compared to investigation levels. Depending on the results of an investigation, the survey unit may require further remediation, reclassification, and/or resurvey. Graphical representations of the data may be generated to provide qualitative information from the survey. The assumptions and requirements in the survey package are reviewed. Additional data needs, if required, are identified during this review.

e. Survey Results

Survey results are documented by survey unit in the document package. The final survey report is prepared summarizing the data for each survey unit and stating the conclusions.

5.1.3 Implementation

In a submittal to the NRC MN99-26 (dated 8/9/99), MY described the schedule for the phased release of site land. Two areas have been determined to be non-impacted (as described in section 2 of the LTP) and may be released at any time. The remaining site areas are subject to a final status survey in accordance with this plan.

The FSS Plan is itself implemented in phases. The first phase is comprised of the survey of the non-Restricted Area (RA) land and buildings outside the Industrial Area fence. The second phase surveys the non-RA buildings, including the Turbine Hall, within the Industrial Area. The third phase includes the survey of the RA land and buildings including the structural concrete. The results of each phase will be described in a written report to the NRC. The actual structures and land included in each written report may vary depending on the status of ongoing decommissioning activities.

5.1.4 Regulatory Requirements and Industry Guidance

This plan has been developed using the guidance contained in the following documents: (The primary guidance for demonstrating compliance with 10CFR20, subpart E is considered to be Regulatory Guide DG-4006. The FSS Plan is organized to be as consistent as possible with the format of DG-4006 (except for the submitted alternate information found acceptable for initial acceptance for review), as described in MYAPC Letter to the NRC dated October 13, 1999, MN99-29)

a. Draft Regulatory Guide DG-4006, "Demonstrating Compliance With the Radiological Criteria for License Termination" (August 1998 draft).

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

c. NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," Revision 1 (June 1998 draft).

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

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

f. NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans", (December 1998, draft).

Other documents used in the preparation of this plan are listed in the References Section.

5.2 Classification of Areas

Prior to beginning the final status survey, a thorough characterization of the radiological status and history of the site was completed. The methods and results from site characterization are described in Section 2 of the License Termination Plan. Based on the characterization results, the structures and open land areas were classified following the guidance in section 2.1 of DG-4006. (Unless the site is repowered, there will be no systems remaining following decommissioning. Contaminated systems will be disposed of as radioactive waste and non-radioactive systems will be disposed of as scrap.) Area classification ensures that the number of measurements, and the scan coverage, are commensurate with the potential for residual contamination to exceed the unrestricted use criteria.

Initial classification of site areas is based on historical information and site characterization data. Data from operational surveys done in support of decommissioning, routine surveillance and any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the final status survey. Once the final status survey has begun, no change in classification will be made until the data evaluation phase. Any reclassification will be from a lower class to a higher class (eg. from Class 2 to Class 1) but no reclassification from a higher class to a lower class (eg. from Class 2 to Class 3) will be allowed. The classification process is performed in accordance with an approved procedure.

5.2.1 Non-Impacted Areas

These are areas that have no reasonable potential for residual contamination. There has been no impact from site operations in these areas. These areas do not need any level of survey coverage and no surveys are performed.

5.2.2 Impacted Areas

These are areas that may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations.

a. Class 1 areas are impacted areas that, prior to remediation, are expected to contain residual contamination in excess of the DCGLw.

b. Class 2 areas are impacted areas that, prior to remediation, are not likely to contain residual radioactivity in excess of the DCGLw.

c. Class 3 areas are impacted areas that have a low probability of containing residual radioactivity.

5.2.3 Initial Classification

Based on more than 19,000 measurements made during the site characterization and the information evaluated as part of the Historical Site Assessment, the scope of the final status survey includes land and structures south of the Old Ferry Road. The areas to the north and west have been shown to meet the non-impacted criteria (Site Characterization Summary). The scope and boundaries of the FSS will be increased if survey data show significant levels of radioactivity above background in peripheral areas.

Characterization was performed and reported by survey area. Each area was identified by a unique package number. The initial classification of survey areas used these area package numbers for consistency. For example, R0500 was the characterization survey package number for Bailey Point and R0500 will be the designator for the FSS package for Bailey Point as shown on Table 1B. Survey area classifications are presented in the following tables, 1A and 1B. Classification of the survey units within these areas is made in accordance with LTP Section 5.3. Building classifications include roofs and drains. The sigma values (either dpm/100 cm^2 or pCi/g) were calculated from characterization data. The sigma values given in Tables 1A and 1B were calculated in accordance with accepted methods using the sample values below the expected $DCGL_w$ since it is expected that most areas above the $DCGL_w$ will be remediated. However, actual sigmas may be recalculated based on data obtained from post-remediation or post-demolition surveys. Structure sigmas were based on 7 to 98 measurements while soil sigmas were based on 5 to73 measurements per survey area. Some areas have been assigned more than one survey unit and classification based on the levels of activity found (e.g., B0500, R0200, R0500, R1000 and R1300).

Table 1A Survey Area Classification-Structures			
Package Number	**Survey Area-Structures**	**Sigma (dpm/100 cm²)**	**Classification**
A0100	Containment-El.-2ft	5,744	1
A0200	Containment-El.20ft	8,339	1
A0300	Containment-El.46ft	5,631	1
A0400	Fuel Bldg	3,732	1
A0500	DWST (Tk-21)	760	2
A0600	PAB-El.11ft	3,811	1
A0700	PAB-El.21ft	3,027	1
A0800	PAB-El.36ft	1,576	1
A0900	Service Bld. Hot Side	1,456	1
A1100	LSA Bld	2,180	1
A1200	RCA Bld	4,880	1
A1300	Equipment Hatch	240	1
A1400	Personnel Hatch	1,390	1
A1500	Mechanical Penetration	812	2
A1600	Electrical Penetration	319	3
A1700	Containment Spray Bld	7,550	1
A1800	Aux Feed Pump Rm	247	2
A1900	HV-9 Area	510	1
A2100	RWST (Tk-4)	7,762	1
A2200	BWST	3,761	1
A2300	PWST	1,262	1
A2400	Test Tanks	778	1
B0100	Turbine Bld-El.61ft	239	3
B0200	Control Rm	317	3
B0300	Motor Control Center	279	3

Table 1A Survey Area Classification-Structures			
Package Number	**Survey Area-Structures**	**Sigma (dpm/100 cm²)**	**Classification**
B0400	Fire Pump House	317	3
B0500	Condenser Bay/Comp Cool.	727	3,1
B0600	Turbine Bld-El.39	284	3
B0700	Service Bld. Cold Side	299	3**
B0800	Fuel Oil Storage Bld.	298	3
B0900	Diesel Generators	223	3
B1000	Aux. Boiler Rm.	373	2
B1100	Circ Water Pump House	319	3
B1200	Administration Bld.	432	3
B1300	WART Bld.	542	3
B1400	Information Center	313	3
B1500	Warehouse 2	208	3
B1600	Training Annex	144	3
B1700	Staff Bld.	374	3
B1800	Spare Generator Bld.	177	3
B1900	Bailey House	1595	3
B2000	Bailey Barn	245	3
B2100	Lube Oil Storage Rm.	134	3
B2200	Cold Machine Shop	221	3
B2300	Cable Vault	194	3
B2400	Staff Bld.-Turbine Tunnel	381	3
B2500	Relay House	257	3
B2600	Warehouse 5	212	3
D0400	Sewage Treatment Plant	1052	3
D0500	Circulating Water Piping	NA	NI*
D0600	Service Water System(Discharge)	348	3

Table 1A Survey Area Classification-Structures			
Package Number	**Survey Area-Structures**	**Sigma (dpm/100 cm²)**	**Classification**
D3400	LLWSB	1300	1

Table 1B Survey Area Classification-Land			
Package Number	**Survey Area-Land**	**Sigma (pCi/g)**	**Classification**
R0100	RCA Yard West	1.37	2,1**
R0200	Yard East	0.21	2,1
R0300	Roof and Yard Drains	***	3
R0400	Forebay	0.23	1
R0500	Bailey Point	0.29	3,1
R0600	Ball Field	0.02	3
R0700	Construction Debris Landfill	0.01	3
R0800	Admin and Parking Area	0.22	3
R0900	Balance of Plant Areas	0.80	3
R1000	Foxbird Island	0.45	3,1
R1100	Roof and Yard Drains	NA	NI*
R1200	LLWSB Yard	0.03	2
R1300	ISFSI	0.11	3,1
R1500	Ash Road Area	N/A	NI*
R1600	Area West of Bailey Cove	N/A	NI*
R1700	Area North of Ferry Road	N/A	NI*
R1800	Bailey House Land Area	0.21	3
R2000	Diffuser	***	3
R2100	Maintenance Yard	0.13	3
R2300	SFPI Substation Slab Area	0.23	1

*NI - Non Impacted
** Contains known sub-surface or sub-slab residual activity
*** To be determined upon opening the system

5.2.4 Changes in Classification

Area classification will be revised as necessary in accordance with the applicable procedure. However, no change in classification will be allowed once final status survey activities have begun in a given survey unit, or portion of the unit, until data analyses are completed. Re-classification will be from a lower classification to a higher classification only. If residual activity is found in a Class 3 survey unit, the unit or position of the unit may be reclassified as Class 2. If activity exists in a Class 2 survey unit above the DCGL, the unit, or portion of the unit, it may be reclassified Class1.

5.3 Establishing Survey Units

5.3.1 Survey Unit

Each plant area listed in Table 1 will be divided into discrete survey units. Survey units are contiguous areas which have similar characteristics and contamination levels. Survey units are assigned only one classification. The site and facility is surveyed, evaluated, and released on a survey unit basis.

d. Survey Unit Size

The identification of survey units is based on the area classification and the modeling assumptions used to develop the DCGLs. DG-4006 provided suggested sizes for survey units as indicated in Table 2 below. The rubblized structure survey unit size was based on a site-specific dose assessment described in Section 6. Scale drawings of building or land area are used to calculate the surface area of the walls, floors, ceilings or soil within the survey area. Floors and lower walls within a room or cubicle that fall within the recommended area for a given classification are taken as a survey unit, provided they have consistent radiological characteristics. Land areas that meet the area recommendations are taken as an individual survey unit, provided their radiological characteristics are similar. In the interest of promoting process efficiency, large areas will be combined into single survey units of similar characteristics and classification (e.g., the walls, ceiling and floor of the primary drain tank (PDT) cubicle) as allowed by NUREG-1575, section 4.6. Survey unit areas exceeding those listed in Table 2 will be justified on a risk-informed basis using the assumptions in the dose model. However, regardless of survey unit size, measurement densities (number of samples/m^2) will be maintained. Attachment A provides an example of this process.

The largest area that will administratively be allowed for sample density purposes for a standing building Class 1 survey unit will be 100 m^2 of floor space and up to two meters of the wall (or 180 m^2). Any survey unit larger than 180 m^2 would have the same sample density (samples/m^2) as that established for the 180 m^2 unit. Although, the NUREG-1575 method allows lower sample sizes, the minimum (for the sign test with an α and $\beta = 0.05$ and relative shift = 3 as presented in Section 5.2) sample density for this size survey unit is based on 14 samples. The corresponding sample density of 1 sample per 13 m^2 which will be the administrative minimum limit for sample density for any size Class 1 survey unit in a standing building. Class 2 and Class 3 survey units will be sampled in accordance with the MARSSIM guidance.

For the concrete fill scenario, the dose model reference volume for a survey unit is 1929 m^3. The surface area of a survey unit is based on the surface area/volume ratio of m^2 /m^3. For the Fuel building concrete, the building with the smallest surface area to volume ratio (1.6), the surface area of a survey unit would be 1929 m^3 x 1.6 m^2 /m^3 = 3086 m^2. (Factors for other concrete areas can be found in Table 8 of Section 6.2.5.) A minimum of 14 samples for this size survey unit, or 1 sample/220 m^2, will be the smallest required sample size. It will be administratively required that at least one sample per 220 m^2 be collected in Class 1 survey units for structures to be demolished. Class 1 and Class 2 areas will be sampled in accordance with the MARSSIM guidance.

For soil sampling, the minimum frequency in Class 1 areas will be 14/2000 m^2 or 1 sample per 143 m^2. The sample size in Class 2 and Class 3 areas will be in accordance with MARSSIM.

If techniques such as in-situ gamma spectroscopy are applied, it may be justified to take fewer than the minimum samples outlined above for concrete and soil. The acceptability of using a lower sample frequency would depend on the field-of-view covered by the in-situ detector and would be justified in a technical basis document.

Table 2 Survey Unit Areas			
Class	Suggested Survey Unit Area		
	Standing Structures	Rubblized Structures	Land
1	180 m^2	3086 m^2 *	2000 m^2
2	180 to 1000 m^2	3086 m^2 *	2000 to 10^4 m^2
3	No Limit	No Limit	No Limit

* May be larger but sample density will be maintained (1 sample/220 m^2. See LTP Section 6.2 and 6.4.

b. Site Reference Coordinate System (Reference Grid)

A reference coordinate system is used for impacted areas to facilitate the identification of survey units within the survey area. The reference coordinate system is basically an X,Y plot of the site area referenced to a fixed location or benchmark as shown in Figures 1 and 1A. Once the reference point is established, grids are overlaid parallel to lines of latitude and longitude.

5.4 Survey Design

This section describes the methods and data required to determine the number and location of measurements or samples in each survey unit, the coverage fraction for scan surveys, and requirements for measurements in background reference areas. The design activities described in this section will be documented in a survey package for each survey unit. Survey design includes the following:

- Scan Survey Coverage
- Sample Size Determination
- Background Reference Areas for WRS Test
- Reference Grid and Sample Location

Structures remaining after decommissioning (e.g., if the site is repowered) could require additional surveys such as removable surface activity surveys. The requirement for such surveys will be specified in the survey design package by the FSS specialist or engineer.

5.4.1 Scan Survey Coverage

The area covered by scan measurement is based on the survey unit classification as described in DG-4006 and as shown in Table 3 below. A 100% scan of Class 1 survey units will be required because they have the highest potential for elevated measurements. The emphasis will be placed on scanning the higher risk areas of Class 2 survey units such as soils, floors and lower walls. Scanning percentage of Class 3 survey units will be performed on likely areas of contamination based on the judgement of the FSS survey design specialist or FSS engineer.

Table 3
Scan Measurements

	Class 1	Class 2 *			Class 3
	Structures and Open Land	**Structures and Open Land DCGL Fraction**			**Structures and Open Land**
		% Scan	**Single Measurement**	**Avg.**	
Scan Coverage	100%	10%	0 - 0.49	<0.25	0 to 10%
		25%	0.50 - 0.74	0.25 - 0.50	
		50%	0.75 - 0.99	0.51 - 0.75	
		75%		>0.75 & <0.99	

* For Class 2 Survey Units, the amount of scan coverage will depend on the characterization data used to design the survey for the Survey Unit.

5.4.2 Sample Size Determination

The purpose of the survey is to demonstrate that residual radioactivity meets the release criteria and the null hypothesis is rejected. NUREG-1575 and DG-4006 describe the process for determining the number of survey measurements necessary to ensure a data set sufficient for a robust statistical analysis. Sample size is based on the relative shift, the Type I and II errors and the specific statistical test used to evaluate the data.

a. Determining Which Test Will Be Used

If the contaminant is not in the background or constitutes a small fraction of the DCGL, the Sign test is used. Based on the soil and concrete DCGLs determined for the MY site, the Sign Test will be used almost exclusively. The Sign Test may also be used for gross measurements where contamination is a part of background, by subtracting an average background measurement from

the gross sample results.

b. Establish Type I and II Errors

Section 2.7.2 of DG-4006 recommends using a Type I error rate of 0.05 and states that any value for the Type II error rate is acceptable. Following the DG-4006 guidance, the Type I error rate will be 0.05. A Type II error rate of 0.05 was selected based on site specific considerations. The Type II error rate may be modified as necessary.

c. Relative Shift

The relative shift (Δ /) is calculated. Delta (Δ) is equal to the $DCGL_w$ minus the Lower Boundary of the Gray Region (LBGR). Calculation of sigmas has been discussed in 5.2.3 and values are provided in Table 1 "Survey Area Sigmas- Structures/Land". The LBGR is normally set at 0.5 times the $DCGL_w$, but may be adjusted to obtain an optimal value for the relative shift.

Lower Boundary Of The Gray Region

The Lower Boundary of the Gray Region (LBGR) is the point at which the Type II (β) error applies. The default value of the LBGR is set initially at 0.5 times the DCGL. If the relative shift is greater than 3 for the Sign test or greater than 4 for the WRS test, then the number of data points, N, listed for the relative shift values of 3 or 4 from Table 5.5 or Table 5.3 in NUREG -1575, respectively will be used for the minimum number of samples. The minimum sample densities described in Section 5.3.1 will be maintained. The LBGR may be adjusted to reach the optimal value of 3 or 4 where reasonable.

Sigma

Sigma values (estimate of the standard deviation of the measured values in a survey unit) are initially calculated from characterization data and may be used for final status survey design. Where areas are remediated or changed, new sigma values may be calculated from at least 5 samples taken in the remediated or changed area to establish new sigmas for survey design purposes.

d. Sample Size

The sample size for a survey unit is based largely on the ratio of the DCGL to the sigma value for that area. In most cases at the MY site, because of the large difference between the DCGL and the sigma value, a minimum sample size of 14 per survey unit, based on the sign test, will be used. Smaller sample sizes may be used according to NUREG-1575. However, for Class 1 survey units an administrative minimum limit of one sample per 13 m^2 in standing buildings and a limit of 1 sample per 262 m^2 for rubblized concrete will be used. Using these sample density limits is a conservative approach as NUREG-1575 requires sample size according to the sigma to relative shift ratio, and not to survey unit area size.

There are some areas in containment, RCA, Fuel and Spray buildings that presently show large sigma values that would require more survey unit samples than the minimum. When these areas are decontaminated and remediated, their new sigma values are expected to result in sample sizes close to or equal to the minimum. With the exception of the minimum sample density, the sample size will be determined by the measured sigma. Tables 5.3 and 5.5 of NUREG-1575 will normally be used for sample size determination.

Only Type II decision errors are made in the gray area and making the gray region smaller increases the precision of the measurements. DG-4006 recommends that if the relative shift is greater than 3, the LBGR should be adjusted until the relative shift is equal to three. Use of a relative shift greater than 3 requires approval by the FSS Specialist or Engineer.

e. Wilcoxon Rank Sum (WRS) Test Sample Size

The number of data points, N, to be obtained from each reference area or survey unit for areas in which the contaminant is present in the background are determined using Table 5.3 in NUREG-1575. The table makes the recommended 20% adjustment to ensure adequate data.

f. Sign Test Sample Size

The number of data points for the survey unit is determined from Table 5.5 in NUREG-1575 for application of the Sign Test. This table includes an addition of 20% to the sample number. It should be noted that if the same values are used for the variables α , β and Δ / σ , the same number of measurements will be required regardless of the survey unit classification. This means that the same number of samples could be taken in a Class 1, Class 2, or Class 3 survey unit of various sizes.

g. Sample Size Adjustment for EMC

It is not anticipated that EMC sample size adjustments will be required at the MY site. However, the method for making such an adjustment is described below.

The length (L) of the survey unit sample spacing (grid) for square grids is determined as follows:

$$L = \sqrt{\frac{A_{su}}{N}}$$

Where A_{su} = the area of the survey unit, and

N = the required number of measurements.

The area bounding the elevated measurement location (A) is then determined as follows:

$$A = L^2$$

The Area Factor (AF) is then determined from the values in Tables 7 or 8 of Section 5.6.3. An acceptable upper limit concentration for the grid block area (A) is calculated by multiplying the area factor (AF) by the $DCGL_W$ as shown below.

$$DCGL_{EMC} = (AF)(DCGL_W)$$

The $DCGL_{EMC}$ is then compared to the $Scan_{MDC}$ for the survey instrument used. If the $Scan_{MDC}$ is greater than the acceptable concentration for the grid block area, then the grid spacing (L) must be reduced by increasing the number of measurement points. The new grid spacing is calculated by first determining the new area factor (AF_{NEW}):

$$AF_{NEW} = MDC_{scan} / DCGL_{EMC}$$

The new area factor is compared to Tables 7 or 8 values to determine the new grid block area (A_{NEW}). The new area is used to produce a new grid spacing (L_{NEW}) as follows:

$L_{new} = \sqrt{A_{new}}$

The revised required number of measurements is then determined by dividing the survey unit area (A_{su}) by the new grid block area (A_{new}).

5.4.3 Background Reference Areas for WRS Test

Background reference areas will not normally be used for site survey units, as background, being a small fraction of the DCGL, is not significant and the Sign Test will be applied to survey data. An area located several miles from the site having the same characteristics as the site was located and sampled in order to establish background radioactivity levels for soil. These locations will be used as a background reference area if the WRS test is applied to site survey measurements. A local school of similar vintage as the plant was used for building material background determinations (see Table 7 of Section 2.0).

5.4.4 Sample Grid and Sample Location

Direct measurements and volumetric samples are taken at randomly determined locations within the survey unit. For background reference areas, sample locations are determined using a randomly-started systematic pattern. Sample location is a function of the number of measurements required, the survey unit area, and the contaminant variability.

a. Sample Grid

The reference grid is primarily used for reference purposes and should be illustrated on sample maps. Physical sample and reference gridding will only be used when necessary. For most survey units, a sample start point will be identified and sample locations will be laid out in a square grid pattern at distance, L, from the start point in both the horizontal and vertical directions. An example is shown in Figures 1 and 1A. GPS instruments may be used in open land areas to determine reference or sample grid locations within the survey area. Digital cameras may be employed to provide a lasting record of survey location within the survey unit.

b. Measurement Locations

Measurement locations within the survey unit are clearly identified and documented for purposes of reproducibility. Actual measurement locations are

identified by tags, labels, flags, stakes, paint marks, geopositioning units or photographic record. An identification code matches a survey location to a particular survey unit.

Sample points for Class 1 and Class 2 survey units are positioned in a systematic pattern or grid throughout the survey unit by first randomly selecting a start point coordinate. A random number generator is used to determine the start point of the square grid pattern. The grid spacing, L, is a function of the area of the survey unit as shown below:

$$L = \sqrt{\frac{A}{n}} \qquad \text{for a square grid}$$

where:

A = the area of the survey unit,

n = the number of data points in the survey unit.

Sample points are located, L, distance from the random start point in both the X and Y directions.

Random measurement patterns are used for Class 3 survey units. Sample location coordinates are randomly picked using a random number generator.

Measurement locations selected using either a random selection process or a randomly-started systematic pattern that do not fall within the survey unit or that cannot be surveyed due to site conditions, including safety considerations, are replaced with other measurement locations using the random selection process.

5.5 Survey Methods and Instrumentation

5.5.1 Survey Measurement Methods

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable procedure. The techniques for performing survey measurements or collecting samples are specified in approved procedures. Final site survey measurements include surface scans, direct surface measurements, and gamma spectroscopy of volumetric materials. In situ measurements may also be used for concrete and soil radioactive concentration determinations..

On-site lab facilities are used for gamma spectroscopy, liquid scintillation and gas proportional counting in accordance with applicable procedures and Departmental Instructions. Off-site facilities are used for special, radiochemical analyses of hard-to-detect radionuclides. No matter which facilities are used, analytical methods will be administratively established to detect levels of radioactivity at 10% to 50% of the DCGL value.

a. Structures

Structures will receive scan surveys, direct measurements and, when necessary, surveys of volumetric building materials.

Scan Surveys

Scanning is performed in order to locate small areas of residual activity above the investigation level. Structures are scanned for beta-gamma emitting radionuclides using a thin window, gas-flow proportional detector. For special conditions, other appropriate beta-gamma detectors as indicated in Table 4 may be used. The measurement is made at a distance of 1 cm or less from the surface and at a scan speed of 5 cm/sec for hand-held instruments unless authorized by FSS Engineer. Thus, the detection level is a function of scan speed and surveyor efficiency as described in section 5.5.2. Scanning surveys may be effectively substituted by in situ gamma spectroscopy if technically justified.

Direct Measurements

Direct measurements are performed to detect surface activity levels. Direct measurements are conducted by placing the detector on or very near the surface to be counted and acquire data over a pre-determined count time. A count time of one minute is used for surface measurements and provides detection levels well below the DCGL. (The count time may be varied provided the required detection level is achieved and unnecessarily large uncertainties are avoided.)

Concrete With Contaminated or Activated Radionuclides

Residual radioactivity within activated and contaminated building materials has been measured volumetrically. More than 40 representative samples were obtained, sectioned and analyzed.

Activation activity has been determined to be relatively uniform in deposition within the concrete while surface beta-gamma contamination has been determined to be within an average depth of 1 mm. Detection sensitivities have been established by comparing surface measurements to radionuclide analyses such that reasonable approximations of sub-surface activity can be made using an indirect technique such as a surface contamination measurement.

Concrete structures will be surveyed by direct beta-gamma measurements in order to determine if they meet the radiological criteria for rubblization. In situ measurements may also be made.

b. Soils

Soils will receive scan surveys at the coverage level described in Table 3 and volumetric samples will be taken at designated locations. Volumetric samples will normally be taken at a depth of 0 to15 cm. Areas of sub-slab soil contamination will require sampling at depth exceeding 15 cm. The possibility of sub-surface contamination will be considered during the survey design process and the survey design package will contain requirements for sampling soil below 15 cm. Samples will be collected and prepared with existing station procedures. In Situ gamma spectrometry may also be used for soil measurements if technically justified.

Scans

Open land areas are scanned for gamma emitting nuclides. The gamma emitters are used as surrogates for the DCGLs for the radionuclide mixture actually expected. Sodium iodide detectors are used for scanning. The detector is held within a few centimeters of the ground surface and is moved in a pendulum pattern at a rate of 0.5 m/sec while the surveyor walks at a speed appropriate for the desired level of detection. The area covered by scan measurements is based on the survey unit classification as described in section 5.4.1.

Volumetric Samples

Soil materials are analyzed for gamma radioactivity. Soil samples of approximately 1500 grams are normally collected from the surface layer (top 15 cm). Sampling at depths below 15 cm will be performed using a split spoon sampler in areas determined to have soil

contamination below 15 cm. The areas around the RWST and Fuel Building are two of the areas requiring sub-surface sampling.

Sample preparation includes removing extraneous material, homogenizing, and drying for analysis. Separate containers are used for each sample and each container is tracked through the analysis process using a chain-of-custody record. Samples are split when required by the applicable FSS Quality Control procedure.

Sub-Slab Soils

Grade level foundation slabs will be removed during demolition which will afford the opportunity to obtain sub-slab soil samples. Those floor slabs remaining in place after demolition(@ elevations less than 3 feet below grade) will be sampled by coring the slab or by taking samples immediately adjacent to the slab using a split spoon or core sampler.

c. Systems

Systems identified during characterization with detectable radioactivity will be removed and disposed of as radioactive waste. An evaluation of the amount of embedded piping was conducted and determined that less than 1000 feet of possibly contaminated, embedded piping exist. The majority of embedded piping is located in structures above grade or within 3 feet of grade and will therefore be demolished with the buildings.

If the site is repowered, some contaminated systems may be kept. Those systems remaining will receive internal beta-gamma surveys in which detectors are inserted into the pipes or the internal pipe surfaces or components are removed for analysis. Accessible surfaces are surveyed in the same manner as other structure surfaces. Residual activity levels will be compared to the radiological criterion.

d. Special Areas and Conditions

Cracks, Crevices, Wall-Floor Interfaces and Small Holes

Surface contamination on irregular structure surfaces (e.g., cracks, crevices, and holes) are difficult to survey directly. Where no remediation has occurred and residual activity has not been detected above background, these surface blemishes may be assumed to have

the same level of residual activity as that found on adjacent surfaces. The accessible surfaces are surveyed in the same manner as other structural surfaces and no special corrections or adjustments have to be made.

Surface cracks and crevices are amenable to remediation using a needlegun to remove the material contained within the crack or crevice. In situations where remediation has taken place or where residual activity has been detected above background, a representative sample of the contamination within the crack or hole may be obtained or an adjustment for instrument efficiency may be employed. The accessible surfaces are surveyed in the same manner as other structure surfaces except that they are included in areas receiving judgmental scans when scanning is performed over less than 100% of the area.

Paint Covered Surfaces

Painted surfaces that have fixed contamination in place have been evaluated. Final status surveys will consider the effect of painted surfaces. No special consideration must be given to wall or ceiling areas painted before plant startup and which have not been subjected to repeated exposure to materials that would have penetrated the painted surface.

Pavement-Covered Areas

Parking lots, roads and other paved areas are treated the same as soil measurements. Scan and static gamma and beta-gamma surveys are made as determined by the survey unit design. Where remediation has occurred or expected residual activity could exceed background levels, direct surface measurements are taken and representative sub-surface samples are taken for analysis. Paved areas may be separate survey units or they may be incorporated into other, larger survey units.

Standing Structures

The site structures are slated for demolition. However, if the site is repowered or there is a need to retain a building, it will be surveyed prior to release. Standing structures will receive scan and direct surveys.

Backgound Areas and Materials

Background surveys of buildings materials and soils were taken during site characterization. While background levels were determined to be a very small fraction of the DCGLs and they can be considered to be part of the sample activity, the actual values are available if background correction of survey results are needed.

5.5.2 Instrumentation

Radiation detection and measurement instrumentation for the final status survey is selected to provide both reliable operation and adequate sensitivity to detect the radionuclides identified at the site at levels sufficiently below the DCGL. Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field. The instrumentation will, to the extent practicable, use data logging with bar code scanning capability.

Commercially available portable and laboratory instruments and detectors are used to perform the three basic survey measurements: 1) surface scanning, 2) direct surface contamination measurements, and 3) spectroscopy of soil and other bulk materials. The Instrumentation Program Procedure controls the issuance, use, and calibration of instrumentation. If new survey technology or specialized instruments become available, they will be specified, calibrated and used in accordance with applicable procedure. Records supporting the instrumentation program are maintained by Document Control.

a. Selection

Radiation detection and measurement instrumentation is selected based on the type and quantity of radiation to be measured. (The instruments used for direct measurements are capable of detecting the radiation of concern to a Mimimum Detectable Concentration (MDC) of between 10% and 50% of the applicable DCGL. The use of 10% to 50% of the DCGL is an administrative limit only. Any value below the DCGL is acceptable in Class 1 or 2 survey units.) MDCs of less than 50% of the DCGL allow detection of residual activity in Class 3 survey units at an investigation level of 0.5 times the DCGL. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at the $DCGL_{EMC}$ however, the instruments used are capable of detecting the radiation of concern to an MDC below the DCGL. Instrument MDCs are discussed in section 5.5.2 and are listed in Table 5. Instrumentation used for final status survey is listed in Table 4.

b. Calibration And Maintenance

Instruments and detectors are calibrated for the radiation types and energies of interest at the site. The calibration sources for beta survey instruments are Tc-99 or Co-60 because the average beta energy (100 keV) approximates the beta energy of the radionuclides found on surfaces or in systems on site. The alpha calibration source is Am-241 which has an appropriate alpha energy for plant-specific alpha emitting nuclides . Gamma scintillation detectors are

Table 4
Final Status Survey Instruments

Measurement Type	Detector Type	Detector Area/Density	Typical Manufacturer & Model ##	Units
Surface Alpha/Beta-Gamma	Gas Proportional	100 cm^2 1 mg/ cm^2	Eberline SHP-100C	cpm
Surface Alpha/Beta-Gamma	Gas Flow Proportional	126 cm^2 0.8 mg/ cm^2	Ludlum 43-68	cpm
Surface Beta -Gamma	G-M	15.5 cm^2 2mg/ cm^2	LND, TGM Eberline SHP-360	cpm
Gamma Scan	NaI(Tl)	2"x2"	Eberline SPA-3	cpm
Gamma Exposure Rate	Pressurized Ion Chamber	8 L Sphere	Reuter-Stokes RSS-112	uR/hr
Gamma Exposure Rate	Compensated G-M	N/A	Eberline SHP-300A	uR/hr
Liquid Beta	Scintillation	N/A	Beckman	uCi
Smear Beta-Gamma	Gas Proportional	15.5 cm^2 0.8 mg/ cm^2	Tennelec	dpm
Gamma Spectroscopy	High Purity Ge	N/A	Canberra	Geometry Dependent

calibrated using Cs-137 but the energy response to Co-60 has also been determined since discrete areas of Co-60 contamination have been found by soil surface scans.

Instrumentation used for final status survey will be calibrated and maintained in accordance with the Instrumentation Program procedure. Radioactive sources

used for calibration are traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. If vendor services are used, these services will be obtained in accordance with the Field Procurement procedure, to ensure the same level of quality.

c. Response Checks

Instrumentation response checks are conducted to assure proper instrument response and operation. An acceptable response is an instrument reading of $\pm$ 20% of the established check source value or a reading within $\pm$ 3 sigma of the established check source value when performance charts are used. Response checks are performed daily before instrument use and again at the end of use. Check sources use the same type of radiation as that being measured in the field and are held in fixed geometry jigs for reproducibility. If an instrument fails a response check, it is labeled "Do Not Use" and is removed from service until the problem is corrected in accordance with applicable procedures. Measurements made between the last acceptable check and the failed check are evaluated to determine if they should remain in the data set.

d. Minimum Detectable Concentration (MDC)

The MDC is determined for the instruments and techniques used for final status surveys (Table 5). The MDC is the concentration of radioactivity that an instrument can be expected to detect 95 percent of the time.

<u>Beta-Gamma Scan MDC For Building and Structure Surfaces</u>

The scan MDC (or MDC_{scan}) for structure surfaces for beta-gamma emitting radionuclides is determined from the method given in section 2.5.1 of DG-4006 using the following equation:

$$MDC_{scan} = \frac{(60)(1.38)\sqrt{B}}{\sqrt{p}E_i E_s (A/100)t}$$

where:

MDC_{scan} = minimum detectable concentration for scanning surfaces (dpm/100 cm^2),

60 = conversion factor (sec/min),

1.38 = index of sensitivity, from draft Reg Guide DG-4006,

B = number of background counts in time interval t (counts),

p = surveyor efficiency, 0.5 from draft Reg Guide DG-4006,

E_i = instrument efficiency for emitted radiation (cpm/dpm),

E_s = source efficiency (emissions/disintegration),

A = area of detector (cm^2),

t = time interval of observation (sec).

The index of sensitivity value of 1.38 equals the 95% detection of a concentration equal to the MDC_{scan} with a 60% false positive rate as referenced in DG-4006. The value of p is the mean value for normal field conditions as discussed in NUREG-1507.

The *a priori* MDCs reported in Table 5 use nominal background values. When the *a posteriori* MDCs are reported for FSS surveys, the actual field background is used to calculate the MDC.

The value of E_s is dependent on the material type. Corrections for radionuclide absorption have been made to calculate an E_s for a gas proportional detector on contaminated concrete.

The value t is the actual time that the detector is able to respond to the source. The response time depends on scan speed, detector geometry and the area of residual radioactivity.

Gamma Scan MDC For Land Areas

The MDC_{scan} value for the sodium iodide detector is determined according to the calculation method used for Table 6.7 of NUREG-1575. The table gives a scan MDC value of 3 to 6 pCi/g depending on whether the nuclide of concern is Co-60 or Cs-137. Cs-137 has been established as the surrogate nuclide for volumetric sample surveys with hard-to-detect nuclides scaled to cesium. The

MDC value of 5 pCi/g for Cs-137, as calculated using the methods of paragraph 6.7.2 in MARSSIM, is less than the Cs-137 DCGL of 10.4 pCi/g. and, with the lower MDC for Co-60, assurance is provided that any discrete areas of cobalt contamination will be detected as well.

Static MDC For Structure Surfaces

For static (direct) surface measurements, the MDC is calculated as follows using the general formula given in section 2.5.1 of DG-4006:

$$MDC_{static} = \frac{3 + 4.65\sqrt{B}}{(K)(t)}$$

where:

MDC_{static} = minimum detectable concentration for direct counting (dpm/100 cm^2),

B = background counts during the count interval t (counts),

t = count interval (for paired observations of sample and blank, usually 1 minute),

K = calibration constant (counts/min per dpm/100 cm^2),

The value of K includes correction factors for efficiency (E_i and E_s). The value of Es is dependent on the material type. Corrections for radionuclide absorption have been made.

e. Detection Sensitivity

The nominal detection sensitivity of the detectors used for surface contamination surveys has been determined and is provided in Table 5.

Count times are instrument-specific and are selected to ensure that the measurements are sufficiently sensitive for the DCGL. For example, the count times associated with surface activity surveys (1 minute) and gamma spectroscopy of volumetric materials (17 minutes) are administratively established to achieve MDCs no greater than 50% of the DCGL. The MDC_{scan} values are well below the DCGL.

f. Total Efficiency and Source Efficiency (E_s) for Concrete Radioactive Contamination

A source efficiency (E_s) was calculated for the Ludlum 43-68 gas proportional detector on contaminated concrete. This was accomplished by taking surface contamination measurements on 34 contaminated concrete core samples and then gamma analyzing the core samples to determine the concentration of the
samples for beta-gamma radioactivity. The only non-gamma beta emitter found in radio chemical analysis of concrete core samples was Sr-90 but, this contributed only 0.2 % of the total betas and was not considered to be significant in the calculations. The net counts per minute for the detector were compared to the beta-gamma concentration of the sample to determine a total efficiency (E_t) for each sample. These results were averaged to ascertain an overall total efficiency (E_t). The published NUREG-1507 instrument efficiency (E_i) for this type detector was then used to calculate the source efficiency (E_s). The E_i value was based on the Tc-99 beta energy which gives a more conservative (lower) efficiency than that for the higher energy beta emission of Co-60, Cs-137 and Cs-134, the main beta constituents of the contamination mixture.
The calculation used (from NUREG-1507) for this determination is:

$$E_s = E_t / E_i$$

Using this calculation results in the following source efficiency for a 43-68 detector on the concrete contamination mixture:

Source Efficiency (E_s)	Total Efficiency (E_t)	Instrument Efficiency (E_i)
0.781	0.148	0.190

Table 5 Measurement Detection Sensitivities					
Type of Measurement	**Detector**	**Background ***	**Eff. (c/d)**	**MDC**	**DCGL**
Beta-Gamma Surface Scan	Pancake G-M (SHP-360)	40 cpm	0.10	6600 dpm/100 cm^2	15200 dpm/ 100 cm^2
Beta-Gamma Surface Scan	Ludlum 43-68 126 cm^2 Gas Proportional	400 cpm	0.15	1220 dpm/100 cm^2	15200 dpm/ 100 cm^2
Beta-Gamma Direct	Pancake G-M (SHP-360)	40 cpm	0.10	2100 dpm/100 cm^2	15200 dpm/ 100 cm^2
Beta-Gamma Direct	Ludlum 43-68 126 cm^2 Gas Proportional	400 cpm	0.15	510 dpm/100 cm^2	15200 dpm/ 100 cm^2
Alpha Direct	Ludlum 43-68 126 cm^2 Gas Proportional	1 cpm	0.20	30 dpm/100 cm^2	Surrogate of 15200 dpm/ 100 cm^2
Gamma Scan	NaI(Tl) (SPA-3)	10,000 cpm	0.012	1500 cpm or 5 pCi/g (Cs-137)	10.4 pCi/g (Cs equiv.)
Gamma Spectroscopy	HP Ge	N/A	N/A	0.01 pCi/g	10.4 pCi/g (Cs equiv.)
Gamma Exposure Rate	Pressurized Ion Chamber	N/A	N/A	1-2 uR/hr	N/A
Liquid Beta	Beckman Liquid Scintillation	1.7 E-3 dpm	0.35	1E-5 uCi/ml	N/A
Smear Beta-Gamma	Tennelec Gas Proportional	0.5 cpm Alpha 30 cpm Beta	0.25 Alpha 0.35 Beta	25 dpm - alpha 81 dpm - beta-gamma	N/A
Gamma Exposure Rate	Compensated G-M (SHP-300)	10 uR/hr	1.22	14.7 uR/hr	N/A

*Background values are typical values. These background values are well below the MDCs and are adequate for selecting the instruments for performing surveys.

5.6 Investigation Levels and Elevated Areas Test

During survey unit measurements, levels of radioactivity may be identified that warrant investigation. Depending on the results of the investigation, the survey unit may require no action, may require remediation, and/or may require reclassification and resurvey. Investigation levels and the investigation process are described below.

5.6.1 Investigation Levels

DG-4006 (Table 2.2) provides investigation levels for scan surveys. In addition to investigation levels for scan surveys, direct measurement survey investigation levels have also been developed. These additional investigation levels include a very conservative value for Class 3 survey units as shown in Table 6.

5.6.2 Investigation Process

During the survey process, locations with residual activity exceeding the investigation level are marked for further investigation. The elevated survey measurement is verified. The size and average activity level in the elevated area is determined to demonstrate compliance with the area factors. If any location in a Class 2 area exceeds the DCGL,

Table 6
Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	$>DCGL_{EMC}$	$>DCGL_{EMC}$
Class 2	$>DCGL_w$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_w$.	$>DCGL_w$
Class 3	$>DCGL_w$ or $>MDC_{scan}$ if MDC_{scan} is greater than $DCGL_w$.	$>0.5\ DCGL_w$

scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s). If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area should be reclassified.

The results of the investigation process are reported in the final status survey report.

5.6.3 Elevated Measurement Comparison (EMC)

The elevated measurement comparison is used for Class 1 survey units when one or more scan or static measurements exceed the investigation level. The EMC provides

assurance that unusually large measurements receive the proper attention and that any area having the potential for significant dose contribution is identified. As stated in NUREG-1575, the EMC is intended to flag potential failures in the remediation process and should not be considered the primary means to identify whether or not a survey unit meets the release criterion.

Locations identified by scan with levels of residual radioactivity which exceed the $DCGL_{EMC}$ or static measurements with levels of residual radioactivity which exceed the $DCGL_{EMC}$ are subject to additional surveys to determine compliance with the elevated measurement criteria. The size of the area containing the elevated residual radioactivity and the average level of residual activity within the area are determined. The average level of activity is compared to the $DCGL_W$ based on the actual area of elevated activity. (If a background reference area is being applied to the survey unit, the mean of the background reference area activity will be subtracted before conducting the EMC).

The initial $DCGL_{EMC}$ is established during the survey design and is calculated as follows:

$$DCGL_{EMC} = \text{Area Factor} \times DCGL$$

The area factor is the multiple of the DCGL that is permitted in the area of elevated residual radioactivity without remediation. The area factor is related to the size of the area over which the elevated activity is distributed. That area is generally bordered by levels of residual radioactivity below the DCGL and is determined by the investigation process. Area factors were calculated in Section 6 of the LTP and are provided in Table 7A for standing buildings, 7B for concrete rubble fill and Table 8 for Soils.

Table 7A
Area Factors for Standing Buildings

Area (m^2)	1	2	4	8	12	16	20	24	36	50	100
Area Factor	12.6	7.1	4.2	2.8	2.2	1.9	1.7	1.6	1.4	1.2	1.0

Table 7B
Area Factors For Concrete Rubble Fill

Area (m^2)	0.5	1	2	4	6	8	12	16	20	30	60	80	100
Area Factor	52	28	16	10	9	7	7	6	6	6	6	6	6

Table 8 Area Factors For Soil											
Area (m^2)	1.0	2.0	6.0	10	20	60	100	300	600	1000	2000
Area Factor	15.6	8.7	4.1	3.0	2.2	1.5	1.4	1.1	1.1	1.0	1.0

The actual area of elevated activity is determined by investigation surveys and the area factor is adjusted for the actual area of elevated activity. The product of the adjusted area factor and the $DCGL_W$ determines the actual $DCGL_{EMC}$.

The results of the EMC tests in a given survey unit are evaluated using the equation below. If more than one elevated area is identified in a given survey unit, the unity rule can be used to determine compliance. If the formula value is less than unity, no further elevated area testing is required and the EMC test is satisfied.

$$\frac{\delta}{DCGL_w} + \frac{(\text{average concentration in elevated area} - \delta}{(\text{Area Factor})(DCGL_w)} < 1$$

Where,

δ = the average residual activity in the survey unit.

If the $DCGL_{EMC}$ is exceeded, the area is remediated and resurveyed.

5.6.4 Remediation and Reclassification

As shown in Table 9 areas of elevated residual activity above the $DCGL_{EMC}$ are remediated to reduce the residual radioactivity to acceptable levels. Based on survey data, it may be necessary to remediate an entire survey unit or only a portion of it.

If the average value of survey measurements in a Class 2 or 3 survey unit exceed the DCGL, the survey unit or a portion of it may be reclassified. If an individual survey measurement in a Class 3 survey unit exceeds 0.5 DCGL, the survey unit, or portion of a survey unit , will be evaluated, and if necessary, reclassified to a Class 2.

Table 9 Investigation Actions			
Action If Investigation Results Exceed:			
Class	**$DCGL_{EMC}$**	**DCGL**	**0.5 DCGL**
1	Remediate and Resurvey	Acceptable	Acceptable
2	Remediate	Reclassify portions as necessary	Acceptable
3	Remediate	Increase scan coverage and reclassify	Increase scan coverage and reclassify

5.6.5 Resurvey

Following an investigation, if a survey unit is reclassified or if remediation activities were performed, a resurvey is performed in accordance with procedures. If a Class 2 area had contamination greater than the DCGL it may require reclassification. If the area is reclassified as Class 1, the $Scan_{MDC}$ was sensitive enough to detect the $DCGL_{EMC}$ and, there were no areas greater than the $DCGL_{EMC}$, a 100% scan is performed. If the Class 2 area had contamination greater than the DCGL but the $Scan_{MDC}$ was not sensitive enough to detect the $DCGL_{EMC}$, the affected area is reclassified and resurveyed at the sample density determined from the EMC.

If a Class 1 area shows small areas with activity greater than the $DCGL_w$ by a factor larger than its area factor, the area factor and sample size may be adjusted, or the affected area will be remediated and resurveyed. If the Class 1 area average activity exceeds the DCGL over a large area, the entire area will be remediated and resurveyed.

5.7 Data Collection and Processing

5.7.1 Sample Handling and Record Keeping

A sample tracking record (chain-of-custody record) accompanies each sample from the point of collection through obtaining the final results to ensure the validity of the sample data. Sample tracking records are controlled and maintained and, upon completion of the data cycle, are transferred to Document Control, in accordance with applicable procedures.

Each survey unit has a document package associated with it which covers the design and field implementation of the survey requirements. Survey unit records are quality records and are transferred to Document Control.

5.7.2 Data Management

Survey data are collected from several sources during the data life cycle and are evaluated.

QC replicate measurements are not used as final status survey data.

Measurements performed during turnover and investigation surveys can be used as final status survey data if they were performed according to the same requirements as the final survey data; no remediation was performed after the measurements were taken; and isolation measures were applied to the survey unit to prevent recontamination.

Measurement results stored as final status survey data constitute the final survey of record and are included in the data set for each survey unit used for determining compliance with the site release criteria.

Measurements are recorded in units appropriate for comparison to the DCGL. The recording units for surface contamination are dpm/100 cm^2 and pCi/g for activity concentrations. Numerical values, even negative numbers, are recorded.

Document Control Procedures establish requirements for record keeping. Measurement records include the surveyor's name, the location of the measurement, the instrument used, measurement results, the date and time of the measurement and any surveyor comments.

5.7.3 Data Verification and Validation

The final status survey data are reviewed before data assessment to ensure that they are complete, fully documented and technically acceptable. The review criteria for data acceptability will include at a minimum, the following items:

- The instrumentation MDC was below the $DCGL_{EMC}$ for Class 1, below the $DCGL_W$ for Class 2 survey units and below 0.5 DCGL for Class 3 survey units.

- The instrument calibration was current and traceable to NIST standards.

- The instruments were source checked before and after use each day data were collected.

- The MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey.

- The survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed.

- "Special methods" for data collection were properly applied for the survey unit under review.

- The chain-of-custody was tracked from the point of sample collection to the point of obtaining results.

- The data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility.

- The data have been properly recorded as required by the survey unit package.

If the data review criteria were not met, the discrepancy will be reviewed and the decision to accept or reject the data will be documented in accordance with approved procedures.

5.7.4 Graphical Data Review

Survey data may be graphed to identify patterns, relationships or possible anomalies which might not be so apparent using other methods of review. A posting plot or a frequency plot may be made. Other special graphical representations of the data will be made as the need dictates.

a. Posting Plots

Posting plots may be used to identify spatial patterns in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal patches of elevated radioactivity or local areas in which the DCGL is exceeded. Posting plots can be generated for background reference areas to point out spatial trends that might adversely affect the use of the data. Incongruities in the background data may be the result of residual, undetected activity, or they may just reflect background variability.

b. Frequency Plots

Frequency plots may be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity. Multiple peaks in the data may indicate the presence of isolated areas of residual radioactivity or background variability due to soil types or differing materials of construction. Variability may also indicate the need to more carefully match background reference areas to survey units or to subdivide the survey unit by material or soil type.

5.8 Data Assessment and Compliance

An assessment is performed on the final status survey data to ensure that they are adequate to support the determination to release the survey unit. Simple assessment methods such as comparing the survey data to the DCGL or comparing the mean value to the DCGL are first performed. The statistical tests are then applied to the final data set and conclusions are made as to whether the survey unit meets the site release criterion.

5.8.1 Data Assessment Including Statistical Analysis

The results of the survey measurements are evaluated to determine whether the survey unit meets the release criterion. In some cases, the determination can be made without performing complex, statistical analyses.

a. Interpretation of Sample Measurement Results

An assessment of the measurement results is used to quickly determine whether the survey unit passes or fails the release criterion or whether one of the statistical analyses must be performed. The evaluation matrices are presented in the following two tables.

Table 10
Interpretation of Sample Measurements When A Background Reference Area Is Used

Measurement Results	Conclusion
Difference between maximum survey unit concentration and minimum reference area concentration is less than DCGLw	Survey unit meets release criterion.
Difference of survey unit average concentration and reference average concentrations greater than DCGLw	Survey unit fails.
Difference between any survey unit concentration and any reference area concentration is greater than DCGLw and the difference of survey unit average concentration and reference area average concentration is less than DCGLw	Conduct WRS test and elevated measurements test.

Table 11
Interpretation of Sample Measurements When No Reference Area is Used

Measurement Results	Conclusion
All concentrations less than DCGLw	Survey unit meets release criterion
Average concentration greater than DCGLw	Survey unit fails
Any concentration greater than DCGLw and average concentration less than DCGLw	Conduct Sign Test and elevated measurements test.

When required, one of two statistical analyses is performed on the survey data:

1. the Wilcoxon Rank Sum Test (WRS) or
2. the Sign Test, in accordance with DG-4006.

In addition, survey data are evaluated against the EMC criteria as previously described in section 5.6.3 and as required by DG-4006 (section 2.9). The statistical test is based on the null hypothesis (Ho) that the residual radioactivity in the survey unit exceeds the DCGL. There must be sufficient survey data at or below the DCGL to reject the null hypothesis and conclude the survey unit meets the site release criterion for dose. Statistical analyses are performed using a specially designed software package or, if necessary, using hand calculations.

b. Wilcoxon Rank Sum Test

The WRS test may be used when the radionuclide of concern is present in the background or measurements are used that are not radionuclide-specific. Because gross activity data are not radionuclide-specific, the data need to consider background contribution in each measurement. In most cases, the

actual net data will be used. However, if less than MDC values are reported, they will be treated as described in Section 8.4.2 of NUREG-1575.

The WRS test is applied as follows:

1. The background reference area measurements are adjusted by adding the DCGL to each background reference area measurement, $X_i, (X_i + DCGL)$.

2. The number of adjusted background reference area measurements, m, and the number of survey unit measurements, n, are summed to obtain N, (N = m + n).

3. The measurements are pooled and ranked in order of increasing size from 1 to N. If several measurements have the same value, they are assigned the average rank of that group of measurements.

4. The ranks of the adjusted background reference area measurements are summed to obtain W_r.

5. The value of W_r is compared with the critical value Table I.4 of NUREG-1575. If W_r is greater than or equal to the critical value, the survey unit meets the site release dose criterion. If W_r is less than the critical value, the survey unit fails to meet the criterion.

c. Sign Test

The Sign test is a one-sample statistical test used for situations in which the radionuclide of concern is not present in background, except in very small fractions compared to the DCGL, and the measurements are radionuclide-specific. For those situations in which the radionuclide is present in small amounts in background, the entire amount is considered to be from plant activities and the total concentration is compared to the DCGL. This option is only used when it can be reasonably expected that ignoring the background concentration will not affect the outcome of the Sign test. The advantage for doing this is that a background reference area is not needed.

The Sign test is conducted as follows:

1. The survey unit measurements, X_i, i = 1, 2, 3, ...n; where n = the number of measurements, are listed.

2. X_i is subtracted from the DCGL to obtain the difference (DCGL - X_i , i = 1, 2, 3,...,n).

3. Differences where the value is exactly zero are discarded and n is reduced by the number of such zero measurements.

4. The number of positive differences are counted. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the DCGL and contributes evidence that the survey unit meets the site release criterion.

5. The value of S+ is compared to the critical value given in Table I.3 of NUREG-1575. The table contains critical values for given values of N and α . The value of α is set at 0.05 during survey design. If S+ is greater than or equal to the critical value given in the table, the survey unit meets the site release criterion. If S+ is less than the critical value, the survey unit fails to meet the release criterion.

5.8.2 Data Conclusions

The results of the statistical tests including application of the EMC allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release dose criterion. The data provide statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the release criterion. The decision to release the survey unit is made with sufficient confidence and without further analysis.

The second conclusion that can be made is that the survey unit fails to meet the release criterion. The data are not conclusive in showing that the residual radioactivity is less than the release criterion. The data are analyzed further to determine the reason for the failure.

Possible reasons are that:

1. the average residual radioactivity exceeds the DCGL, or

2. the test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

The power of the statistical test is a function of the number of measurements made and the standard deviation in measurement data. The power is determined from 1- β where β is the value for Type II errors. A retrospective power analysis may be performed using the methods described in Appendices I.9 and I.10 of NUREG-1575. If the power of the test is insufficient due to the number of measurements, additional samples may be collected as directed by procedure. A greater number of measurements increases the probability of passing if the survey unit actually meets the release criterion. If failure was due to the presence of residual radioactivity in excess of the release criterion, the survey unit must be remediated and resurveyed.

5.8.3 Compliance

The final status survey is designed to demonstrate that licensed radioactive materials have been removed from MY station facilities and property to the extent that residual levels of radioactive contamination are below the radiological criteria for unrestricted use as approved by the NRC. The site-specific radiological criteria presented in this plan demonstrate compliance with the criteria of 10CFR20.1402.

If the measurement results pass the requirements of Table 10 of section 5.8.1, and the elevated areas evaluated per section 5.6.3 pass the elevated measurement comparison, then the survey unit is suitable for unrestricted release.

5.9 Reporting Format

Survey results are documented in history files, survey unit release records, and in the final status survey report. Other reports may be generated as requested by the NRC.

5.9.1 History File

A history file of relevant operational and decommissioning data is compiled in accordance with QA Records Management requirements. The purpose of the history file is to provide a substantive basis for the survey unit classification, and hence, the level of intensity of the final status survey. The history file contains:

a. Operating history which could affect radiological status

b. Summarized scoping and site characterization data

c. Other relevant information

5.9.2 Survey Unit Release Record

A separate release record is prepared for each survey unit. The survey unit release record is a stand-alone document containing the information necessary to demonstrate compliance with the site release criteria. This record includes:

- Description of the survey unit
- Survey unit design information
- Survey unit measurement locations and corresponding data
- Survey unit investigations performed and their results
- Survey unit data assessment results

When a survey unit release record is given final approval, it becomes a quality record and is turned over to Document Control.

5.9.3 Final Status Survey Report

Upon completion of each of the 3 major final status survey phases, an interim report will be prepared for review by the NRC. Upon completion of the final status survey phases, a final report will be prepared.

The final status survey report provides a summary of the survey results and the overall conclusions which demonstrate that the MY facility and site meet the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results are included in the report. The level of detail is sufficient to clearly describe the final status survey program and to certify the results. The format of the final report will contain the following topics:

1.0 Overview of the Results

2.0 Discussion of Changes to FSS

3.0 Final Status Survey Methodology

- Survey unit sample size
- Justification for sample size

4.0 Final Status Survey Results
- Number of measurements taken
- Survey maps
- Sample concentrations
- Statistical evaluations
- Judgmental and miscellaneous data sets

5.0 Anomalous Data

6.0 Conclusion for each survey unit

7.0 Any Changes from initial assumptions on extent of residential activity.

5.9.4 Other Reports

Other reports will be prepared and submitted as necessary.

5.10 FSS Quality Assurance Plan (QAP)

The final status survey plan will be developed and implemented by the DOC personnel. The FSS Plan will ensure that the site will be surveyed, evaluated and determined to be acceptable for unrestricted use if the residual activity results in an annual TEDE to the average member of the critical group of 25 mrem or less. Ensuring that the site meets the requirements for license termination is a complex process. Quality must be built in to each phase of the plan and measures must be taken during the execution of the plan to determine whether the expected level of quality is being achieved.

The Quality Assurance activities for decommissioning are based on the requirements of 10CFR50.82. The objective of the FSS QA Plan is to ensure that the survey data collected are of the type and quality needed to demonstrate with sufficient confidence that the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. The plan ensures that: 1) personnel responsibilities and interfaces are delineated, 2) the elements of the final status survey plan are implemented in accordance with the approved procedures, 3) surveys are conducted by trained personnel using calibrated instrumentation, 4) the quality of the data collected is adequate, and 5) corrective actions, when identified, are implemented in a timely manner and are determined to be effective. The FSS QA Plan will be applied to the following aspects of final status survey

activities.

5.10.1 Project Management and Organization

The DOC Project Manager is responsible for ensuring overall quality of decommissioning activities, which includes the responsibility for conducting audits and surveillances of those activities, including FSS.

The Radiation Protection Manager is responsible for overall integration of the final status survey project with other decommissioning activities. The Radiological Engineering Section Head is responsible for the survey design, implementation, and program assessment activities. The Radiological Engineering Section Head is also responsible for approval of the final survey results and determining if the survey units can be released. The FSS Engineers and Specialists are responsible for preparing the survey packages, supervising the survey teams and analyzing the survey data.

MY and the DOC provide oversite of the FSS process.

a. Project Description and Schedule

Each area of the site will be divided into survey units and classified as directed by procedure. The survey measurements for each survey unit will be determined during the survey design phase. Portions of the final status survey will be performed during deconstruction activities as areas become available for survey. The non-impacted areas may be evaluated for release prior to significant decommissioning activities taking place.

The survey process will begin at the north end of Bailey Point (at the Ferry Road) and work south. The least impacted areas will be surveyed first followed by the more impacted areas. Structure surveys will begin with the Turbine Hall and progress to the buildings within the RA.

Upon completion of each major phase, records will be compiled and reviewed in order to determine the status of each survey unit. A report will be made and submitted to the NRC which summarizes the survey area status.

b. Quality Objectives and Measurement Criteria

Type I errors will be established at 0.05 unless authorized by the NRC. Type II errors will be set at 0.05.

c. Training and Qualification

Personnel performing final status survey measurements will be trained and qualified . Training will include the following topics:

- Procedures governing the conduct of the final status survey,
- Operation of field and laboratory instrumentation used in the final status survey, and
- Collection of final status survey measurements and samples.

The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity. Records of training will be maintained in accordance with the approved procedure for Initial and Continuing Training Programs for Decommissioning.

d. Survey Documentation

Each final status survey measurement will be identified by date, instrument, location, type of measurement, and mode of operation. The technician taking the measurement will also be identified. Generation, handling and storage of the original final status survey design and data packages will be controlled. The FSS records have been designated as quality documents and, as such, they will be maintained as such. Upon completion of FSS, the records will be turned over to MY.

5.10.2 Measurement/Data Acquisition

a. Survey Design and Sampling Methods

The site will be divided into survey units. Each survey unit package will specify the type and number of measurements required based on the classification of the survey unit.

b. Written Procedures

Sampling and survey tasks must be performed properly and consistently in order to assure the quality of the final status survey results. The measurements will be performed in accordance with approved, written procedures.

Approved procedures describe the methods and techniques used for the final status survey measurements.

c. Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the final survey results is established by procedure. When custody is transferred, a chain of custody form will accompany the sample for tracking purposes. Samples will be stored for the duration of the project.

d. Quality Control Surveys

Procedures establish built-in Quality Control checks in the survey process for both field and laboratory measurements. For structures and systems, QC replicate scan measurements will consist of resurveys of 5% of randomly selected survey area packages by a different technician with results compared to the original measurement. The acceptance criterion shall be that the same conclusion as the original survey was reached based on the repeat scan.

Quality Control for direct surface contamination and/or exposure rate measurements will consist of the number of repeat measurements specified by Table 4.3 of NUREG-1575 taken by a different technician and the results compared to the original measurements. The acceptance criterion for direct measurements is that the true variance in the measurements is within a factor of two of the estimated variance with a confidence interval of 95%.

For soil, water and sediment samples, Quality Control will consist of taking blind duplicate samples or third party analyses in the number specified in Table 4.3 of NUREG-1575 from each survey unit as specified in the survey unit package. The acceptance criterion for blank samples is that no plant-derived radionuclides are detected. The criteria for blind duplicates is that the two measurements are within 15% of the mean of both measurements and the accuracy is within the value specified by procedure. For third party analyses, the acceptance criteria are the same as those for blind duplicates. Some sample media, such as asphalt or rubble, will not be subjected to split or blind duplicate analyses due to the lack of homogeneity. These samples will simply be recounted to determine if the two counts are within 20% of each other.

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be required by procedure and may result in a resurvey or resampling.

e. Instrumentation Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as specified in Section 5.5.2 as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the Instrumentation Program procedure.

f. Control of Consumables

In order to ensure the quality of data obtained from FSS surveys and samples, new sample containers will be used for each sample taken. Tools used to collect samples will be verified to be free of contamination prior to taking samples.

g. Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, will be procured from appropriate vendors in approved quality and procurement procedures.

h. Data Base Control

Software used for data reduction, storage or evaluation will be fully documented and certified by the vendor. The software will be tested prior to using an appropriate test data set.

i. Data Management

Survey data control from the time of collection through evaluation is specified by procedure. Manual data entries will be second verified.

5.10.3 Assessment and Oversight

a. Assessments

FSS self assessments will be conducted in accordance with approved procedures and the findings will be submitted to the DOC Site Quality for inclusion in the trend data base.

b. Oversight

MY and the DOC will provide oversight of FSS activities through audits and surveillances, technical reviews and observation of field activities.

c. Independent Review of Survey Results

Randomly selected survey packages (approximately 5%) from survey units will be independently reviewed by the DOC Site Quality to ensure that the survey measurements have been taken and documented in accordance with approved procedures.

d. Corrective Action Process

The corrective action process for FSS will be conducted in accordance with procedural instructions. Procedures describe the methods used to initiate Condition Reports (CRs) and resolve self assessment and corrective action issues related to FSS.

e. Reports to Management

Reports of audits and trend data will be reported to management in accordance with procedure.

5.10.4 Data Validation

Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels will be made and measurements exceeding the investigation levels will be evaluated. Procedurally verified data will be subjected to the Sign test or the Wilcoxon Rank Sum test as appropriate. Technical evaluation or calculations used to support the development of DCGLs will be independently verified to ensure correctness of the method and the quality of data.

5.11 Access Control Measures

5.11.1 Turnover

Due to the large scope of the final status survey and the need for some activities to be performed in parallel with dismantlement activities, a systematic approach to turnover of areas is established. Prior to acceptance of a survey unit for final status survey, a number of conditions must be satisfied. These include:

a. Decommissioning activities having the potential to contaminate the survey unit must be complete.

b. Tools and equipment not required for the survey must be removed, and housekeeping and cleanup must be complete.

c. Decontamination activities in the area must be complete.

d. Final operational radiological surveys must be complete in impacted areas to ensure that no additional remediation is necessary and will consist of:

   1. Scan surveys to ensure that surface contamination is within the FSS total surface contamination limits.

   2. Smear surveys to ensure that the removable surface contamination is within the FSS removable surface contamination limits (i.e., 10% of the surface contamination limit).

   3. Volumetric samples to ensure soil remediation is within acceptable FSS concentration limits.

e. Access control or other measures to prevent recontamination must be implemented.

f. Turnover surveys may be performed and documented to the same standards as FSS surveys so that data can be used for FSS.

5.11.2 Walkdown

The principal objective of the walkdown is to assess the physical scope of the survey unit. For systems, it will include a review of system drawings and a physical walkdown of the system. Structures and open land areas will also be walked down. The walkdown is best completed when the final configuration of the area is known, usually near or after completion of decommissioning activities for the area.

The walkdown ensures that the area has been left in the necessary configuration for FSS or that any further work has been identified. The walkdown provides detailed physical information for survey design. Details such as floor coatings, structural interferences or sources needing special survey techniques can be determined.

Specific requirements will be identified for accessing the survey area and obtaining support functions necessary to conduct the final status surveys, such as scaffolding, interference removal, and electrical tag out. Safety concerns such as access to confined spaces, high walls and/or ceilings will be identified.

5.11.3 Transfer of Control

Once a walkdown has been performed and the turnover requirements have been met, control of access to the area is transferred from the Construction and Radiation Protection operations groups to the FSS group. Turnover is accomplished through the Work Order process. Access control and isolation methods are described below.

5.11.4 Isolation and Control Measures

Since decommissioning activities will not be completed prior to the start of the final status survey, measures will be implemented to protect survey areas from contamination during and subsequent to the final status survey. Decommissioning activities creating a potential for the spread of contamination will be completed within each survey unit prior to the final status survey. Additionally, decommissioning activities which create a potential for the spread of contamination to adjacent areas will be evaluated and controlled.

Upon commencement of the final status survey for survey areas within the RA where there is a potential for re-contamination, implementation of one or more of the following control measures will be required:

- Personnel training
- Installation of barriers to control access to surveyed areas
- Installation of barriers to prevent the migration of contamination from adjacent overhead areas
- Installation of postings requiring personnel to perform contamination monitoring prior to surveyed area access
- Locking entrances to surveyed areas of the facility
- Installation of tamper-evident labels

Controls will be implemented in accordance with procedure and will remain in place as appropriate until license termination.

5.12 References

- 10CFR20.1402, Radiological Criteria for Unrestricted Use.
- 10CFR50.82, Termination of License.
- 40CFR141.16, National Primary Drinking Water Regulations.
- MY Post Shutdown Decommissioning Activities Report (PSDAR).
- MY Historical Site Assessment.
- MY Site Characterization Report, Volumes 1-9.
- MY Quality Assurance Program.
- MY Corrective Action Program.
- NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination", June 1992, draft.
- NUREG-1575, MARSSIM.
- Draft NUREG-1507, "Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Field Conditions", June 1998.
- NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," Rev.1, June 1998 draft.
- NUREG-1549, "Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination," July 1998 draft.
- Draft Reg Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August 1998 draft.
- Initial and Continuing Training Programs For Decommissioning procedure.

- Program Assessment procedure.
- Instrumentation Program procedure.
- Operation and Calibration of the Gamma Spectroscopy System procedure.
- Operation of the Packard Model 4430 Liquid Scintillation Counter procedure.
- Final Status Survey Program procedure.
- FSS Survey Procedure.
- FSS Classification procedure.
- FSS Quality Control procedure.
- FSS Survey Package Preparation and Control procedure.
- FSS Survey Area Turnover and Control procedure.
- FSS Data Processing procedure.
- Selection, Training and Qualification of RP Personnel procedure.
- Site Quality Program Plan procedure.
- Document Control procedure.
- Field Procurement procedure.
- Corrective Action Program Interface procedure.
- Operation of the Tennelec procedure.

ATTACHMENT A

Survey Unit Breakdown

A.1 Constructing A Survey Unit For A Single Room for Building Occupancy Scenario

This section of the attachment describes the process for constructing an individual survey unit for a representative room in the Primary Auxiliary Building. The room chosen is the Primary Drain Tank (PDT) cubicle located on elevation 11 feet. The room is 15.5 feet (4.7 m) by 18 feet (5.5 m) and 23 feet (7.0 m) in height. The room surface area is 2002 square feet (186 m2) (2099 square feet less the area of the door opening). It is square with a single entrance opening.

The 11 foot elevation of the PAB (figure 2) is characterized under survey area A0600 with the following results.

Survey Area	Location	Surface Area m^2	Class	Surface Contamination dpm/100 cm^2			Required Measurements
				Mean	Range	Sigma	
A0600	El.11, PAB	2664	1	1700	140-19,000	3900	225
A0600	PDT Cubicle	186	1	1700	140-19,000	3900	15

For this example, it is assumed that the DCGL for surface contamination has been determined to be 1.52×10^4 dpm/100 cm^2. The LBGR is set at 0.5 DCGL which gives a relative shift of 1.95. The Type I and II errors are both assumed to be 0.05. A square grid pattern is used. Use of the Sign Test is assumed.

Based on the survey data shown above, and a minimum survey unit size of about180 m^2, this cubicle would be one of 15 Class 1 survey units located on the 11 foot elevation of the PAB and a total of 15 surface measurements would be required for each survey unit. The measurements would be distributed between the walls, floor and ceiling. The survey unit design is illustrated in figure 3 when the survey locations are drawn in for only the floor.

A.2 Constructing An Integrated Survey Unit Maintaining The Sample Density

The assumptions and values given above are used for constructing one large Class 1 survey unit with the same sample density as the entire 11 foot elevation of the PAB. There are 28,669 ft^2 or 2664 m^2 of surface area in the 11 foot elevation of the PAB. If the total area were broken up into survey units of 180 m^2 each, there would be 15 total survey units requiring 15 survey unit design packages. Each survey unit would require 15 surface measurements for a total of 225 measurements.

However, the entire 11 foot elevation can be designed as a single survey unit with a total of 225 surface measurements . The sample density remains the same.

The survey unit illustrated in figure 4 with the surface measurement locations drawn in for the floor **only**. (combination of 15 survey units)

Figure 1

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Figure 1A

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Figure 2

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Figure 3

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Figure 4

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LTP SECTION 6

COMPLIANCE WITH THE RADIOLOGICAL CRITERIA

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APPENDIX B **RESRAD Report for Rubble Fill Dose Assessment**

APPENDIX C **RESRAD Media Concetration Report for Rubble Fill Dose Assessment**

APPENDIX D **Microshield Runs for Excavation Scenario**

6.0 COMPLIANCE WITH THE RADIOLOGICAL CRITERIA

The goal of the MY decommissioning project is to release the site for unrestricted use in accordance with 10 CFR Part 20. This Section describes the dose assessment method to be used to demonstrate compliance with 10CFR 20.1402.

The dose assessment determines the Derived Concentration Guideline Levels (DCGL's) for concrete rubble, standing building surfaces, and soil. The DCGL's are the levels of residual radioactivity that correspond to 25 mrem/yr (TEDE) to the average member of the critical group. MY intends to dismantle equipment and systems and remediate structures and land areas (per LTP sections 3 and 4) to ensure that residual radioactivity levels are at, or below, the DCGL's. After remediation is completed, a final site survey will be performed (per LTP Section 5) to verify compliance with the DCGL's. The final survey report will document that the DCGL's have been met and serve to demonstrate that the Radiological Criteria for License Termination, as codified in 10 CFR 20 Subpart E, has been fully satisfied.

The dose assessment will be performed for each of the following media: 1) concrete rubble fill, 2) excavated concrete rubble, 3) soil, and 4) standing buildings. For each media type, an appropriate dose assessment model, and the model input parameters, were selected and justified. The combination of doses from the various media will be evaluated when determining the dose to the average member of the critical group.

6.1 Critical Group

The regulations in 10 CFR 20 Subpart E require the dose to be calculated for the average member of the critical group. The critical group is defined in 10CFR20.1003 as "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances." The critical group selected for the MY site dose assessment is the resident farmer. The resident farmer is a person who lives on the site, after the site is released for unrestricted use, and derives all drinking and irrigation water from an onsite well. In addition, a significant portion of the residents' diet is assumed to be derived from food grown onsite. NRC guidance in Regulatory Guide DG-4006, NUREG-1549, and NUREG-5512 identify the resident farmer as a conservative onsite critical group.

It is unlikely that any other set of plausible human activities could occur on the site that would result in a dose exceeding that calculated for the hypothetical resident farmer. It is more probable that any actual future occupants of the site would engage in behaviors that would result in lower dose. For example, it is more likely that a hypothetical future resident would use the municipal water supply, as opposed to well water, since this is the common practice in the vicinity of the site and the yield from onsite test wells has been determined to be low. Another example is assuming that the site will be used to continue supplying electricity through

repowering with gas turbines. In this case the future site occupant would be an industrial worker, as opposed to the resident farmer. The important conclusion from these two examples is that the dose calculated for the hypothetical resident farmer will likely be a very conservative estimate of the dose that an actual site occupant or site visitor would receive.

6.2 Concrete Rubble Fill

As discussed in Section 3, the decommissioning approach for onsite concrete structures is to demolish the buildings, rubblize the structural concrete, and use the rubble as fill in accordance with normal and accepted construction practices. The rubble will be used as fill in the existing building "basements" or other site locations, at least 3 feet below grade. The dose assessment for the concrete rubble fill assumes that the rubble will remain in the geometry and location that it is initially placed in. An additional dose assessment was performed using an assumption that the concrete is excavated at some time in the future (see Section 6.3). However, it is important to note that the results of the excavated concrete assessment reported in Section 6.3 indicate that the potential dose from excavated concrete is lower than the potential dose calculated using the conservative concrete rubble fill dose assessment described below.

6.2.1 Exposure Pathways

The resident farmer is the critical group for the concrete rubble fill dose assessment. NUREG-1549 describes several environmental transport pathways that NRC has determined to be generically applicable to the resident farmer critical group. The generic pathways in NUREG-1549 were developed for contaminated soil located on the ground surface and are identical to the pathways used in NRC's DandD screening code. However, since the type of material being evaluated is different, i.e., concrete as opposed to soil, and the geometry of the material is different, i.e., subsurface as opposed to surface, not all of the pathways listed are applicable for the concrete rubble fill.

The applicability of the generic NUREG-1549 pathways to the concrete rubble fill source term was evaluated. The following resident farmer exposure pathways from NUREG-1549 are applicable to concrete rubble fill:

a. Ingestion of drinking water from an onsite well,

b. Ingestion of plant products irrigated from an onsite well,

c. Ingestion of animal products grown onsite that drink water from an onsite well and feed on plant products irrigated with onsite well water, and,

d. Inhalation, ingestion, and direct radiation from soil that becomes contaminated by irrigation water.

The following resident farmer exposure pathways from NUREG-1549 are not applicable to the concrete rubble fill:

a. External exposure,

b. Inhalation of resuspended concrete,

c. Direct ingestion of concrete,

d. Ingestion of plant products grown in contaminated concrete,

e. Ingestion of animal products grown onsite that feed on plant products grown in contaminated concrete or ingest contaminated concrete directly, and

f. Ingestion of fish from a contaminated surface water source.

The first five pathways are not applicable since the rubble fill scenario assumes that there is a three foot cover over the concrete. In addition, the inhalation and ingestion scenarios apply to soil, not concrete. However, the external exposure pathway is applicable if the fill is excavated at some time in the future. This scenario is evaluated in Section 6.3. The potential for surface water contamination was evaluated and determined to be an insignificant potential pathway. A bounding analysis of a potential site-specific scenario, i.e., the worm digger, demonstrates that the potential dose from this scenario is also insignificant. The inhalation, ingestion, and direct radiation from soil was determined to be an insignificant contributor to dose and was not further evaluated.

6.2.2 Dose Modeling

The approach for estimating the dose from concrete rubble fill consists of two steps: 1) estimate the leach rate from the concrete and the corresponding maximum water concentration, and 2) estimate the dose from drinking the water from an onsite well and ingesting plants and animals affected by irrigating from an onsite well. The assumptions used in the assessment are very conservative and are essentially worst case. These calculations are intended for regulatory compliance and do not represent the dose that an actual future occupant would reasonably be expected to receive. The actual dose is expected to be much lower.

The potential source of exposure from concrete rubble fill is residual radioactivity leaching from the concrete. The NRC DandD, Version 1, screening model and the RESRAD pathway analysis/dose assessment code were evaluated to determine if they were applicable for modeling the leach rate from concrete rubble. There are three major reasons why these codes are not applicable to concrete. First, a significant process affecting the leach rate from concrete is diffusion. Neither code has a diffusion component. Second, the primary geometry of the residual contamination in the concrete is a 0.1 cm layer at the surface. This geometry is different from the assumption of a uniform contamination in the two codes. The third reason is that the models are specifically designed for soil contamination, not concrete, and a number of the model input parameters are not applicable to concrete. Since the DandD and RESRAD codes are not appropriate for modeling the concrete source term, a model specifically designed for this purpose was used.

The model used, DUST-MS (Version 3) allows a direct evaluation of the leach rate from the concrete in the distribution and geometry representative of those actually encountered. No assumptions regarding mixing or uniform contamination are required. The DUST-MS code uses widely accepted mathematical models for calculating the diffusion rate of radionuclides out of the concrete. As a radionuclide leaches from the concrete, the total activity present in surrounding water, as a function of time, is dependent on the diffusion coefficient, the adsorption coefficient, and the half-life of a given radionuclide. It is important to note that the release, or leaching, of radionuclides is not unique to concrete. Residual radioactivity in soil will also be released through a similar process, except that the diffusivity is an important factor for the concrete, while it is not a factor for soil.

A literature search was conducted to determine the range of potential values for the major input parameters, i.e., the diffusion and adsorption coefficients. A preliminary dose assessment using the literature values indicated that Cs-137 resulted in over 95% of the conservatively estimated dose. Therefore, to ensure the highest confidence in the results, a laboratory analyses of the site-specific diffusion and adsorption coefficients for Cs-137 were performed. The analytical results were used in the dose assessment for Cs-137 in place of the literature values.

The maximum accumulated activity was used as the input for estimating the concentration in the leachate, i.e., the concentration in the water surrounding the concrete. The volume of water assumed for the concentration calculation was based on the annual water usage defined in NRC default parameter distribution functions provided in NUREG-1549 and a site-specific irrigation rate. No dilution of the leachate through diffusion, dispersion, or adsorption during groundwater transport was assumed. The concrete-water mixture was treated as a stationary batch mixture. Using

the maximum annual concentration and assuming no dilution are conservative assumptions. This is particularly true since the maximum concentration occurs within the first year after license termination for most of the radionuclides and it is extremely unlikely that a resident farmer would be on the MY site within this time frame. After the first year, the concentration is lowered continuously through radionuclide decay. In the first 30 years, the concentration is reduced by a factor of two.

An additional conservatism in the calculations was the assumption that there will be no mixing and dilution with the lower concentration water that may be associated with adjacent concrete that contains lower levels of residual radioactivity. The highest "aliquot" of water in the center of the entire rubble fill volume was used in the dose calculations.

As stated above, the water surrounding the rubble fill is assumed to be used by an onsite resident farmer. A shallow well is hypothetically drilled into the center of the rubble fill volume that contains the highest residual radioactivity and draws only from the water surrounding the rubble. The shallow well assumption is very conservative since the most likely situation is that a deeper well would be drilled that draws from other water sources. These other sources would cause dilution in the pumped well water and lower concentrations. In addition, the well is assumed to be drilled at the time when the water concentration calculated by the DUST-MS code is at the maximum possible.

The RESRAD (Version 5.82) pathway analysis/dose assessment code was used to estimate the dose to the resident farmer through the drinking water and irrigation pathways using the maximum water concentrations determined using DUST-MS. The RESRAD input parameters were modified in accordance with NRC's "Preliminary Guidelines for Evaluating Dose Assessments in Support of Decommissioning." These modifications ensure that the NRC default parameters from the DandD screening code are used to the extent possible. The resulting dose to the resident farmer was used to derive site-specific DCGL's for concrete rubble.

6.2.3 DUST-MS Modeling

The DUST-MS Version 3 model was used to conservatively estimate the concentration in water surrounding the concrete rubble fill. This section describes in the assumptions and calculations supporting the DUST-MS assessment, and the final results.

a. Assumptions

1. Volume of Annual Water Usage

The annual volume of water required to supply the domestic water needs of a family of four, including irrigation, was calculated. This is the baseline value used to calculate the DUST-MS facility volume and the corresponding concrete volume to be input. This volume is required to calculate the volume of concrete used in the DUST-MS evaluation. An estimate of the annual water volume used for domestic purposes in the State of Maine was reported in Table 3.8.1 of the "Review of Parameter Data for the NUREG/CR-5512 Residential Farmer Scenario and Probability Distributions for the DandD Parameter Analysis." The volume is 124,422 liters for an individual and 497,690 liters for a family of four.

A site-specific irrigation rate of 0.091 l/m^2*d was used. To estimate the total annual irrigation water volume, it was assumed that the area irrigated is10,000 m^2. The resulting annual irrigation volume is then calculated to be 330,000 liters.

Combining the domestic water volume with the irrigation volume results in the total annual usage. This value is 827,690 liters. The rounded value of 827,000 liters (827 m^3) is used in the assessment.

2. Porosity

Porosity is the percent void space between individual, variably-sized, pieces of concrete in a random configuration. This porosity is conservatively assumed to not include the entrained pore space within each concrete piece, which was measured to be 15% for the MY concrete. The rubble is produced by physical battering and/or mechanical crushing and will be comprised of pieces with a range of sizes. A similar waste form would be produced by breaking rock for civil engineering excavations. The porosity of the concrete rubble is estimated to be between 26% and 48%. A porosity of 30% was selected for the DUST-MS analyses.

4. Concrete Volume and Facility Volume

The DUST-MS "facility volume" is calculated by dividing the annual

water use by the porosity (0.3). The resulting facility volume is 2756 m^3. The concrete volume is calculated by multiplying the facility volume (2756 m^3) by 0.7 (1 - porosity). The resulting concrete volume is 1929 m^3.

5. Concrete Surface Area/Volume Ratio

The concrete surface area/volume ratio is required to calculate the source term for the DUST-MS evaluation. The ratio used for the DUST-MS runs was 1 m^2/m^3, and provides a baseline value to use in the DCGL calculations described in Section 6.2.5. The $1m^2/m^3$ ratio is a generic value and does not represent actual building ratios. The actual building area/volume ratios were accounted for in the DCGL calculations.

6. Contaminated Concrete Surface Area

The concrete surface area is required to determine the total inventory for the DUST-MS model. The surface area is calculated as follows:

$$(1929 m^3 Concrete)\left(1.0\frac{m^2}{m^3}\right) = 1929\, m^2 \text{ contaminated surface area}$$

7. Source Term

For each radionuclide modeled, a total inventory is entered. The inventory used is calculated assuming a concentration of 1 dpm/100 cm^2 of contaminated concrete surface. The inventory entered is arbitrary since the DUST-MS results are used in relative terms as input to the RESRAD dose calculation. The 1 dpm/100 cm^2 unit is used to facilitate the DCGL calculation. The total inventory used in the DUST-MS assessment is calculated as follows:

$$(1929\, m^2 \text{ contaminated concrete surface})\, x$$

$$(1.0\frac{dpm}{100\, cm^2})\, x\, (10^4\frac{cm^2}{m^2})\, x\, (\frac{10^{-6}\mu Ci}{2.2\, dpm}) = 0.088\, \mu Ci$$

8. Rubble Size and Shape

The rubble fill is expected to consist of concrete pieces of various sizes. Detailed studies of concrete cores from the MY structures have

determined that the residual radioactivity is contained in the first 0.1 cm of the concrete surface. The larger the size, the slower the diffusion process. In addition, rectangular pieces, with residual activity in the first 0.1 cm will diffuse slower than spherical pieces with a 0.1 cm diameter. In order to be conservative, the rubble pieces are assumed to all be 0.1 cm diameter spheres, for purposes of DUST-MS modeling.

9. Density of the concrete

2.2 g/cm^3 based on laboratory measurements on site samples.

10. Porosity of the concrete

15 percent based on laboratory measurements on site samples.

11. Diffusion Coefficient and Distribution Coefficient

A literature search was conducted to compile published data on diffusion coefficients and distribution coefficients for radionuclides in concrete or Portland cement mixtures. The range, together with the conservatively selected values that were used in the modeling, for the radionuclides identified in MY structural concrete are provided in Table 1. The values for Ni-63 and Fe-55 are assumed to be the same as for Co-60. For H-3, all of the source term is assumed to be released. However, due to the low fraction of H-3 present and the very low dose factor for H-3, it contributes a negligible fraction of the dose.

b. DUST-MS Modeling Results

DUST-MS was used to estimate potential concentrations of radionuclides in water surrounding the rubble, using a conservative assumption of a closed system with no through-flow of water or mixing with adjacent volumes. Concentrations of each radionuclide were allowed to decay according to their respective radioactive half-life. The results of the DUST-MS runs are shown graphically in Figures 1 to 7. In addition to the graphical output, the DUST-MS input files and the important parts of the output files for the radionuclides listed in Table 1 run are provided in Appendix A.

The graphs display the water concentration in the concrete rubble fill as a function of time. Figures 1 through 7 show the results for Cs-137, Cs-134, Co-60, Ni-63, Fe-55, Np-237 and Sr-90. DUST-MS runs were not

performed for H-3 since it was assumed that all of available H-3 was immediately released and mixed in the surrounding water. Note the decrease in concentration with time due to radioactive decay. The maximum water concentrations and time when the maximum concentration occurred are listed in Table 2. Note that for all of the radionuclides except for cesium the most conservative diffusion

Table 1
Range of published magnitudes of adsorption distribution coefficient, K_d, and diffusion coefficient, D, for key radionuclides and the values used in DUST-MS modeling

Nuclide	K_d Range cm^3/gm	Values used for Modeling	D Range cm^2/sec	Values used for Modeling
Co-60	100	100	1E-14 to 1E-09	1E-09
Cs-134	1 to 5	3.0*	4E-12 to 4E-06	2E-10*
Cs-137	1 to 5	3.0*	4E-12 to 4E-06	2E-10*
Fe-55	100	100	1E-14 to 1E-09	1E-09
Ni-63	100	100	1E-14 to 1E-09	1E-09
Sr-90	0.7 to 3.0	0.7	1E-11 to 2E-09	2E-09
Np-237	500 to 15,000	500	1E-17 to 1E-11	1E-11

* Note that the values for Cs-134 and Cs-137 K_d and D value were based on laboratory measurements.

coefficient identified during the literature search was used in the assessment. The values for cesium were determined through laboratory analyses. Using the highest diffusion coefficient found in literature is likely to be very conservative, particularly since the literature indicates that construction concrete, such as that at MY, had significantly lower diffusion coefficients than the worst case values used in the DUST-MS runs (see references in Section 6.10). If more realistic diffusion coefficients were used, the time when the maximum concentration occurs would be later than listed in Table 2. The maximum concentration would also decrease if more realistic diffusion coefficients were used.

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DUST-MS results for Cs-137

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DUST-MS results for Cs-134

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DUST-MS results for CO-60

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DUST-MS results for Ni-63

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DUST-MS results for Fe-55

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DUST-MS results for Np-237

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DUST-MS results for Sr-90

6.2.4 RESRAD Dose Calculation for Concrete Rubble Fill

a. Assumptions and Input Parameters

The RESRAD code Version 5.82 was used to calculate the dose from the hypothetical well pumping scenario. The maximum concentrations in the water that were calculated using the DUST-MS model was input into the RESRAD code. As described in Section 6.2.1, only the groundwater dependent pathways are applicable for the concrete rubble fill assessment, including drinking water and all agricultural pathways affected by irrigation. The radionuclide mixture used in the calculation was determined through an evaluation of representative concrete core samples (see LTP Section 2).

The initial well water concentrations selected for the RESRAD run were calculated using the results of the DUST-MS code and the radionuclide mixture in the MY concrete. The concentrations for the various radionuclides were calculated based on the radionuclide fractions in the concrete relative to Cs-137. An additional consideration was the correction of the1 dpm/100 cm^2 source term assumption used in the DUST-MS runs to properly account for the actual radionuclide mixture and to correlate the mixture to the assumptions implicit in the use of gross beta for the units of the DCGL.

The method and rationale for calculating RESRAD input water concentrations that can be directly correlated to concrete surface dpm/100 cm^2 gross beta and the final DCGL is described below. The explanation and discussion is based on the inputs and results presented in Table 3.

<u>Column 1</u>

Column 1 lists the radionuclides in the Maine Yankee concrete mixture.

<u>Column 2</u>

Column 2 provides the ratio of each radionuclide to Cs-137. The nuclides are ratioed to Cs-137 in order to calculate the water concentrations for the hard-to-detect radionuclides. Cesium-137 was selected as the basis for the ratio since it is the predominant radionuclide and is a beta emitter.

Table 2 Maximum concentration in water surrounding concrete rubble, and time when maximum occurs.		
Radionuclide	**Maximum Concentration Time (year)**	**Maximum Concentration pCi/l**
Cs-137	0.57 years	6.4E-03
Cs-134	0.185 years	6.0E-03
Co-60	0.08 years	2.0E-04
Sr-90	0.05 years	2.3E-02
H-3	NA	NA
Ni-63	0.08 years	2.1E-04
Fe-55	0.06 years	2.0E-04
Np-237	5.9 years	4.1E-05

TABLE 3 Calculation of water concentrations for input to RESRAD analysis Column 6 provides water concentrations used.					
	Radionuclide Concentration		**Water Radionuclide Concentration**		
Nuclide	**Relative to Cs-137**	**Corrected for Gross Beta Assumptions**	**From DUST-MS pCi/l (1)**	**From DUST-MS Corrected for Gross Beta Assumptions (2)**	**RESRAD Input pCi/l (3)**
Fe-55	2.9E-02	2.3E-02	2.0E-04	4.6E-06	9.2E-02
Cs-134	2.2E-02	1.8E-02	6.0E-03	1.1E-04	2.2E+00
Co-60	2.3E-01	1.8E-01	2.0E-04	3.6E-05	7.2E-01
Ni-63	1.9E+00	1.5E+00	2.1E-04	3.1E-04	6.2E+00
Sr-90	2.3E-03	1.8E-03	2.3E-02	4.1E-05	8.2E-01
Cs-137	1.0E+00	8.0E-01	6.4E-03	5.1E-03	1.0E+02
Np-237	2.4E-04	1.9E-04	4.1E-05	7.8E-09	1.6E-04

(1) units: pCi/l per 1dpm/100 cm^2 of each radionuclide
(2) units: pCi/l per 1dpm/100 cm^2 gross beta
(3) units: pCi/l per 20,000 dpm/100 cm^2 gross beta

Column 3

Column 3 provides corrected radionuclide/Cs-137 ratios to properly correlate the RESRAD inputs to the DCGL calculation, which is based on measurable gross beta activity. Recall that the DUST-MS inventory input was based on 1 dpm/100 cm^2 for each radionuclide. The DUST-MS results that were generated using the constant1 dpm/100 cm^2 source term require two corrections: 1) each radionuclide source term requires correction to reflect the actual fraction of the given radionuclide in the concrete, and 2) the relative source term fractions require correction to correlate the results of the RESRAD runs directly to the DCGL, which is calculated in terms of dpm/100 cm^2 gross beta. The corrections for the radionuclide fractions in the concrete are provided in Column 2. The correction made to correlate the RESRAD results to a gross beta DCGL is described below.

The generic DCGL was calculated in units of dpm/100 cm^2 gross beta (see Section 6.2.5). The measurable beta-emitters in the radionuclide mixture are Cs-137, Cs-134, and Co-60 (Sr-90 contributes a negligible fraction). Therefore, 1 dpm gross beta is comprised of these three radionuclides at their respective fractions. These fractions are 80% Cs-137, 18% Co-60, and 2% Cs-134. Explained another way, if 1 dpm/100 cm^2 gross beta is measured on a concrete surface, this would include 0.80 dpm Cs-137, 0.18 dpm Co-60, and 0.02 dpm Cs-134. Of course, all of the other radionuclides would also be present at their respective fractions. Since Column 2 provides the mixture fractions as a ratio to Cs-137, multiplying all of the ratios in Column 2 are multiplied by 0.80, provides the value that reflects the dpm of each radionuclide present if the Cs-137 dpm were 0.80. Further, since 0.80 dpm Cs-137 is present when the gross beta dpm is 1, the resulting fractions represent the dpm of each radionuclide when the gross beta measurement is 1 dpm. Column 3 lists the values that result from multiplying Column 2 by 0.80.

Column 4

Column 4 simply lists the water concentration outputs of the DUST-MS runs in units of pCi/l. Recall that the pCi/l outputs from the DUST-MS runs were calculated assuming that each radionuclide is present at a concentration of 1 dpm/100 cm^2 over a concrete surface area of 1929 m^2.

Column 5

Column 5 is generated by multiplying Column 3 by Column 4. The values in Column 3 represent the dpm of each radionuclide that would be present if the

total gross beta were 1 dpm. Since the values in Column 4 were calculated assuming that the radionuclide is present at a concentration of 1 dpm/100 cm^2, multiplying Column 4 by Column 3 results in the pCi/l concentration for each radionuclide that would be present if the total gross beta were 1 dpm/100 cm^2.

Column 6

Column 6 contains the RESRAD input water concentrations and are obtained by multiplying the values in Column 5 by 20,000 dpm/100 cm^2. This is an arbitrary adjustment to ensure that the RESRAD input concentrations are high enough to generate doses greater than zero. The maximum concentration of H-3 due to leaching from the concrete is 41 pCi/l assuming that the entire H-3 inventory is released to the water. However, past monitoring well sampling identified a maximum concentration of 4023 pCi/l H-3 in groundwater. The concentration used in the RESRAD run, is the sum of 41 and 4023 pCi/l , ie., 4064 pCi/l H-3 .

The groundwater concentrations from Column 6 of Table 3 were input directly into the RESRAD code. This was accomplished through the use of the "Water Concentration Option" in the "Source Transfer Factors" module in RESRAD. Due to an artifact in the RESRAD code, the concentration that was input into the RESRAD water concentration field did not exactly match the resulting concentration in the "well water concentration" report. This was handled in a conservative way by ensuring that the "well water" concentrations used in the dose calculation were equal to or greater than the concentrations listed in Table 3, Column 6. As stated in Section 6.2.1, only the groundwater dependent pathways are applicable to the concrete rubble fill dose assessment. The excavation scenario evaluated in Section 6.4, assumes that the rubble is excavated and placed in a geometry where other pathways, such as external exposure, apply.

Three RESRAD parameter modifications are required to enable the code to calculate dose directly from an input water concentration. These modifications do not affect the calculated dose since they are related to the RESRAD groundwater transport modules that were not used in the assessment. The changes were required to allow the RESRAD well water concentrations at time equal to zero years (the time used for the dose calculation) to be equal to or greater than the concentrations listed in Table 3, Column 6. The lack of effect of the RESRAD transport parameters is verified by a review of the RESRAD media concentration output form which shows the well water concentrations used in the analysis are equal to or greater than those listed in Column 6 of Table 3. In addition, soil concentration values of 1E-06 pCi/g were entered for

each radionuclide. A non-zero concentration value is required before RESRAD will accept the "water concentration" data and has no bearing on the assessment. The parameters changed and the values used are listed below:

- thickness of contaminated zone - 0.05 meters,
- time since material placement - 1 year
- unsaturated zone thickness - 0.01 meters

A number of additional RESRAD parameters were modified in accordance with NRC's guidance in "Preliminary Guidelines for Evaluating Dose Assessments in Support of Decommissioning." These parameter changes ensure consistency, to the extent possible, with the default parameters in NRC's DandD code. The parameters modified and the values used are listed in Table 4. Two site specific parameters were used. The irrigation rate was determined to be 0.033 m/y. The well pumping rate was 827 m/y, consistent with DUST-MS analysis.

The basis for the well pumping rate was provided in Section 6.2.3. The irrigation rate was based partly on discussions with a local extension representative who stated that Maine was not a big irrigation state. Annual rainfall is usually on the high side of the national average. The extension representative also stated that worst case non-commercial irrigation rate, under drought conditions, was about 0.09 m/y. However, the worst case value was not considered appropriate for the dose assessment and additional evaluations were as discussed below.

A local hydrogeologist provided information that local well water usage averages about 250 gallons per day. If all the well water were diverted to irrigation, the resultant value of 250 gallons/day is 9.5E-2 L/m2-d or 0.033 m/y. The hydrogeologist also stated that the groundwater flow into the containment foundation sump was between 0.22 to 1.59 gallons per minute and this rate represents the rate a shallow well at the location could yield. The lower value would be indicative of the yield from the well during dry or drought conditions. This would provide about 318 gallons per day. Therefore, it would appear that the irrigation rate should actually be less than the 0.033 m/y value assumed since there would not be enough water for domestic use requirements plus irrigation. It is also important to note that the irrigation rate is a relatively insensitive parameter in the dose assessment except for the cow fodder pathway. In Maine, as the extension representatives stated, irrigation is not used for fodder, especially during times of drought.

b. RESRAD Results

The RESRAD output files are provided in Appendix B and C. Page 11 of Appendix B provides the most important information, i.e., that the dose from the water concentrations listed in Table 3, Column 6 is 4.4 mrem/yr. Recall that this 4.4 mrem/yr is the dose that would result if 20,000 dpm/100 cm^2 gross beta were uniformly distributed over 1929 m^2 of concrete surface. The results clearly show that Cs-137 is the primary contributor to dose from concrete rubble fill, contributing 91% of the total dose. Considering all of the radionuclides and pathways analyzed, the dose from drinking water contributes 63% of the total, with milk, meat, and plants affected by the irrigation pathway contributing the remaining 37%. Page 2 of Appendix C shows that the well water and drinking water concentrations used in the RESRAD analysis were equal to or greater than the concentrations listed in Table 3, Column 6.

Table 4
Behavioral and Metabolic Parameters from NRC Guidance used in the RESRAD Analysis

Parameter	Value	Units
Inhalation rate	1.169e+4	m^3/y
Mass loading for inhalation	3.14e-06	g/m^3
Shielding factor for external gamma radiation	0.5512	
Fraction of time spent indoors	0.6571	
Fraction of time spent outdoors	0.1101	
Fruits, vegetables, and grain consumption	112	kg/y
Leafy vegetables consumption	21.4	kg/y
Milk consumption	233	L/y
Meat and poultry consumption	65.1	kg/y
Fish consumption	20.6	kg/y
Soil ingestion	18.26	g/y
Drinking water intake	478.5	L/y
Contamination fraction of drinking water	1	
Contamination fraction of livestock water	1	
Contamination fraction of irrigation water	1	
Contamination fraction of aquatic food	1	
Contamination fraction of plant food	1	
Contamination fraction meat	1	
Contamination fraction of milk	1	
Livestock fodder intake for meat	27.1	kg/d
Livestock fodder intake for milk	63.25	kg/d
Livestock water intake for meat	50	L/d
Livestock water intake for milk	60	L/d
Growing season for non-leafy vegetables	0.25	y
Growing season for leafy vegetables	0.123	y
Growing season for fodder	0.15	y
Storage time for fruits, non-leafy veg., and grain	14	d
Storage time for leafy vegetables	1	d

Table 4 Behavioral and Metabolic Parameters from NRC Guidance used in the RESRAD Analysis		
Parameter	**Value**	**Units**
Storage time for milk	1	d
Storage time for meat and poultry	20	d
Storage time for livestock fodder	0	d
Fraction of grain in beef cattle feed	0.0743	
Fraction of grain in milk cow feed	0.0308	

6.2.5 DCGL Calculation

The Generic DCGL is calculated in units of dpm/100 cm^2 gross beta activity. As described below, the gross beta surface activity DCGL is directly correlated to the radionuclide mixture and total inventory in the MY concrete, and to the results of the DUST-MS and RESRAD evaluations. The basic equation for determining a Generic DCGL is provided below, assuming a generic surface area to volume ratio of 1 m^2/m^3.

$$\text{Generic DCGL (dpm/100 cm}^2\text{)} = \frac{(20{,}000 \text{ dpm/100 cm}^2 \text{ gross beta})}{(4.4 \text{ mrem/yr})} \times 25 \text{ mem/yr}$$

$$= 113{,}500 \text{ dpm/100 cm}^2 \text{ gross beta}$$

(assuming concrete surface area to volume ratio of 1 m^2/m^3)

The gross beta DCGL is calculated in units of gross beta surface activity since surface activity measurements are the preferred method for demonstrating compliance. To convert the gross beta DCGL to specific radionuclide activity simply multiply the gross beta DCGL by the value listed in Table 3, Column 2. For example, to determine the amount of Cs-137 present when the gross beta DCGL is 113,500 dpm/100 cm^2, multiply this value by 0.8.

As mentioned above, the gross beta DCGL was developed to suit a measurement technique. The total activity in a given 1929 m^3 is the basic compliance DCGL and is called the "Fundamental DCGL." The Fundamental DCGL is directly linked to dose modeling and is the underlying basis for compliance with the unrestricted use criteria. . For example, the Generic DCGL is calculated to be 113,500 dpm/100 cm^2 gross beta. Consistent with the Generic gross beta DCGL assumptions, the total gross beta activity per 1929 m^2 of concrete surface area is 5.1 uCi. Since this area represents 1929 m^3 of concrete under the generic assumption of 1 m^2/m^3, the Fundamental DCGL can be expressed as 5.1 uCi/m^3. However, it is important to understand that the dose assessment assumes that the contamination is on the structure surface and that no mixing or blending is required to meet the concrete fill DCGL's.

The gross beta DCGL's listed in Table 5 may change as long as the Fundamental DCGL of 5.1 uCi/m^3 gross beta is demonstrated to be met. If measurements other than gross beta are used, such as in-situ gamma spectroscopy, it may be more appropriate to apply the Fundamental DCGL (expressed as individual isotope concentrations) as opposed to the gross beta DCGL. In addition, the Fundamental DCGL can be expressed as individual radionuclide DCGL's. In this case, the unity rule, combined with the use of surrogates as necessary, would be used to determine the 25 mrem/yr unrestricted use level.

The above Generic DCGL requires a correction for the actual surface area to volume ratios in the MY structures. Because of the different construction specifications in each building, a separate correction, and corresponding DCGL, was required for each building.

To calculate building specific DCGL's, the Generic DCGL is divided by the building specific surface area to volume ratios. The is required since the surface area is used in the inventory calculation for the DUST-MS runs (see Section 6.2.3.1). The inventory is directly proportional to the surface area to volume ratio and the resulting dose. In addition, the final DCGL's require consideration of the source term in the building basements as described in the following text. If rubble fill is used on other parts of the site, ie., not placed in building basements, no basement correction factor is required.

To account for the building basements, it is assumed that the activity in the basements is released and distributed uniformly throughout their volumes. The DCGL's for the basements, and the corrected DCGL's for concrete fill, are derived based on the fact that the Fundamental DCGL can be expressed as total activity per unit volume. Measurable gross beta dpm/100 cm^2 is converted to total uCi for ease of calculation. As discussed above, each 1929 m^{3}of concrete is determined to contain 5.1 uCi/m^3 gross beta at the DCGL level.

To conserve the Fundamental DCGL (expressed as uCi/m^3 gross beta), the total activity attributable to the basement concrete surfaces must be subtracted from the activity allowed in the concrete fill, and the concrete fill DCGL lowered accordingly. A correction factor is calculated for each building basement. The lowest correction factor is applied to the concrete fill DCGL calculation for all buildings to ensure that the concrete debris from any building can be used as fill in any basement. It is very important to note that as the project proceeds, the DCGL for any building basement and/or the DCGL for any concrete fill may be adjusted as long as the Fundamental DCGL, which is based on activity per volume, is demonstrated to be met for each 1929 m^{3}of concrete. In addition, adjustments to building basement DCGL's may be made based on the known mean contamination level of the concrete fill to be placed in the basement. For example, assume that the final survey for all concrete fill to be

placed in a given basement has been completed and has shown that the mean residual contamination level in the concrete fill is below the DCGL level. In this case, the basement DCGL can be adjusted upward as long as the Fundamental DCGL continues to be satisfied. The use of the mean assumes that the MARSSIM statistical test have been passed at the 95% confidence level.

The physical volume occupied by the 1929 m^3 concrete is 2756 m^3, consistent with the DUST-MS runs. This requires that the 5.1 uCi/m^3 Fundamental DCGL calculated based on concrete volume be adjusted by a factor of 1929/2756, resulting in 3.6 uCi/m^3. This ensures that the dose assessment for demonstrating compliance with the unrestricted use criteria is satisfied.

The ratio of basement concrete surface area/open volume is listed below for each building:

- Containment: 3659 m^2 / 6760 m^3
- Spray: 1444 m^2 / 1325 m^3
- PAB: 1827 m^2 / 1615 m^3
- Fuel 212 m^2 / 627 m^3

The total gross beta activity contribution from the concrete surfaces is calculated assuming the following gross beta activity level DCGL's for the building basement concrete surfaces. The basement DCGLs listed below were selected based on engineering judgement considering the expected contamination levels in building basements.

- Containment: 40,000 dpm/100 cm^2
- Spray: 40,000 dpm/100 cm^2
- PAB 30,000 dpm/100 cm^2
- Fuel 30,000 dpm/100 cm^2

The total gross beta activity on the concrete surfaces is calculated as follows:

<u>Containment:</u>

(3659 m^2)(40,000 dpm/100 cm^2)(4.5E-05 uCi/m^2 per dpm/100 cm^2) = 6.59E+03 uCi

<u>Spray:</u>

(1444 m^2)(40,000 dpm/100 cm^2)(4.5E-05 uCi/m^2 per dpm/100 cm^2) = 2.60E+03 uCi

PAB:

(1827 m^2)(30,000 dpm/100 cm^2)(4.5E-05 uCi/m^2 per dpm/100 cm^2) = 2.47E+03 uCi

Fuel:

(212 m^2)(30,000 dpm/100 cm^2)(4.5E-05 uCi/m^2 per dpm/100 cm^2) = 2.86E+02 uCi

The activity per volume in each basement is calculated as follows:

- Containment = (6.59E+03 uCi)/(6760 m^3) = 0.97 uCi/m^3
- Spray = (2.60E+03 uCi)/(1325 m^3) = 1.96 uCi/m^3
- PAB = (2.47E+03 uCi)/(1615 m^3) = 1.53 uCi/m^3
- Fuel = (2.86E+02 uCi)/(627 m^3) = 0.46 uCi/m^3

A final adjustment is required for the Spray and PAB basements to account for the fact that the basement volumes are less than the 2756m^3 physical volume required to contain the annual volume of drinking water. This means that a well drilled into these basements would require an additional, clean, water source to makeup the difference required for annual domestic water use. To account for this dilution, the allowed source term in the concrete is adjusted as follows:

Spray:

(2756 m^3 physical volume / 1325 m^3 basement volume) x 3.6 uCi/m^3 = 7.49 uCi/m^3

PAB

(2756 m^3 physical volume / 1615 m^3 basement volume) x 3.6 uCi/m^3 = 6.14 uCi/m^3

Fuel

(2756 m^3 physical volume / 627 m^3 basement volume) x 3.6 uCi/m^3 = 15.8 uCi/m^3

A correction factor was calculated for each building basement as shown below:
Basement Correction Factors:

Containment:	[(3.6 uCi/m^3 fill) - (0.97 uCi/m^3 basement)] / (3.6 uCi/m^3) = 0.73
Spray:	[(7.49 uCi/m^3 fill) - (1.96 uCi/m^3 basement)] / (7.49 uCi/m^3) = 0.74
PAB:	[(6.14 uCi/m^3 fill) - (1.53 uCi/m^3 basement)] / (6.14 uCi/m^3) = 0.75
Fuel:	[(15.8 uCi/m^3 fill) - (0.46 uCi/m^3 basement)] / (15.8 uCi/m^3) = 0.97

The concrete fill DCGL was calculated for each building as shown below. The lowest basement correction factor, i.e. 0.73, was applied to each building DCGL calculation. This ensures that the concrete debris from any building can be placed into any basement.

Containment Concrete Fill DCGL :
DCGL Containment Fill = 0.73(113,500/2.3) = 36024 dpm/100 cm^2

Spray Concrete Fill DCGL:
DCGL Spray Fill = 0.73(113,500/2.4) = 34523 dpm/100 cm^2

PAB Concrete Fill DCGL:
DCGL PAB Fill = 0.73(113,500/3.1) = 26727 dpm/100 cm^2

Fuel Concrete Fill DCGL:
DCGL Fuel Fill = 0.73(113,500/1.6) = 51784 dpm/100 cm^2

RCA Concrete Fill DCGL:
DCGL RCA Fill = 0.73(113,500/2.6) = 31867 dpm/100 cm^2

Table 5

Building Specific DCGL's (dpm/100 cm^2 gross beta)

Building	Concrete Fill Surface Area/ Volume Ratio (m^2/m^3)	Concrete Fill DCGL* (dpm/100 cm^2 gross beta)	Basements (3' below grade) DCGL (dpm/100 cm^2 gross beta)
Containment **	2.3	36,000	40,000
PAB	3.1	26,700	30,000
Fuel	1.6	51,700	30,000
Spray	2.4	34,500	40,000
RCA	2.6	31,800	No Basement

* The DCGL's for the limited concrete structures not listed in this table will be set at the Standing Building DCGL or an area specific Concrete Fill DCGL will be calculated based on the area specific surface area to volume ratio.
** The DCGL for the steel liner in the containment building will be set at the standing building DCGL or adjustments will be made to the basement correction factors as necessary to allow a separate DCGL to be calculated.

6.2.6 Unity Rule for Soil and Concrete Fill DCGL's

The resident farmer scenario applies to both the soil and the concrete fill source terms. The dose from the two sources are additive. Therefore, to demonstrate compliance with the unrestricted use criteria, the unity rule will be applied to the soil DCGL's listed in Section 6.6 and the DCGL's listed in Table 5. The unity rule will apply within a 100 m radius of areas where significant volumes of contaminated concrete fill exists. This is consistent with the assumption of 10,000 m^2 land use assumption in the resident farmer scenario.

6.2.7 Concrete Survey Unit Size

As stated above, the rubble fill dose assessment results in the most conservative DCGLs and will be used to demonstrate compliance with the unrestricted use criteria. In accordance with MARSSIM, the survey unit size is based on the assumptions used in the dose assessment, and compliance is based on individual survey units. For buildings to be demolished, the survey unit size is based on the rubble fill dose assessment.

Section 6.2.4. describes the basis for the concrete fill dose assessment, which includes the assumption that the assessment concrete volume is 1929 m^3. From Table 5, it is seen that the surface area to volume ratio for the concrete walls and floors in the MY buildings ranges from 1.6 to 3.1. That is, the structure surface areas corresponding to 1929 m^3 range from 3086 m^2 to 5980 m^2. Because this range of surface areas is based directly on the dose assessment, this is also the range of acceptable survey unit sizes.

6.3 Excavated Concrete Rubble

The dose assessment described in Section 6.2 assumes that the concrete debris remains in the location where it was placed during decommissioning. A second, less likely, possibility is that the concrete is excavated from this location at some time in the future after the site is released for unrestricted use. The time when this hypothetical excavation occurs will affect the calculated dose. Based on radionuclide decay, the residual radioactivity, and corresponding dose, are reduced by about a factor of two every 30 years. However, the dose assessment for the excavation scenario conservatively assumes that the concrete fill is excavated immediately after the area is released for unrestricted use.

The NRC evaluated potential scenarios and exposure pathways for concrete debris such as that which would be encountered after the fill is excavated. This evaluation is described in NUREG-1640, "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities." The scenarios evaluated in NUREG-1640 are based on current American industries and include concrete debris handling, processing, transportation, product use, and

disposal. These scenarios are considered applicable to the excavation of concrete rubble. The landfill scenario in NUREG-1640 is not applicable since the results of the resident farmer dose assessment described above in Section 6.2 are a site-specific replacement for the results of the NUREG-1640 landfill scenario.

In addition to the scenarios described in NUREG-1640, a dose assessment was performed assuming that the excavated concrete is used as the foundation for a house. A generic, bounding, evaluation of the potential direct exposure from excavated concrete was also performed to demonstrate that any other remotely credible reuse scenarios, such as use of concrete for rip/rap, etc., would not result in doses exceeding 25 mrem/yr. The conclusion of the excavation scenario assessments is that the rubble fill scenario is more restrictive than the excavation scenario, i.e., the rubble fill scenario requires a lower DCGL, and lower residual contamination levels, than would be required for the excavation scenario.

We believe that the probability of a resident farmer excavating the concrete fill from the building at some time in the future for the purpose of occupying the basements is so remote that it does not meet the threshold of a critical group.

6.3.1 Discussion and Results

The dose assessment for concrete that is hypothetically excavated from its original location is based on a material flow model developed through a review of the concrete recycling industry. Typical end products for recycled concrete debris include use as the base for roads, as a stabilizer for asphalt, and as aggregate in non-structural materials. According to NUREG -1640 recycled concrete is not used as aggregate for structural concrete, such as that used in houses and buildings.

Since the residual contamination is predominantly in the first 1 mm of the concrete surface, the surficial dose factors from Table 7.3 of NUREG-1640 were used to illustrate potential dose for the excavation scenario. A more direct comparison is provided through the NUREG -1640 mass dose factors as described below. The critical group dose factors were used to be consistent with NRC regulations and guidance.

Table 6 provides the surface contamination levels that correspond to 25 mrem/yr from the various dose scenarios evaluated. These values were derived from Table 7.3 in NUREG-1640, using the critical group values. However, for H-3, Ni-63, Sr-90, and Np-237, the critical group is identified as the landfill scenario. As discussed above, the site specific rubble fill dose assessment in Section 6.2 replaces the generic landfill assessment in NUREG - 1640. The dose factors selected for these radionuclides were the highest mean values for the remaining scenarios in NUREG-1640, Appendix I. A subsequent calculation was performed to estimate the surrogate beta concentrations

(dpm/100 cm^2) that would result from the surface contamination levels listed in Table 6. The results are provided in Table 7. The surrogate beta concentration calculated in Table 7 accounts for the hard-to-detect nuclides, such as Ni-63 and Fe-55, in the same manner as described in the dose assessment for standing buildings (see Section 6.7). This calculation allows for a direct comparison to the concrete rubble fill DCGL provided above in Section 6.2.

As noted in Table 7, the beta surface contamination value corresponding to 25 mrem/yr using the NUREG-1640 clearance scenarios is 6.5E+05 dpm/100 cm^2. The general conclusion to be reached from this evaluation is that the highest building rubble DCGL calculated from the dose assessment for concrete fill is about 12 times lower than the 25 mrem/yr surface contamination levels for low-probability recycle scenario. This assessment verifies that the dose assessment for the concrete rubble fill is the limiting scenario and provides the critical group for demonstrating compliance with the 25 mrem/yr dose limit.

The volumetric dose factors for Cs-137 from NUREG-1640 were also reviewed. The volumetric average from the rubble fill scenario DCGL corresponds to 2.4 pCi/g. The volumetric Cs-137 concentration from NUREG-1640 (critical group) that corresponds to 25 mrem/yr is 109 pCi/g. The rubble fill scenario average is 45 times lower than the NUREG-1640 value.

NUREG-1640 acknowledges that there is considerable uncertainty in the dose assessments for concrete construction debris similar to that expected for the excavation of the rubble. This is to be expected since the potential future use scenarios are relatively uncertain. However, there is sufficient margin to conclude that the concrete rubble fill scenario is much more limiting than the NUREG-1640 scenarios and that the NUREG-1640 excavation scenarios require no further consideration.

Table 6 Excavation Scenario Surface Contamination Levels Derived from NUREG-1640					
Structure Nuclides	**NUREG 1640 Vol 2 Table I. (x)**	**NUREG 1640* mrem/y per pCi/cm²**	**1/x pCi/cm² per mrem/y**	**dpm/100 cm² per mrem/y**	**dpm/100 cm² per 25 mrem/y**
H-3	1	2.74E-08	3.65E+07	8.11E+09	2.03E+11
Fe-55	1	3.00E-07	3.33E+06	7.40E+08	1.85E+10
Co-60	2	2.20E-02	4.55E+01	1.01E+04	2.52E+05
Ni-63	1	3.37E-07	2.97E+06	6.59E+08	1.65E+10
Sr-90	1	1.22E-04	8.19E+03	1.82E+06	4.55E+07
Cs-134	2	1.40E-02	7.14E+01	1.59E+04	3.96E+05
Cs-137	2	5.40E-03	1.85E+02	4.14E+04	1.03E+06
Np-237	1	3.00E-02	3.34E+01	7.41E+03	1.85E+05
* NUREG-1640 Volume 2, Table I mean (Critical Group) value					

Table 7 Excavation Scenario Surrogate Gross Beta Concentration Derived from NUREG-1640						
Structure Nuclides	**Nuclide Fraction**	**NUREG 1640 Derived Concentration**	**Beta Fraction**	**HTDN Fraction/ Concentration**	**Beta Fraction/ Concentration**	**Alpha Fraction/ Concentration**
H-3	1.52E-03	2.03E+11		7.52E-15		
Fe-55	9.01E-03	1.85E+10		4.87E-13		
Co-60	7.08E-02	2.52E+05	7.08E-02		2.81E-07	
Ni-63	6.01E-01	1.65E+10		3.65E-11		
Sr-90	7.14E-04	4.55E+07	7.14E-04		1.57E-11	
Cs-134	6.88E-03	3.96E+05	6.88E-03		1.74E-08	
Cs-137	3.10E-01	1.03E+06	3.10E-01		3.01E-07	
Np-237	<7.57E-5	1.85E+05				
Total	1.00E+00		3.88E-01	3.6969E-11	5.995E-07	4.09E-10
			$DCGL_W$ ==>	6.47E+05	<=== dpm/100 cm²	

Two additional excavation scenarios were evaluated that were not included in the NUREG-1640 assessment. These scenarios are provided as bounding examples of the potential dose from excavated concrete. The first scenario evaluates the use of excavated rubble as fill under a house foundation. This is considered a very unlikely scenario since concrete fill would not be normally be used directly under a house foundation due to potential compaction problems. If it were used as fill, there would certainly be some amount of soil over the top of the concrete. Therefore, the assessment assumes 1 foot of clean soil over the concrete. The second scenario is intended to be a generic evaluation of all credible uses of excavated concrete rubble other than the NUREG-1640 scenarios and the house foundation scenario. Both the house foundation and generic scenario are based on the very conservative assumption that a 100 m^2 wall or floor area of structural surface is demolished, used as rubble fill in an onsite building basement, excavated from the basement at some time in the future, and moved to the location for the house foundation or generic scenario, without any mixing with concrete from the wall or floor areas that were adjacent to the original 100 m^2 area. The concrete pieces are assumed to be randomly oriented, allowing simulation using a uniform geometry.

The dose assessments for the two scenarios were conducted using the Microshield code. The Microshield output files showing all assumptions and results are provided in Appendix D.

The assumptions for the house foundation scenario are listed below:

- House dimensions - 10 m x 10 m
- 3.5 inch concrete foundation
- 1 foot of soil over concrete rubble
- Concrete Density 2.2 g/cm^3
- 5,760 hr/yr occupancy (DandD default)
- 100 m^2 fill under foundation is 0.30 m thick
- 100 m^2 wall or floor area not mixed with concrete from adjacent areas
- Exposure at 1 m from floor surface
- 1 mR/hr is assumed to equal 1mrem/hr

- surface contamination at the DCGL level with random orientation under slab that is modeled as 5.3E-06 uCi/cm^3.

The assumptions for the generic scenario are similar to the foundation scenario except that there is no shielding from a foundation. The hypothetical occupancy time is calculated based on the resulting exposure rate and used as the basis for justifying that no remotely credible scenarios exist, other than the house foundation scenario, that could result in 25 mrem/yr from the excavated concrete:

- Concrete rubble dimensions - 10 m x 10 m
- No shielding
- 100 m^2 concrete area is 0.30 m thick
- Density 2.2 g/cm^3
- Original 100 m^2 wall or floor area not mixed with concrete from adjacent areas
- Exposure at 1 m from floor surface
- 1 mR/hr is assumed to equal 1mrem/hr
- surface contamination at the DCGL level with random orientation under slab that is modeled as 5.3E-06 uCi/cm^3

The results of the house foundation scenario show that the dose would be less than 0.13 mrem/yr if the initial surface activity were at the DCGL level. The results of the generic scenario assessment show that the exposure rate from any remotely possible future use of the concrete such as for rip/rap, a jetty, etc., would result in an exposure rate of 1.8E-03 mR/hr. Assuming a person is present 3 hours per day, 365 days per year, the dose would be less than 2 mrem/year. This is very unlikely considering, among other things, that the climate of Maine would preclude extended occupancy during several months of the year.

It is evident that the very low probability of future use of excavated rubble for rip/rap, jetty, or any other use actually occurring, coupled with the low potential exposure, demonstrate that the generic excavation scenario will not be limiting and does not require further consideration. Both of the excavation scenarios result in lower dose than the rubble fill scenario. Therefore, the assumptions and results of the rubble fill

scenario were used for determining the DCGL and compliance with the 25 mrem/yr unrestricted use criteria. However, the results of the house foundation scenario were considered in the area factor calculations described in Section 6.4 to ensure that no100 m^2 pre-demolition wall or floor area exceeds a surface activity level that would result in 1.0 mrem/yr from the house foundation scenario.

6.4 Concrete Area Factors

Regulatory Guide DG-4006, Section 2.4, "Methods to Evaluate Survey Results," describes an acceptable method for evaluating areas within a given survey unit that exceed the DCGL (elevated areas) using Area Factors (AF's) and the elevated measurement comparison (EMC) (see LTP Section 5). These methods are used for determining the acceptability of elevated areas within a given survey unit, when the survey unit has passed the Sign or WRS test. However, as stated in MARSSIM, Section 8.5.1, the AF's and the EMC should not be considered the primary means for determining the acceptability of a survey unit. This decision is based on the results of the Sign or WRS test.

The AF is an acceptable multiple of the DCGL for a given elevated area, and is a function of the elevated area size. Area factors are dose-based and are calculated using the same dose assessment methods used to determine the DCGL level. Elevated areas contaminated at levels equal to the AF x DCGL will result in 25 mrem/yr assuming that no other contamination is present in the survey unit. The presence of multiple elevated areas, combined with some

existing contamination at uniform levels below the DCGL, are addressed using the methods described in LTP Section 5.6.3.

To understand the bases for the area factor determination, certain aspects of the DCGL dose assessment method are particularly important and are reviewed here. As discussed in Section 6.2.3, the DCGL for concrete rubble fill assumes a concrete volume of 1929 m^3. This is the minimum volume of concrete required to contain the volume of water to supply the annual water needs for domestic and irrigation purposes, and was used in the DUST-MS calculations. In addition, for conservatism, no mixing was assumed to occur between adjacent 1929 m^3 volumes of concrete, i.e., each volume is considered a closed system. This assumption is conservative since the 1929 m^3 volume with the highest average contamination level, after remediation, will be used for determining compliance with the 25 mrem/yr unrestricted use limit. Allowing for mixing between adjacent volumes would reduce the calculated water concentration, and result in a lower dose and higher DCGLs. In fact, mixing between adjacent volumes is likely to be what actually occurs.

Consistent with the conservative assumption that each 1929 m^3 concrete volume is a closed system, the calculated annual exposure rate is not affected if, for example, half of the water in the closed system contains two times the assumed uniform concentration level and the other half

contains zero. This is equally true if, for example, one tenth of the water contains 10 times the uniform concentration level and the remaining nine tenths contain zero. As noted above, the most likely situation is that the water would be essentially uniformly mixed and the concentrations in the water would be relatively uniform throughout the 1929 m^3 regardless of the contamination distribution in the concrete. Nonetheless, the argument is important to calculating AF's for concrete rubble fill since it demonstrates that the distribution of the residual radioactivity on the pre-demolition concrete surfaces associated with a given 1929 m^3 concrete volume does not affect the annual dose as long as the total inventory is conserved. Total inventory is conserved if the inventory in the volume does not exceed that which would be allowed if the pre-demolition wall or floor surface areas were uniformly contaminated at the DCGL level. The method to ensure that the total inventory is conserved is the test for multiple elevated areas described in Section 5 (using the $AF_{rubble\ fill}$ as defined below). It can also be demonstrated by using instruments capable of integrating the inventory over the survey unit such as in-situ gamma spectrometry or scanning devices capable of continuous data collection.

Note that the AF calculation provided below applies to the rubble fill scenario only. The dose assessment for the excavated concrete scenario must also be considered when determining AF's for concrete rubble. This is considered in a subsequent discussion.

The technically acceptable AF for the concrete rubble fill scenario ($AF_{rubble\ fill}$) provided below is based on conserving total inventory and complying with the Fundamental DCGL. However, this is not the AF that is proposed for the MY site. The AF proposed is designated "$AF_{concrete}$" in a subsequent paragraph. The $AF_{concrete}$ area factor is more conservative than the $AF_{rubble\ fill}$ area factor, as explained below, and requires lower levels of activity per elevated area than would technically be allowed based on the rubble fill dose assessment. However, it is important to note that the $AF_{rubble\ fill}$ will be used in the calculation for determining the acceptability of multiple elevated areas, , i.e., the EMC, as described in Section 5.6.3. The activity allowed to remain in elevated areas will be determined using the $AF_{concrete}$. The $AF_{rubble\ fill}$ will only be used in the EMC to ensure that total inventory is conserved.

$$AF_{rubble\ fill} = (1929\ m^3)(\text{survey unit } m^2/m^3) / (\text{elevated area size in } m^2)$$

The $AF_{rubble\ fill}$ applies to the concrete rubble fill scenario. The $AF_{concrete}$ calculation was performed to account for the house foundation excavation scenario to ensure that the average contamination level in any pre-demolition 100 m^2 wall or floor area is below the surface activity level that corresponds to less than 1.0 mrem/yr in the house foundation assessment. Since the dose from the house foundation scenario is 0.13 mrem/yr at the DCGL level, any 100 m^2 area could be less than 6 times the DCGL and still result in less than 1 mrem/yr. A dose of 1 mrem/yr was selected to ensure conservatism as compared to the 25 mrem/yr unrestricted use limit. The $AF_{concrete}$ for elevated areas that are less than 100 m^2 in size are calculated to ensure that, regardless of the elevated area size, the exposure rate will be equal to or less than that which corresponds to 1.0 mrem/yr in the house foundation scenario. This is very conservative

since as the elevated area size is decreased, the dose will decrease assuming that the 100 m^2 house floor area is randomly occupied. Based on this assumption, the dose from a 1 m^2 floor area would be less than 0.01 mrem/yr.

The $AF_{concrete}$ calculations are made starting with the baseline assumption that the average contamination level in any 100 m^2 pre-demolition wall or floor area is less than 6 times the DCGL level. This contamination level corresponds to 1.0 mrem/yr in the hypothetical house foundation scenario. The Microshield code was used to determine the $AF_{concrete}$ for other elevated area sizes by ensuring that the exposure rate does not increase for any elevated area size. The results are provided in Table 8 below. Although Table 8 stops at 100 m^2, larger areas may be calculated as long as the exposure rate does not exceed 1mrem/yr using the assumptions in the house foundation scenario. The $AF_{concrete}$ values listed in Table 8 are more conservative than would be technically acceptable under the rubble fill scenario. For example, using the $AF_{rubble\ fill}$ area factor calculation described above and the survey unit size for the containment building (4,436 m^2) it is seen that the 100 m^2 $AF_{concrete}$ is at least 7 times less and the 1 m^2 $AF_{concrete}$ is at least 158 times less.

Table 8
Area Factors For Concrete Fill ($AF_{concrete}$)

Area Size (m^2)	0.5	1	2	4	6	8	12	16	20	30	60	80	100
Area Factor	52	28	16	10	9	7	7	6	6	6	6	6	6

6.5 Activated Concrete Rubble Fill DCGL

The rubble fill DCGL's calculated above pertain to structure surface contamination. The results of laboratory analyses performed by Brookhaven National Laboratory indicated that there is essentially no leaching from activated concrete. This conclusion is not unexpected since the radionuclides generated through activation are an intrinsic part of the concrete structure and are not subject to diffusion as is the case for surface contamination. The long-term degradation of activated concrete is not considered a significant pathway, especially when radioactive decay is accounted for. Since the activated material leach rate is essentially zero, the Building Occupancy DCGL's listed in Table 15 will be used.

6.6 Soil

The dose from soil was evaluated using the DandD screening values. The critical group selected is the resident farmer as recommended in NRC guidance. The following pathways will be evaluated:

- Ingestion of drinking water from an onsite well,

- Ingestion of plant products irrigated from an onsite well,
- Ingestion of plant products grown in contaminated soil,
- Ingestion of animal products grown onsite (after the animals ingest contaminated drinking water, plant products, and soil),
- External exposure while indoors and outdoors,
- Ingestion of soil tracked indoors,
- Ingestion of fish from a contaminated surface-water source,
- Inhalation of resuspended soil while indoors and outdoors, and
- Direct ingestion of soil

6.6.1 Radionuclide Inventory and Derived Concentration Guideline Values

The radionuclide mixture of the residual contamination in the soil at the MY facility is presented in Table 9. The soil DCGL values provided in Table 9 are the NRC Screening values published in the Federal Register 12/7/99.

Table 9 below presents the soil radionuclide mixture, observed radionuclide fraction and associated DCGL's for soil. The observed fraction represents the nuclide fraction on November 1, 1999. The original soil mixture was comprised of 27 radionuclides. Twenty of these radionuclides were reported at Minimum Detectable Activity (MDA) values. The activity level of these radionuclides are below the analytical limit of detection. Two of the MDA level results were for radionuclides that due to their short half lives (^{58}Co and ^{242}Cm) are no longer present. The remaining 18 radionuclides reported at MDA levels were removed from the mixture either because they have not been observed above MDA levels over the history of the facility or, in accordance with Regulatory Guidance DG-4006 Section 2.9, the nuclides are likely to contribute to less than ten percent of the total effective dose equivalent (TEDE) and may be ignored. Seventeen of the reported MDA level radionuclides result in contributing less than 7.6% of the TEDE. The remaining radionuclide in this group, ^{129}I, has not been observed in historical, characterization or continued characterization analysis results at concentrations above the MDA levels and is therefore deemed not present in the mixture. Varying sample MDA values reported for this radionuclide range from 0.28 to 1.2 pCi/g. The variation is due to sample size used by the vendor laboratory for this individual analysis, subsequent solution dilution values used in the process and sample interference.

Table 9 Soil Radionuclide Mixture and DCGL Values		
Radionuclide	**Observed Fraction**	**DCGL (pCi/g) 0.90 Quantile**
^{3}H	2.62E-02	1.1E+02
^{60}Co	1.20E-02	3.8E+00
^{63}Ni	4.04E-02	2.1E+03
^{137}Cs	9.02E-01	1.10E+01
^{234}U	9.40E-03	1.3E+01
^{235}U	1.89E-04	8.0E+00
^{238}U	9.90E-3	1.4E+01

a. Soil Radionuclide DCGL Determination

For determining the soil radionuclide DCGL the concentration of each nuclide will be compared to the respective $DCGL_W$ referenced in 6.6.1. The sum of these concentrations relative to each radionuclide's $DCGL_W$ will be used as the primary method using the unity rule as mentioned in Regulatory Guidance DG-4006 and provided in Chapter 11 of NUREG 1505. Section 6.6.1.b presents the equation. In conjunction with the above, the unity rule for hard to detect radionuclides (HTDN) will be addressed by adjusting the ^{137}Cs $DCGL_W$ as described in section 6.6.1.b below. This methodology is consistent with recommendations of NUREG 1505 Chapter 11.

Radionuclides observed in characterized soil samples at the MY facility originated from incident related events. The soil samples relating to these events were analyzed following the events and samples were collected from the areas during both initial and continuing characterization studies. For soils in which the ratio of the radionuclides is well established, correlation to a surrogate radionuclide (^{137}Cs) is an acceptable method. The surrogate radionuclide method will be used to account for hard to detect radionuclides in the soil mixture.

b. Using the Unity Rule for the Soil Radionuclide DCGL

The soil radionuclide $DCGL_W$ is determined using equation 11.1 of NUREG-1505.

$$DCGL_W = \frac{C_1}{D_1} + \frac{C_2}{D_2} + \frac{C_3}{D_3} + \ldots + \frac{C_n}{D_n} < 1.0(unity)$$

Where:

C1 = Concentration of radionuclide i

D_1 = $DCGL_W$ of radionuclide i

$R_{1\ to\ n}$ = are the individual radionuclide ratios to ^{137}Cs (see below)

And,

for the hard to detect nuclides (HTDN) in the mixture, the ^{137}Cs DCGL is adjusted using the ratios in Table 10 and the equation below (NUREG 1505, equation 11-4) modified for the HTDN nuclides only.

To adjust ^{137}Cs for HTDN:

$$D_{HTDN} = \frac{1}{\frac{1}{D_1} + \frac{R_2}{D_2} + \frac{R_3}{D_3} + \ldots + \frac{R_n}{D_n}}$$

$$^{137}Cs_{(HTDN)} = \frac{1}{\frac{1}{11.0} + \frac{0.0291}{110} + \frac{0.0448}{2100} + \frac{0.0104}{13} + \frac{0.00021}{8} + \frac{0.011}{14}} = 10.78 pCi\ per\ g$$

Then,

10.78 pCi/g is the Adjusted $DCGL_W$ for ^{137}Cs to account for the HTDN radionuclides in the soil mixture when combined with the Unity rule.

c. Soil Radionuclide Correlation to ^{137}Cs as the Surrogate Radionuclide DCGL

For investigations, Cs-137 is used as the surrogate radionuclide for all radionuclides in the soil. Table 10 presents the soil radionuclide mixture and the individual radionuclide to Cs-137 ratio. The derived Cs-137 ratio (10.38pCi/g) represents the nuclide concentration on November 1, 1999.

Table 10 Soil Radionuclide Inventory, Mixture Fraction And Correlation Factor to ^{137}Cs		
Radionuclide	**Radionuclide Fraction**	**Correlation (ratio) to ^{137}Cs**
^{3}H	2.62E-02	2.91E-02
^{60}Co	1.20E-02	1.33E-02
^{63}Ni	4.04E-02	4.48E-02
^{137}Cs	9.02E-01	1.00E+00
^{234}U	9.40E-03	1.04E-02
^{235}U	1.89E-04	2.10E-04
^{238}U	9.90E-03	1.10E-02

Using the values provided in Table 10, the following calculation is used to determine the Cs-137 surrogate DCGL. This calculation is found in NUREG-1505, Equation 11-4. The D_1 term is the DCGL for Cs-137 and D_{total} is the surrogate Cs-137 DCGL:

$$D_{total} = \frac{1}{\frac{1}{D_1} + \frac{R_2}{D_2} + \frac{R_3}{D_3} + \ldots + \frac{R_n}{D_n}}$$

Where,

D_i is the DCGL for radionuclide *i* and R_i is the ratio of the concentration of the radionuclide *i* to that of the surrogate radionuclide.

Using the values provided in the above tables, the derived Surrogate DCGL is:

10.38 pCi/g Cs-137 (11/1/99)

d. Soil Area Factors

As discussed above, the DandD screening values were used for soil DCGL's since they have been approved by NRC and are conservative. However, a disadvantage of using the DandD DCGL's is that the DandD code does not allow the adjustment of exposure area to calculate area factors. To address this issue, RESRAD was used to generate conservative area factors for soil.

It is recognized that there are differences between the RESRAD and DandD codes that could affect the development of area factors using RESRAD given

that the DCGL's are based on DandD. To account for the potential affect of these differences in a conservative way, only the direct radiation pathway was considered in the area factor development. The direct radiation pathway is a known physical process that can be sufficiently represented by RESRAD in a way that is applicable to DandD.

The major differences in the two codes arise when pathways other than direct radiation are considered. To ensure a conservative area factor, the non-direct radiation pathways were not included in the calculation. This is a conservative approach since the area factors for the non-direct radiation pathways are expected to increase at a higher rate than the direct radiation pathway due to physical limitations such as the land area required to raise livestock. In addition, for relatively small land areas, such as 20 m^2 or less, when the area factors begin increasing more rapidly, the higher of either the direct radiation or the ingestion/inhalation pathways must be selected because there is not enough land for both pathways to occur simultaneously. For the radionuclides in the MY soil, direct radiation is the predominant pathway in both the DandD and RESRAD codes. For all of the reasons described above, using only the direct radiation pathway is a conservative approach to calculating the area factors. It is assumed that the relative change in the RESRAD direct pathway, with appropriate parameter modifications, is sufficiently similar to that which would be expected in the DandD code to justify the use of RESRAD for the area factor calculations.

A unit concentration of Cs-137, i.e., 1 pCi/g, and the corresponding concentration of Co-60 for the MY soil, i.e., 0.023 pCi/g, were used as input into RESRAD. To simulate the DandD code to the maximum extent possible, the RESRAD input parameters were modified to include the values listed in Table 4, as was done for the rubble fill dose. A series of RESRAD runs were made starting with an area of 2000 m^2 and reducing the area to a minimum of 1m^2. The area factor was calculated by dividing the direct radiation dose at 2000 m^2 by the direct radiation dose at the respective areas listed in Table 11.

Table 11 lists the soil area factors. It is important to note that the RESRAD runs used to generate the area factors showed that, as expected, the area factors for non-direct radiation pathways increased faster than the area factors for the direct radiation pathways. For example, for a 20 m^2 area, the area factor for direct radiation was 1.7 and the area factor for the non-direct pathways was 150. Under reasonable assumptions, the DandD code would be expected to produce the same relative results as RESRAD. This confirms that the method for determining the area factor is conservative. Although the area

Table 11 Area Factors for Soil											
Area (m^2)	1.0	2.0	6.0	10	20	60	100	300	600	1000	2000
Area Factor	15.6	8.7	4.1	3.0	2.2	1.5	1.4	1.1	1.1	1.0	1.0

factors in Table 11 are not linear as a function of area, a linear interpolation between the values listed is acceptable within the ranges listed and will be used.

6.7 Standing Buildings

If standing buildings remain on the site for some purpose, for example, to support the implementation of the re-power option, the NRC screening surface contamination limits from the DandD screening values will be used.

6.7.1 Standing Building Radionuclide Mixture and Derived DCGL

The radionuclide mixture for concrete surfaces was determined through the analysis of 42 concrete core samples. The large sample population and areal distribution of the 42 cores provides a good representation of the site. The core analyses evaluated 24 radionuclides that could have potentially been present in the concrete. Seventeen of these nuclides were reported as less than MDA values of which two, ^{58}Co and ^{242}Cm were removed from the mix due to their short half life. Fourteen of the remaining 15 radionuclides reported as below MDA were removed from the mixture either because they have not been observed above MDA levels over the history of the facility or, in accordance with Regulatory Guidance DG-4006 Section 2.9, the nuclides are likely to contribute to less than ten percent of the total effective dose equivalent (TEDE) and may be ignored. Fourteen of the reported MDA level radionuclides contribute less than 6.1% of the TEDE. The remaining MDA radionuclide, ^{237}Np has been reported in past operational waste stream samples and is conservatively maintained in the mixture at the MDA level. Note that the characterization sampling did not report this radionuclide above MDA. Recent waste stream analyses also report less than MDA for this radionuclide.

Table 12 lists the concrete structure surface radionuclide mixture and the corresponding DCGL values. The DCGL values are taken from the screening table in SECY-98-051, "Guidance in Support of Final Rule on Radiological Criteria for License Termination," March 16, 1998. Specifically, Table 4.6 in the report "Review of Parameter Data for the NUREG/CR5512 Building Occupancy Scenario and Probability Distribution for the DandD Parameter Analysis," that was provided with SECY-98-051. The DCGL's used are for the 90 percentile ($P_{Crit} = 0.10$).

Table 12 Structure Surface Radionuclide Mixture and DCGL Values		
Radionuclide	**Observed Fraction**	**DCGL* (in dpm/100 cm²)**
^{3}H	1.52E-03	1.24E+08
^{55}Fe	9.01E-03	4.50E+06
^{60}Co	7.08E-02	7.05E+03
^{63}Ni	6.01E-01	1.82E+06
^{90}Sr	7.14E-04	8.71E+03
^{134}Cs	6.88E-03	1.27E+04
^{137}Cs	3.10E-01	2.80E+04
^{237}Np	<7.57E-05	2.21E+01

6.7.2 Structure Surface Gross Beta DCGL

A significant fraction of the investigated samples represent the top layer of concrete which comprise the site structures and characterize the expected structure condition. Continued site characterization activities provided 42 additional cores from which the principal gamma emitters (^{60}Co and ^{137}Cs) representative fractions were determined. Seven core samples were submitted for HTDN analysis. The analytical results of these samples augment the current data and provide a sound basis to qualify and adjust current observations. The results of representative surface samples from each structure are used to ensure that appropriate gross beta DCGL values are assigned to the concrete structures that will remain standing after license termination. The equation below is from equation 4.4 of NUREG 1575.

$$DCGL_{surface} = \frac{F_{detectable}}{\frac{f_1}{D_1} + \frac{f_2}{D_2} + \ldots \frac{f_n}{D_n}}$$

Where:

$DCGL_{surface}$ = The acceptable level of measurable gross beta levels on structure surfaces after adjustment for hard to detect radionuclides (dpm/100 cm²).

$F_{detectable}$ = Detectable nuclide fraction (sum of the detectable surface radionuclide activity divided by the sum of the total radionuclide activity).

f_i = Fraction of total activity contributed by each radionuclide.

D_i = DCGL for each radionuclide.

Table 13 presents the derived DCGL for structure surface gross beta radioactivity. An item that affects the gross activity DCGL is the fraction of hard-to-detect nuclides (HTDN) in the mixture (62 %). The data from Table 10 is used to develop the gross beta activity $DCGL_W$.

6.7.3 Structure Surface Area Factors

Based on the characterization data the predominant gamma emitting radionuclide observed in the concrete structure is ^{137}Cs. As presented in previous tables this radionuclide accounts for80 percent of the measurable radionuclide beta emitters present in the mixture. Table 14 provides the RESBUILD Version 2.36 derived area factors.

The DandD code does not have the capability of calculating dose for various structure surface areas. Therefore, RESBUILD was used to conservatively calculate area factors. RESBUILD is acceptable for calculating area factors for DandD-based DCGL because the physical and technical bases for calculating direct gamma exposure are relatively straightforward and well known. However, it is possible that area factors for the non-gamma pathways may significantly differ between RESBUILD and DandD because of differences in the models for inhalation, ingestion, etc. To ensure that these potential differences do not affect the area factors calculations, only the RESBUILD direct gamma pathway is used in the area factor calculation. This is a conservative approach because the non-gamma area factors are expected to increase at a faster rate than the gamma pathways. This is illustrated in the generic area factor tables in NUREG-1575, Section 5.5.2.4. If the non-gamma pathways were included, the area factors would be higher, i.e., more activity would be allowed to remain in elevated areas.

Specific RESBUILD parameters were adjusted to account for some DandD site-specific parameters related to the occupational pathway scenario. The RESBUILD default room floor area was modified from 36 m^2 to 100 m^2. The exposure duration default (365 days) was maintained. The indoors time fraction was set to 0.267 which results in 97.46 total days exposure, the same as the DandD default value. The breathing rate in RESBUILD was set to 11.2 m^3 per day (DandD default value). The

area source geometry was used.

Table 13 Derived Gross Beta Activity Surface DCGL Adjusted for Hard To Detect Radionuclides							
Nuclide	**Nuclide Fraction**	**Ratio to ^{137}Cs**	**DCGL**	**Beta Fraction**	**HTDN Frac/DCGL**	**Beta Frac/DCGL**	**Alpha Frac/DCGL**
^{3}H	1.5E-03	4.9E-3	1.24E+08		1.23E-11		
^{55}Fe	9.0E-03	2.9E-02	4.50E+06		2.00E-09		
^{60}Co	7.1E-02	2.3E-01	7.05E+03	7.08E-02		1.00E-05	
^{63}Ni	6.0E-01	1.9E+00	1.82E+06		3.30E-07		
^{90}Sr	7.1E-04	2.3E-03	8.71E+03	7.14E-04		8.19E-08	
^{134}Cs	6.9E-03	2.2E-02	1.27E+04	6.88E-03		5.42E-07	
^{137}Cs	3.1E-01	1.0E+00	2.80E+04	3.10E-01		1.11E-05	
^{237}Np	7.6E-05	2.4E-04	2.21E+01				3.43E-06
Total				3.88E-01	3.32E-07	2.17E-05	3.43E-06
			DCGL	1.52E+04	dpm/100 cm^2		

The area factors in Table 14 are based on evaluations of Co-60, Cs-137, and Cs-134 at the relative percentages found in the contaminated concrete.

Table 14 Area Factors for Standing Structures											
Area (m^2)	1	2	4	8	12	16	20	24	36	50	100
Area Factor	12.6	7.1	4.2	2.8	2.2	1.9	1.7	1.6	1.4	1.2	1.0

6.8 Standing Building Activated Concrete

Buildings that may remain standing do not contain activated areas. However, for completeness, the calculations are provided. Concrete cores collected from the In-Core Instrumentation (ICI) sump were submitted for gamma spectroscopic analysis of activation products. The uniformity of the activation product concentrations throughout the ten slices examined is indicative of activation processes.

6.8.1 Activated Concrete Derived Gross Beta $DCGL_w$

The derived gross beta $DCGL_w$ for activated concrete is provided in Table 15. All radionuclides in the mixture are beta emitters however, the decay mode for ^{152}Eu is complex. Two isometric states of Eu-152 are very short lived and are no longer present in the concrete. The mode of decay for the third and long half life state is by both electron capture and beta decay. The branching ratio for the beta decay is 27.9% . This fraction is multiplied by the mixture fraction to derive the Eu-152 beta fraction for Table 15. All other calculations are the same as those described in section 6.7.2. Because ^{152}Eu comprises 58-65 percent of the concrete mixture and only 28.0 percent is directly observable (beta decay), the gross beta $DCGL_w$ is low. The activated concrete $DCGL_w$ was based on the radionuclide mixture decayed to 1/1/03. The 1/1/03 mixture resulted in a lower $DCGL_w$ than using the mixture decayed to the standard date of 11/1/99.

Table 15
Activated Concrete Derived Gross Beta DCGL

Nuclide	Nuclide Fraction	DCGL	Beta Fraction	Beta Frac/DCGL
^{60}Co	2.68E-01	7.05E+03	2.68E-01	3.80E-05
^{134}Cs	2.07E-02	1.27E+04	2.07E-02	1.63E-06
^{152}Eu	6.52E-01	1.27E+04	1.82E-01	5.14E-05
^{154}Eu	5.90E-02	1.15E+04	5.90E-02	5.14E-06
Total	1.00E+02		5.30E-01	9.61E-05
		DCGL	5.51E+03	dpm/100 cm^2

6.8.2 Activated Concrete Area Factors

The is no need for development of area factors or graphs for activated concrete, the associated gamma energies are comparable to those used for structure surfaces. Tables 14 of Section 6.7.3 may conservatively be used.

6.9 Embedded Piping / Remaining Systems (including buried piping)

Most systems and components are expected to be removed during decommissioning. Repowering may reserve some systems or components for future use. The amount of embedded piping currently on the site is estimated to be on the order of 1000 feet. Contaminated and embedded piping is expected to be removed. Any remaining embedded piping, systems or components will be surveyed to the default NRC surface contamination

criteria for standing buildings.

6.10 References

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- NRC Draft Guidance, Preliminary Guidelines for Evaluating Dose Assessments in Support of Decommissioning.

APPENDIX A

DUST-MS Input And Output Files

APPENDIX B

RESRAD Report for Rubble Fill Dose Assessment

APPENDIX C

RESRAD Media Concetration Report for Rubble Fill Dose Assessment

APPENDIX D

Microshield Runs for Excavation Scenario

MAINE YANKEE

LTP SECTION 7

UPDATE OF SITE- SPECIFIC DECOMMISSIONING COSTS

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7.0 UPDATE OF SITE- SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

In accordance with 10 CFR 50.82(a)(9)(ii)(F) and the guidance of Regulatory Guide 1.179, the site specific cost estimates and funding plans are provided. Regulatory Guide 1.179 discusses the details of the information to be presented.

TheLicense Termination Plan (LTP) must:

Provide an estimate of the remaining decommissioning costs, and compare the estimated costs with the present funds set aside for decommissioning. The financial assurance instrument required by 10 CFR50.75 must be funded to the amount of the cost estimate. If there is a deficit in the present funding, the LTP must indicate the means for ensuring adequate funds to complete the decommissioning.

The decommissioning cost estimate should include an evaluation of the following cost elements. The section of this submittal with the detailed discussion of each element is noted in parentheses.

- Cost assumptions used, including a contingency factor
  (Appendix 7.1, Sections 3.1, 3.2, & 3.3)

- Major decommissioning activities and tasks
  (Appendix 7.1, Section 4)

- Unit cost factors
  (Appendix 7.1, Appendices A, B,&C)

- Estimated costs of decontamination and removal of equipment and structures
  (Section 7.2.2 and Appendix 7.1, Appendix C)

- Estimated costs of waste disposal, including applicable disposal site surcharges
  (Appendix 7.1, Section 5)

- Estimated final site survey costs
  (Appendix 7.1, Appendix C)

- Estimated total costs
  (Tables 7.1 & 7.2)

The cost estimate should focus on the remaining work, detailed activity by activity including costs of labor, materials equipment, energy, and services.

MYAPC has docketed a site specific cost estimate prepared by TLG Services in accordance with 10 CFR50.82(a)(8)(iii). (Reference Letter: G. Zinke, MYAPC to USNRC; 10 CFR50.82(a)(8)(iii) Site Specific Decommissioning Cost Estimate and PSDAR Update; MN-98-65, dated November 3, 1998). This cost estimate focuses on all decommissioning costs from 1997 through 2023, with the assumed final removal of all fuel from the site.

MYAPC has received an order from the Federal Energy Regulatory Commission (FERC), dated June 1, 1999 and effective August 1, 1999, concerning the recovery of decommissioning costs.

Maine Yankee has established a unique approach to decommissioning costs in that it has awarded a fixed price contract with the Decommissioning Operations Contractor (DOC)-Stone & Webster (S&W). This fixed price contract includes the scope of the decommissioning, radiological remediation, waste disposal, major component removal and disposal, final site surveys, remediation activities, and ISFSI construction and fuel transfer. In addition MY has completed the initial radiological site characterization and an extensive asbestos removal program. MYAPC's current financial planning cost estimate, which is used to demonstrate financial assurance in Section 7.3, includes costs incurred since 1997 plus the fixed price contract and projected MY costs through 2023.

7.2 Decommissioning Cost Estimate

7.2.1 Cost estimate previously docketed in accordance with 10CFR50.82

As stated earlier, MYAPC has docketed a site specific cost estimate prepared by TLG Services in accordance with 10 CFR50.82(a)(8)(iii). (Reference Letter: G. Zinke, MYAPC to USNRC; 10 CFR50.82(a)(8)(iii) Site Specific Decommissioning Cost Estimate and PSDAR Update; MN-98-65, dated November 3, 1998).

This section provides the result of and the basis for a cost estimate for decommissioning prepared by MYAPC with the assistance of TLG. The TLG estimate in its entirety, Document No. MO1-1258-002 which was previously docketed with the USNRC, is appended to this report as Appendix 7.1. This estimate was prepared using unit cost factors and site specific and schedule driven considerations in accordance with the methodology suggested in AIF/NESP-036, "Guidelines to Producing Decommissioning Cost Estimates". PSDAR Page 15, revision 1, summarizes decommissioning costs and is appended to this report as Table 7-1. This table presents costs derived from the TLG estimate but organized in accordance with Reg. Guide 1.179 guidance.

MYAPC's current financial planning decommissioning cost estimate is based on the fixed price contract with S&W to dismantle and decontaminate the plant plus MYAPC's budget for decommissioning costs including contingency, remediation and ISFSI engineering, licensing, construction and operation. These costs totaling $541M are presented in mid-1998 dollars.

This current financial planning cost estimate is consistent with the TLG report (within 3%) of $508M in mid-1997 dollars or $527M in mid-1998 dollars when escalated at 3.8%. Therefore, MYAPC is relying on the TLG cost estimate for this submittal and will continue to monitor future estimates to ensure that costs be within 20% of the TLG estimate.

7.2.2 Radiological Decontamination Costs

The costs for radiological decontamination activities, estimated to be $343.3M in mid-1997 dollars, are summarized in Table 7-1. Consistent with current NRC policy, MY decontamination costs consider those costs only to be associated with normal decommissioning activities necessary for termination of the Part 50 license and release of the site for unrestricted use. It does not include costs associated with spent fuel management or the disposal of non-radioactive materials and structures beyond that necessary to terminate the Part 50 license.

7.2.3 Spent Fuel Management Costs

Spent fuel management costs are separately summarized in Table 7-1 and estimated to be $128.7M in 1997 dollars. These costs include ISFSI engineering, licensing, construction, and operation until possession of the spent fuel is transferred to the Department of Energy (DOE) which is assumed in this estimate to begin in 2018. The cost of decommissioning the ISFSI facility is included in the fuel management costs.

7.2.4 Site Restoration (Remediating)

The cost of site restoration is estimated to be $35.7M in mid-1997 dollars as shown in Table 7-1. This cost includes demolition of non-radiological affected buildings and costs associated with non-radiological remediation required by Federal and State agencies, e.g. Resource Conservation and Recovery Act (RCRA) closure, asbestos disposal, etc.

7.2.5 Summary of MY Decommissioning Cost Estimate

The TLG cost estimate, as provided in Appendix 7.1 of this report, is the basis for the company's decommissioning cost estimate and is provided in a format consistent with regulatory guidance with additional detail. MYAPC will continue to monitor that future financial planning cost estimates, which include a significant portion of fixed costs, continue to be within 20% of the TLG estimate. Finally, MYAPC has used its current financial planning cost estimate of $541M in mid-1998 dollars (which exceeds the formal cost estimate of $527M in mid-1998 $) which includes both the fixed price contract and remaining MY projected decommissioning costs through 2023, to project Decommissioning Trust Fund (DTF) balances and to demonstrate financial assurance. MYAPC's projection of DTF balances are attached as Table 7-2.

Table 7-1

M YAPC
Summary of Decommissioning Costs [1] (thousands of 1997 dollars)
Key Tasks/Milestone

Plant Radiological Decontamination	
Staffing	$91,128
LLW Burial	$64,816
Equipment Removal	$44,310
LLW Packaging and Shipping	$16,663
Decontamination Activities	$ 6,376
Contingency	$60,265
Other Costs[2]	$59,719
Subtotal	**$343,279 [3]**
Spent Fuel Management	
Staffing and Security	$33,189
Property Taxes	$25,445
Construction Costs	$52,249
NRC and State Fees	$10,093
Insurance	$ 3,018
Other Costs	$ 4,683
Subtotal	**$128,677 [3]**
Site Restoration (Remediating)	
Licensing Termination Survey	$10,701
Major Component Removal	$10,805
Close-out activities	$ 3,222
Demolition of site buildings	$10,973
Subtotal	**$35,701 [3]**
Total Decommissioning Costs Estimates	**$508,000 [3]**

Notes:
(1) Prompt Decommissioning Technique (DECON)
(2) Other costs include insurance, property taxes, energy, NRC and State fees, etc.
(3) Sums may not be exact due to rounding to nearest thousand

7.3 Decommissioning Funding Plan

As stated above, MYAPC has used its current financial planning estimate of $541M and projections of decommissioning collections which considers the recent FERC rate case settlement to project DTF balances and to evidence financial assurance along with other funding avenues available to the Company as described below. (See Table 7-2.).

Table 7-2 column 2 combines the annual projections for the costs specifically associated with plant (radiological) dismantlement, spent fuel management, and site-restoration as "Escalated Expenditures".

MYAPC is currently collecting decommissioning funds through its Power Contracts and Amendatory Agreements under FERC regulation. These contracts have been filed with the FERC. Table 7-2 column 1 identifies the decommissioning funds currently being collected and those projected to be collected under the contracts and includes the funding of radiological decommissioning, spent fuel management, and remediating.

As a result of the FERC order dated June 1, 1999 and effective August 1, 1999,MYAPC has agreed to file with the FERC no later than January 1, 2004 for the purpose of examining any further rate adjustments specifically, although not limited to the future cost of spent fuel storage management. MYAPC expects that case to determine any adjustments to decommissioning collections. MYAPC's plan to fully fund all decommissioning costs and spent fuel storage costs by 2008 (MYAPC's current operating license expiration date) would require an increase in annual collections from approximately $24M to approximately $28M.

As a result of State of Maine Legislative action effective September 18, 1999, MYAPC has access to it's state mandated Spent Fuel Disposal Trust (SFDT). As of December 31, 1998 the SFDT balance was $129,400,000. These funds, are separate and distinct from the DTF pursuant to 10CFR50.75 and 10CFR50.82. Effective October 1, 1999,MYAPC is permitted by State law to withdraw funds from the SFDT to meet expenditures for interim spent fuel storage costs and to offset those interim spent fuel storage costs already incurred byMYAPC. Expenditures from the SFDT are incorporated in the FERC Rate Settlement. Table 7-2 column 5 identifies the estimated costs associated with the construction of an ISFSI which will be funded from the SFDT.

As of December 31, 1998, the accrued MY Decommissioning Trust Balance was $212.7 million. This balance includes amounts in the trust for all decommissioning costs including remediating and long term spent fuel management as well as decommissioning as defined in 10CFR50.75 and the PSDAR.

Also note that as of December 31, 1998, MYAPC had incurred $42.4 million of decommissioning expenditures.

As indicated in Table 7-2 column 4, the DTF pursuant to 10CFR 50.75 is sufficient, together with current FERC approved collections and an assumed rate increase in 2004, to cover all of the expenditures related to decommissioning and spent fuel management except for ISFSI capital related expenditures which are being funded from the State mandated SFDT Fund as shown in Table 7-2 column 5. Also refer to Section 7.4 for a description of other financing options available to MYAPC.

MYAPC

Decommissioning Trust - Cash Summary-Projected Balances

Decommissioning Cost $541 (excl. Texas) @ 3.8%: FERC Rate Settlement Effective 8/1/99:
Legislative Action Effective 10/1/99; and New Rate Filing 1/1/04

TABLE 7-2

(Dollars in Thousands)

DECOMMISSIONING TRUST

	(Col. 1)	(Col. 2)	(Col. 3)	(Col. 4)	(Col. 5)
Year	**Annual Decomissioning Cash Contributions**	**Escalated Expenditures***	***After-Tax trust* Earnings and Adjustments**	**Decomissioning Trust Balance**	**ISFSI Capital Expenditures from Spent Fuel Disposal Trust**
1997				$201,422	$0 (1)
1998	$33,901	$43,756	$19,746	$211,314	9,130
1999	33,878	83,327	8,495	170,361 (2)	27,500
2000	24,393	102,444	4,597	96,906 (2)	16,060
2001	24,393	66,550	2,654	57,404 (2)	9,570
2002	24,393	62,284	1,346	20,858 (2)	335
2003	24,393	39,325	469	6,395 (2)	
2004	27,609	29,972	252	4,484 (2&3)	
2005	27,901	5,970	(1,309)	24,906	
2006	27,901	5,891	(302)	46,614	
2007	27,901	6,111	852	69,256	
2008	27,974	6,319	2,027	92,938	
2009	0	4,934	6,201	94,205	
2010	0	5,116	6,322	95,411	
2011	0	5,305	6,441	96,548	
2012	0	5,501	6,560	97,607	
2013	0	5,704	6,677	98,580	
2014	0	5,916	6,148	98,812	
2015	0	6,135	4,787	97,464	
2016	0	6,363	4,714	95,815	
2017	0	6,600	4,626	93,841	
2018	0	6,846	4,521	91,516	
2019	0	7,101	4,398	88,814	
2020	0	7,366	4,257	85,704	
2021	0	7,642	4,094	82,156	
2022	0	7,928	3,910	78,138	**$62,595.00**
2023	0	80,044	1,906	0	
Total	**$304,639.00**	**$620,451.00**	**$114,390.00**		

*Excludes ISFSI Capital-related expenditures

Notes:
1. The Spent Fuel Disposal Trust Fund Balance as of December 31, 1998 was $129.4M.
2. Includes adequate reserve for SAFSTOR as discussed in Section 7.4
3. Annual decommissioning collections increase from $24M to $28M with approval effective 1/1/04. Tax Billing Change (Column 3) where Trust
pays for all decommissioning income taxes commencing 1/1/04.

7.4 Reserve Requirements

10CFR50.82(8)(i)(B) and 10CFR50.82(8)(i)(C) requires that a reserve be maintained in the DTF to accommodate a sudden unexpected delay in decommissioning activities.

All spent fuel is expected to have been transferred to the completed ISFSI and the existing Fuel Storage Building (SFB) is expected to be decommissioned by the end of 2004. Fuel management costs after 2004 will consist of operational costs until 2023 and decommissioning costs associated with the ISFSI.

All D&D activities are scheduled for completion by year end 2004. Assuming a SAFSTOR condition, DTF expenditures would be minimized to D&D activities only. Therefore, MYAPC forecasts sufficient DTF balances should a SAFSTOR condition occur during the period between 1999 and 2004.

After 2004, the majority of expenditures from the DTF are related to the ISFSI. As shown in Table 7-2 column 4, sufficient funding will exist, based on MYAPC's assumption that the DOE will take all responsibility for Spent Fuel storage by 2023.

Finally, as demonstrated in Table 7-2, MYAPC maintains a positive DTF balance for **ALL** $541M of decommissioning expenditures including spending the full contingency. Pursuant to 10 CFR 50.75 and 10 CFR 50.82 regulations, we believe we have demonstrated a financial plan which includes adequate reserves for the entire decommissioning and ISFSI-related costs which therefore meet the requirements for decommissioning costs associated with decommissioning and dismantlement as defined by these regulations. To further strengthen this position, MYAPC has other funding options as listed below:

- Agreements with FERC to file no later than January 1, 2004 for further rate adjustments.
- Adjustments to collections as defined under the power contracts, as amended.
- MYAPC's ability to borrow funds under current financial and DOC agreements.
- MYAPC's ability to defer decommissioning activities in order to reduce DTF expenditures.

APPENDIX 7.1

10CFR50.82(a)(8)(iii) Site-specific Decommissioning Cost Estimate and PSDAR Update

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LTP SECTION 8

SUPPLEMENT TO THE ENVIRONMENTAL REPORT

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8.0 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1 Introduction and Purpose

10CFR50.2 defines "decommissioning" as those activities".... to remove (the reactor) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license." Under this definition, decommissioning does not include the removal and disposal of nonradioactive structures beyond that necessary to allow unrestricted access to the site. However MYAPC has gone beyond this limit to include those activities to return the site for unrestricted use and possible development as an industrial park.

10CFR 51.53, "Supplement to environmental report," requires that a utility applying for authorization to decommission a nuclear power reactor submit a supplement to the environmental report to address the post operating license stage of the facility. This report should reflect any new information or significant environmental change associated with the proposed decommissioning activities. This section of the LTP is intended to satisfy this requirement.

This document supplements MYAPC's - PSDAR discussion providing the reasons for concluding that the environmental impacts associated with the site-specific decommissioning activities are bounded by previously issued environmental impact statement. Additionally MYAPC has submitted revisions to its FSAR in the form of a DSAR. The DSAR includes an entire Section 7.0 for Decommissioning.

8.2 Plant Site and Surrounding Area

8.2.1 Site Description

The MY site is irregularly shaped, consisting of approximately 30 developed acres of the original 740 (1972 Environmental Report, page 9-1). The site is located on a coastal region marked by glacially formed hills and estuaries. The hills generally vary in elevation from 100 - 200 feet above sea level. The estuaries run inland 10 to 20 miles forming long fingers of land between tidal rivers. Glacial action and erosion have left the hills with a gently rolling topography. The site itself varies in elevation from sea level to 80 feet above sea level. The plant proper is founded upon bedrock at elevation 20'. Although bedrock is extensively exposed around the shoreline, it is commonly buried beneath a blanket of unconsolidated non-stratified clay-silt over most of the gently rolling surface of [Bailey] Point. No soluble or cavernous lithologies are believed to exist in the area of the Point, and seismic, drilling, and outcrop evidence suggest that no deep weathering zones occur in the bedrock in the area of the generating facility. Unconsolidated overburden covering Bailey Point consists of a massive-textured -clay-

silt. Occasional sandy lenses and pebbly sand beds lie horizontally within or beneath the clay-silt blanket, and isolated sandy beach deposits are found on hillsides in the general area." (1972 Environmental Report, page 2.4-1)

The irregularly shaped plant site developed approximately 30 acres of the original 740 acres (1972 Environmental Report page 9-1). The Back River forms the eastern edge of the property; Bailey Cove (tidal estuary) runs into the property, creating the peninsula on which the plant is located.

8.2.2 Population Density (Data from MN-96-189)

The site is located in a rural area that has relatively low population density of approximately 90 per square mile within 10 miles of the plant. The population is higher at distances of 30-40 miles (with a slightly lower density in the area between 10 and 20 miles). At 40-50 miles out the population density drops.

The population of the two nearest cities Lewiston @ 39,757 and Auburn @ 24,309 are significantly less than the GEIS model large city population of 64,000 within 29 miles.

The GEIS evaluation considers the closest large city at about 30 miles away with a population of 1,800,000. There are no large cities (pop>1 million) within 100 miles of the MY site. The Maine State 1990 population was 1.2 million and Boston Metro Statistical Area 1990 census pop was 4,171,747.

The GEIS assumes the total population in a radius of 50 miles at 3.52 million, where as the total population within 50 miles of MY is approximately 650,000.

This difference between the MY site and the generic site does not affect the conclusions of the environmental statement.

8.2.3 Land Uses

Land use within 5 miles of the site, is largely home sites, small businesses, summer houses, idle farmland and forest.

The smaller percentage of land used for farming, combined with the low population density, and the commitment to continued application of radiation protection and contamination controls during decommissioning, results in radiologic consequences to the public lower than those calculated in the GEIS.

8.2.4 Geology

Soils map and report Site soils were surveyed and mapped by Robert G. Gerber, Inc. in 1991.

"Seismic activity in New England is small and typified by infrequent shallow focus earthquakes of low magnitude and intensity. ... The closest earthquake to the plant was a magnitude $4.0m_b$ earthquake that occurred on April 17, 1979, approximately 4 miles from the site; ... Based upon a relatively long historical record, on the attenuation of earthquake intensity with distance in the northeastern United States and eastern Canada region, on the geology of the area and on the character of the competent bedrock on which the facility is founded, an estimate of the probable maximum intensity at the site is a Modified Mercalli Intensity V to low Intensity VI, corresponding to a ground acceleration of 0.04g." (UFSAR section 2.5.2, pages 2-39 and 2-41)

Although bedrock is extensively exposed around the shoreline, it is commonly buried beneath a blanket of unconsolidated non-stratified clay-silt over most of the gently rolling surface of [Bailey] Point.... No soluble or cavernous lithologies are believed to exist in the area of the point, and seismic, drilling, and outcrop evidence suggest that no deep weathering zones occur in the bedrock in the area of the proposed generating facilities. Unconsolidated overburden covering Bailey Point consists of a massive-textured -clay-silt. Occasional sandy lenses and pebbly sand beds lie horizontally within or beneath the clay-silt blanket, and isolated sandy beach deposits are found on hillsides in the general area." (1972 Environmental Report, page 2.4-1)

Decommissioning activities will have little of no impact on existing terrain or vegetation in these undeveloped areas

8.2.5 Hydrology

The maximum water levels at the MY due to the probable maximum hurricane are predicted to be at elevation 19.9 feet and elevation 21.4 feet on the plant site and screen well structure, respectively. .... Safety measures have been implemented in the design of the plant regarding this design basis flooding. The screen well is protected up to elevation 22 feet 0 inches, while the floor grade of the principal power station structures is elevation 21feet 0 inches and site grade which varies from elevation 20 to 21 feet should preclude water from entering (UFSAR Section 2.3.3)

"Ground water in the region occurs as free ground water within the clay-silt soil mantle and joints in the underlying bedrock.....Precipitation at the power station site will percolate downward to the water table and then move with the normal ground water

flows toward the adjoining salt water areas. Percolation rates however, are low due to the low permeability of the local soils and limited bedrock jointing.... Water wells in areas adjacent to the site are either dug wells, usually less than 25 feet deep, or drilled wells penetrating the bedrock for depths of 100 feet or more. Such wells are for domestic or farm use. Although adequate for the purpose, their yield seldom exceeds 5-10 gallons per minute for short term pumping and even less for sustained pumping. There are no municipal or other important well water supply systems in the area" (1972 Environmental Report 2.5-1)

8.2.6 Meteorology

" The general climatic regime is maritime with its cool air moving in from the North Atlantic. Of special importance , from an engineering standpoint, are the extremes in annual snowfall for the coastal region: the occasional heavy rains, the coastal storm or "Nor'easter" with its resultant strong winds and heavy rains or snow; and sometimes glaze or "ice storms'." (UFSAR section 2.2.1).

The PSDAR states; " MY will continue to maintain and protect systems and area critical to the storage for the spent fuel." (PSDAR page 9, Section III Description of Planned Decommissioning Activities - Storage of Spent Fuel) [reference NRC Bulletin 94-01).

8.2.7 Aquatic Ecology

NUREG-0586 generic plant model considers the river as a source of drinking water for the larger communities. At MY the river, being saline is not used for drinking water. Since the water is not used for drinking , the radiologic consequences to the public will be lower than those calculated in the generic environmental statement.

The septic tanks at the Bailey and Eaton houses will be abandoned in place. The sewer system lift pumping station and the sewer main discharging from that station are the properties of the Wiscasset Sewer District and will remain to serve the ISFSI. The existing water supply main for town water will be re-termninated and left in place to serve the ISFSI.

The total water use for decommissioning is estimated to be about 18,000 m^3 [4.75 million gallons] for the generic plant in NUREG-0586. Prior to shutdown MY was connected to the local water and sewage service (Wiscasset Water District). The billable cubic feet water usage for the years, 97, 98 and 99 was reviewed, with the highest usage for one year at 4,206 cubic feet (155m^3). Based on this early usage it is concluded that MY's projected water use will be less than the 4.75 million gallons and

does not affect the conclusions of the environmental statement.

8.2.8 Terrestial Ecology

The coastline around the site varies between saltmarsh and mudflat with some rocky areas where the gradient is steepest. The saltmarsh vegetation is dominated by *Spartina patens* and *Spartina alternaflora* which are both obligate wetland species. In some locations, seaweed is also prevalent in the intertidal zone. The mudflats are generally devoid of vegetation with the exception of saltmarsh species along the edge. Mudflats are typically found in areas that are slightly sheltered such as small coves. Landward of the saltmarsh and mudflat areas, the coast has a steep incline up to the upland areas, which are dominated by trees such as white pine (*Pinus strobus*).

The entire coastline of Bailey Point and Foxbird Island within the area of the MY plant to be demolished has been delineated. The delineated area is undeveloped coastline consisting of tidal saltmarsh, rocky shore, mudflat, extending up a fairly steep gradient to a tree line and the power plant. The western side of the Point and Foxbird Island is an inlet of Montsweag Bay whereas the eastern side is bounded by the Back River.

Soils in the tidal wetland areas are inundated and submerged during large portions of the day. The upland soils were found to have a matrix color of 10YR5/4 from 0-2 inches, 10YR6/4 from 2-4 inches, and 5YR6/3 from 4-10 inches.

The property around the Eaton farmhouse and barn were also surveyed on June 2, 1999. No wetland resources were found in the vicinity of that property.

"The land around MY has been cut over at least once. The forest now produces lower quality lumber and pulp wood." (1972 Environmental Report, Pg 2.2-1)

"Land animals inhabiting the site include deer, racoons, and smaller mammals. Non-poisonous snakes can be found. ... The bird population varies greatly between seasons and between periods of migration and residency...." (1972 Environmental Report, Pg 2.7-1). Osprey nest in the treeline on the west side of the plant and on Foxbird Island. Decommissioning activities will have no adverse effect on this nesting place.

8.3 Environmental Effects of Decommissioning Activities

As stated in the MY DSAR; "A detailed site characterization was conducted prior to commencing decommissioning activities. Surveys established the contamination and radiation levels throughout the facility. This information was and is used in developing the detailed procedures ensuring that contaminated materials are removed and that worker exposure is

maintained ALARA. For the same purpose, surveys of the outdoor areas were and are performed in order to confirm the locations of known contaminated soil and identify any previously unknown contaminated soils."

8.3.1 Effects on Human Activities

The number of workers on site at any time (max 400 or less) will be no greater than when the PWR was in operation, and will be much less than when the PWR was under construction. It appears that the peak work force on site during final operations was approximately 450 (MY personnel dept) during construction was approximately 1300 (1280 CMP weekly project report week of 6/25/71, 1338 -man peak in June 1971 - 1972 Environmental Report (Pg 4-2)

Many of the decommissioning professional and skilled workers are former MYAPC employees who live in the surrounding area. There are no significant adverse impacts expected on temporary housing or schools as a result of the decommissioning activities.

MY's property taxes from the operation of the plant had a positive impact on the Town of Wiscasset. The reduction in taxes concurrent with the decommissioning of the plant has had a significant impact on the town budget planning based on operating plant tax base, however MY has negotiated a phased reduction of the taxes to minimize the financial impact on the town. Once the site is given unrestricted use release, the property has the potential of being developed as an industrial park which would increase the local tax revenue.

8.3.2 Effects on Terrain, Vegetation and Wildlife and Archeological Sites

The program will establish erosion control barriers on downhill slopes around the perimeter of the primary project area and provide storm drain inlet protection. The program will implement best management practices in accordance with the Maine Erosion and Sediment Control Handbook for Construction (3/91), and include a routine inspection program to properly maintain the barriers. Additional controls will be established, as necessary, for work encroaching on coastal wetlands

Decommissioning activities will have little of no impact on existing terrain or vegetation in these undeveloped acreage of the site.

Solid wastes will be managed in accordance with the MY Decommissioning Project Waste Management Plan and its supporting procedures.

a Concrete Rubble Evaluation (Solids)

Concrete debris that meets 10CFR20, Subpart E derived FSS acceptance criteria defined in the 10CFR50 LTP will be used on-site as fill.

Abandoning below grade concrete foundations in place, and or filling voids with concrete rubble fill will not cause any significant adverse impacts to ground water. Cured concrete does not leach free caustics (calcium hydroxide). This is recognized in the standards listed in various sections of 06-096 CMR Chapter 305 (Natural Resources Protection Act-Permit By Rule) which allows for the placement of concrete directly into bodies of water after it has cured for at least three weeks. Curing progresses from the exterior surface inward. After about 28 days of curing, concrete reaches about 99% of its strength and contains minimal free caustics that can leach into groundwater. There is no difference in the interaction of concrete rubble with brackish water or freshwater.

Concrete rubble has been successfully used in below grade applications for some time. According to Colin Lobo, of the National Ready Mix Concrete Association, the City of Chicago currently uses concrete rubble as a base material in roadbed foundations and concrete rubble has been used as fill material for many years. At Deer Island, in Boston Harbor, rubblized concrete from unwanted structures was disposed of on-site while preparing the property for construction of the wastewater treatment facility required for the EPA mandated cleanup of the harbor. The Deer Island project went through a full NEPA process prior to construction and received EPA approval. In the State of Maine, 20,000 cubic yards of concrete from the South Portland Bridge Project was processed through the John Shaw Construction Co. crusher to make gravel for use on site and mixed with other gravel for use in road beds. With MDEP approval, slightly contaminated dredge soils were blended with gravel to dispose of it.

Structures scheduled for demolition at MY have been curing for over 20 years. Crushing or breaking the concrete will not change its composition, however, some inner areas of the concrete that were not previously exposed to air will contain higher concentrations of calcium hydroxide than portions of the concrete that were closer to the surface. Since the foundations containing the concrete rubble will remain open for several days/weeks, during the demolition process, free caustics will further dissipate. As a result, although the pH in groundwater that passes through the buried concrete will be raised, the change should not be significant enough to warrant concern.

FSS results [confirmation of Subpart E of 10CFR20 compliance] negates radiological concerns associated with the use of this concrete rubble as fill. It is MYAPC's conclusion that the use of this concrete rubble as fill will have no adverse environmental effect on the groundwater.

b. Special or Hazardous Waste

MY is a RCRA subtitle C large quantity generator of hazardous waste and is subject to the closure provisions of Chapter 851, Section 11. Accordingly, appropriate closure processes will be developed to address potential hazardous waste contamination, as applicable.

No on-site mixed waste has been identified in the site characterization program. Decommissioning activities will be designed to prevent the generation of mixed waste. If generated, mixed wastes will be managed in accordance with NRC and MDEP regulations.

Asbestos (Galbestos siding panels, flooring, etc.) and other special wastes identified during the decommissioning will be managed and disposed of in accordance with MDEP regulations in an approved landfill.

8.3.3 Effects on Adjacent Waters and Aquatic Life

A barge-unloading area was constructed at the site during initial plant construction and will be used during decommissioning for shipment of specific large components (i.e. reactor vessel, pressurizer, and steam generators).

This project includes eliminating potential radioactive sources to ensure that any potential for groundwater contamination from radioactive sources is well within federal limits.

Following demolition and site restoration, contamination levels of any demolition debris used as fill will have met FSS criteria for on-site disposal. During demolition, some short-term storage of hazardous materials from the clean up of the existing RCRA site may be required. A groundwater protection plan will be developed that will ensure no discharges to groundwater from the storage area.

Following demolition at the decommissioned site, no petroleum products, pesticides, herbicides, fertilizers, road salt, solvents, acids or other material above remediation criteria with the potential to contaminate groundwater will remain on-site. Therefore no long-term groundwater protection plan is required.

Spills and leaks during the facility demolition will be controlled by the MY Spill Prevention, Containment, and Countermeasure Plan. Following demolition of the decommissioned site, contaminated material above remediation criteria will be removed and disposed of off-site at an appropriately licensed facility.

a. Effect on Local Estuaries

The principal streams in the vicinity of the site are the Sheepscot River and Montsweag Brook, with average flow rates of 307 ft^3/sec and 18 ft^3/sec respectively (UFSAR section 2.3.2). The Back River and Montsweag Bay are tidal estuarial waters, with a mean tidal range of 9.44 feet (UFSAR section 2.3.3). MY has no effect on freshwater rivers as described in the GEIS(tidal estuaries vs. freshwater river). The waters near the plant are reported to be relatively low in productivity of fish and shell fish. Some lobstering is carried out in Montsweag Bay.

The primary type of boating in the Montsweag Bay-Back River area is shallow craft pleasure boats" (UFSAR - section 2.1.2). The river, being saline, is not used for irrigation (ODCM App. A.1- no irrigation pathway).

Differences between MY Site and the generic site does not affect the conclusions of the environmental statement. Since the water is not used for irrigation , the radiologic consequences to the public will be lower than those calculated in the generic environmental statement.

8.3.4 Effects of Released Radioactive Materials

The MY DSAR states; "A detailed site characterization will be conducted. Surveys will establish the contamination and radiation levels throughout the facility. This information has and will be used in developing detailed procedures to ensure that contaminated materials are removed and worker exposure is maintained ALARA. For the same purpose, surveys of the outdoor areas have been performed to confirm the locations of known contaminated soil and identify any previously unknown contaminated soils" (DSAR page 7-2, Section 7.1.1.2 - Site Characterization).

a. Gaseous radioactive effluents

When the facility was permanently shutdown gaseous radioactive waste production ceased. The radioactive waste gases have been released in accordance with the restrictions of the plant's operating license. D&D

activities will not generate additional gaseous radioactive materials. There will be minor gaseous activity continuing to be released from the spent fuel (escaping from micro pores in the fuel cladding) and the spent fuel pool (tritium released during spent fuel pool evaporation). Some dismantlement activities will generate radioactive airborne particulate matter. The release of these particles are limited by the HEPA filtration systems in the building exhaust ventilation systems.

b. Liquid radioactive effluents

Gaseous and liquid effluents are monitored and maintained within the limits by compliance with the ODCM.

c. Effects of liquid releases

During decommissioning, waste stream discharges will be made in accordance with limitations and conditions of the NPDES waste discharge permit.

d. Accidental releases

LTP section 2-Site Characterization, recorded this information in the review of radiological incident files while developing the HSA [LTP Section 2, page 2.1 Records Review].

8.3.5 Estimated Radioactive Waste Volumes

e. High Level Waste

The MY PSDAR and MY Decommissioning Project Waste Management Plan estimates 227 cubic feet (or 6.5 cubic meters) of high level waste. When compared to the GEIS estimate of 85 cubic meters, MY's estimates are significantly less than the volumes presented in the generic model. These estimates support the conclusion that the previously issued environmental statements are bounding, since the volume of high level waste is smaller than that considered in the GEIS

b Low Level Waste

The MY PSDAR and MY Decommissioning Project Waste Management Plan estimates the low-level waste burial volume at 209,000 cubic feet (or 5,290 cubic meters). When compared to the GEIS estimate of 18, 340 cubic meters,

MY's estimates are significantly less than the volumes presented in the generic model. These estimates support the conclusion that the previously issued environmental statements are bounding, since the volume of waste will is smaller than that considered in the GEIS:

8.3.6 Radiation Exposure Estimates [MY DSAR, Sections 4.0 & 7.2]

a. Nuclear Worker Radiation Exposure Estimates

The MY DSAR Section 4..1 provides radiation protection guidance and principals during decommissioning to ensure that protective techniques, clothing and breathing apparatus are used as appropriate [MY DSAR, Section 4.0,]. Section 7.2 states:..."For decommissioning workers external exposure to radioactive materials is the dominant exposure pathway during decommissioning since the inhalation and ingestion can be minimized or eliminated as pathways by protective techniques, clothing and breathing apparatus." [MY DSAR, Section 7.2].

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

b. General Public Radiation Exposure Estimates (UFSAR section 2.1.2 & MY DSAR, Section 7.2.2)

Land use within 5 miles of the site, is largely home sites, small businesses, summer houses, idle farmland and forest.

MYAPC has committed to maintaining radioactive dose to the public below comparable levels when the plant was operating through continued application of radiation protection and contamination controls combined with the reduced source term available in the facility.

In NUREG 0586, the minimum distance from the point of plant airborne releases to the outer boundary for the generic plant is 1km, whereas the similar distance for the MY plant is 2000 feet, or 609 meters (0.6km).

The smaller percentage of land used for farming, combined with the low population density, and the commitment to continued application of radiation

protection and contamination controls during decommissioning, results in radiologic consequences to the public lower than those calculated in the GEIS.

c. Normal Transportation Radiation Exposure Estimates [MY DSAR Section 7.2.2]

The MY Decommissioning Project Waste Management Plan established transportation plans for components and categories of components. The decommissioning activities schedule establishes projected shipments of the same. Section 7.2.2 of the MY DSAR addresses radiation exposure to the public from normal transportation and the processes for assuring protection.

The MY Decommissioning Project Waste Management Plan, the DSAR and the applicable waste management procedures provide for transport of radioactive waste in accordance with applicable regulations, and the requirements of the receiving facility.

8.4 Environmental Effects of Accidents and Decommissioning Events

MYAPC concluded that the consequences of the postulated accident or occurrences would have no significant adverse environmental effects.

MYAPC's original discussion on environmental issues and the DSAR addresses the postulated limited event for MY for whole body exposure during decommissioning, as a spent resin cask drop accident in the LLRWe Building. It is calculated that more potential airborne activity can be released from that accident scenario that any other decommissioning accidents. The radionuclide distribution assumed for the spent resin cask includes more transuranics (the major dose contributor) than the distribution in the components involved in the other potential decommissioning accidents analyzed. The following is a list of the remaining potential accidents reviewed in the DSAR.

8.4.1 In Plant Event Evaluated

a. Explosion of liquid petroleum gas (LPG) leaked from a front end loader or forklift.

The pressure spike created by such an explosion will create a pressure spike, but not in excess of the containment design pressure, therefore no structural damage to the containment will occur. The high efficiency particulate (HEPA) filter systems in operation supporting decommissioning activities will capture any

loose contaminated dust made airborne by the heat, mechanical and aerodynamic disturbances created by such an explosion.

b. Explosion of oxyacetylene during segmenting of the reactor vessel shell.

The pressure spike created by such an explosion will create a pressure spike, but not in excess of the containment design pressure, therefore no structural damage to the containment will occur.

c. Fire involving contaminated clothing or combustible waste.

Administrative controls are in place that limit the accumulation of large quantities of these materials in one place, thereby maintaining a small potential for airborne contamination in the event of such an event.

d. Loss of local airborne contamination control during blasting or jackhammer operations.

In many instances contamination control envelopes are used in addition to a fine fog spray, during jackhammer operations. This would be readily apparent to the operator who would stop the operation immediately.

e. Temporary loss of services.

The loss of water, power, or airflow would be immediately evident to the operator who would take compensatory action.

f. Dropping a container of contaminated concrete rubble.

If a container was accidentally spilled, the fine particles adhering to the rubble would become suspended in the air. The quantity of such material would be significantly small since the readily suspendible particles would be removed during routine decontamination operations, or coating prior to rubblization.

g. Natural phenomena events are characterized as low probability occurrences, which are not increased by decommissioning activities.

h. Non-radiological safety evaluation.

NUREG/CR-0130 generically evaluated chemical pollutants that could be released during decommissioning activities and the quantities were found to be

insignificant. Potentially hazardous chemicals were found to come from three sources: residuals from PWR operations, chemicals employed in decontamination, and non-radioactive fission products resulting from radioactive decay. Due to the small quantities of these chemicals on the MY site and the low probability of their dispersal into the environs, it is concluded that they do not pose a significant public hazard.

8.4.2 Radioactive Release from subsystem or component

DSAR concluded that the potential radioactive releases from a subsystem or component are conservatively bounded .

8.4.3 Fuel handling accidents outside containment

This is a very remote possibility due to the administrative controls and physical limitations imposed on these operation. The potential dose to an individual at the exclusion area boundary would be a small fraction of the 10CFR100 limits.

8.4.4 Loss of spent fuel heat removal capability

The SFP is designed to maintain fuel cladding integrity in the event that forced cooling is lost. In that event, cooling occurs by boiling at the surface of the SFP with evaporative losses being made up by a supply of makeup water.

The analysis indicated that a sufficient time is available to restore make up water to the SFP prior to any uncovering of spent fuel. Therefore, there are no radioactive releases associated with this postulated event.

8.4.5 Decommissioning events

Analyzed a number of potential events that could occur during decommissioning that could result in release of radioactive materials (included the potential for liquid and/or airborne radioactive releases).

Reviewed administrative limits that ensure that any potential liquid release are within the limits of 10CFR20, Appendix B.

8.4.6 Offsite radiological events

These events were limited to shipment of radioactive materials made in conformance with regulatory requirements. MY's radioactive waste management program and the

nuclear quality assurance program ensure regulatory compliance and that the consequences of an offsite event does not significantly affect the public health and safety.

8.5 Environmental monitoring activities

8.5.1 FSS criteria - FSS plan

Radiation monitoring instruments used during the FSS will be selected as appropriate for the physical and environmental conditions and the type of radiation being measured. Measurements will be performed by trained individuals using calibrated instruments. The survey instruments will be controlled by specific procedures that define accuracy and calibration requirements.

8.5.2 Environmental radiological surveillance program

The MY Environmental Radiological Surveillance Program includes monitoring, sampling, analyzing and reporting of radiation and radionuclides in the environment and assures that monitoring is conducted in accordance with the methodology and parameters provided in the ODCM as required by Tech Specs.

There is no indication that the operation of MY has any significant impact on the environs around the site.

8.6 Environmental approvals

8.6.1 Federal requirements

10CFR50.2 defines "decommissioning" as those activities".. to remove (the reactor) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license." Under this definition, decommissioning does not include the removal and disposal of nonradioactive structures beyond that necessary to allow unrestricted access to the site. However MYAPC has gone beyond this limit to include those activities to return the decommissioned site to a "greenfield" condition for unrestricted use possibly as an industrial park.

10CFR 51.53, "Supplement to environmental report," requires that a utility applying for authorization to decommission a nuclear power reactor submit a supplement to the environmental report to address the post operating license stage of the facility. This report should reflect any new information or significant environmental change associated

with the proposed decommissioning activities. This report is intended to satisfy this requirement

8.6.2 State of Maine Regulations address

a. Odors

Generally odors created by the demolition activities will be restricted to the immediate site area and should not affect locations off MY property.

b. Water Vapor

Demolition activities are not anticipated to create any significant new water vapor emissions in sufficient quantity to change the local climate..

c. Air Emissions

Air pollutant emissions associated with the decommissioning of the MY plant will be comprised mainly of gaseous emissions from diesel-powered equipment and fugitive dust emissions from demolition activities and heavy equipment traveling on paved and unpaved roads. Air quality impacts from this type of activity are usually mitigated to the extent that potential offsite nuisance conditions (or a condition of air pollution) are prevented. The emission factors used in the estimates for demolition activities are taken from the 5th edition of EPA's AP-42 document (EPA 1995) assuming reasonable levels of emissions control as needed to minimize dust emissions. Diesel powered vehicle emissions are derived from the latest version of EPA's MOBILE5b emissions estimating model (EPA 1996).

d. Noise

The State of Maine limits the sound levels from the construction and operation of developments. Although this project proposes demolition, we are assuming that the construction noise codes will apply. The regulations applicable to this project are summarized below.

1. Construction sound level requirements

Demolition activities that occur between 7 p.m. and 7 a.m. must meet nighttime operational noise limits that depend on existing ambient sound levels in the noise sensitive residential areas adjacent to the site. The

most stringent levels requirements apply to "protected areas" defined as areas with predevelopment nighttime ambient sound levels of 35 dBA or less; higher levels of nighttime noise are allowed by permit only. The allowable nighttime limit on noise in protected areas is 45 dBA. It is assumed that the nearest residential area to the east of the site is a protected area with an allowable nighttime limit of 45 dBA. Permits for nighttime construction lasting 90 days or less are issued by the local municipality. For longer periods, the permit also needs the approval of the State. Sound level limits for daytime construction activities (between 7 a.m. to 7 p.m.) are dependent on the duration of the noise. For example, a limit of 87 dBA is required for a 12 hour daytime period. Construction equipment is required to comply with applicable federal noise emission regulations.

2. Operational noise

Since the project will remove the structures and restore the decommissioned site to naturally vegetated open space, there will be no facilities remaining on-site to create noise once site restoration is completed.

3. Noise impact assessment

The nearest noise sensitive receptors are 2400 ft., east across the Back River on Westport Island, and 2400 feet NW of the site on the mainland. The daytime sound level for the mobile demolition equipment on site is expected to be well within the 87 dBA limit for 12 hour day time activities.(the level is expressed in terms of Leq, the energy average sound level).

To reduce noise levels at night, the crusher will not be used between the hours of 7 p.m. and 7 a.m. The nighttime sound level at the nearest noise sensitive receptors from loudest equipment that will operate at night during project demolition is expected to be approximately 59 dBA..

d. Geotechnical investigation

Demolition material that is suitable for on-site use, will be crushed and deposited as fill within the lower elevations of the buildings , or used as fill on other parts of the site.

The Robert G. Gerber report provides sufficient information to provide an analysis of the effects of the demolition program on soils and groundwater. The soils map from the Gerber report are available for review. No changes to soils should occur as a result of the demolition project. Groundwater levels and monitored conditions will be submitted as required.

e. Visual quality

Visibility of above ground, man-made, features at the decommissioned site will be removed as part of this project, with the exception of the 345kV switchyard, the 345kV transmission lines and towers, and the ISFSI. The decommissioned site will be restored to a naturally vegetated state that visually conforms with the surrounding shoreline.

f. Storm water management.

Except along the shorelines, there is no flooding at the site which is above the 100-year flood plain and is also above the probable maximum flood elevation, as described in Section 2.3.3 of the MY DSAR. Current and post-demolition discharges will not cause flooding to occur in the receiving waters.

In summary, the proposed demolition will reduce the amount of impervious area by approximately 16.8 acres, primarily due to revegetation of areas currently occupied by buildings, roads, and parking lots. Storm water drainage that is currently achieved through a combination of sheet flow and many point discharges will be achieved mainly through sheet flow runoff following demolition.

g. Runoff analysis

The worst case runoff for the 2-year, 10-year, and 25-year storms was calculated. Storm rainfall amounts were obtained from the National Weather Service TP 40 Rainfall Frequency Atlas of the United States.

Approximately 26 acres will be disturbed as part of project demolition activities. Erosion and sedimentation control methods to be used for this project will ensure that project demolition and site restoration activities are carried out with minimal impact to adjacent areas due to erosion or sedimentation. Controls will be consistent with the Maine Erosion and Sediment Control Handbook for Construction: Best Management Practices March 1991

The project schedule will ensure that measures are in place prior to initiation of demolition activities.

The length of time bare soils will be exposed during clearing and excavation operations will be minimized.

After demolition is complete, temporary erosion and sedimentation control barriers (silt fences, bales, matting, etc.) will be removed when an acceptable stand of vegetation is established.

As a result of these prevention actions, work activities along the waterfront are expected to have a minimal effect on coastal wetlands and waters.

8.7 Conclusion

MYAPC's original environmental report demonstrated that no unacceptable effects on the environment would result form the construction and operation of the Plant. The acceptable environmental effects associated with the operation of the plant are expected to be significantly reduced or eliminated by the proposed decommissioning activities.

This supplement concludes that those activities will be accomplished with no significant adverse environmental impacts. This plan does not describe any different or additional plant activities beyond those in the decommissioning schedule and the DSAR. Therefore the environmental impacts associated with the D&D activities described herein are bounded.

The environmental impacts associated with the D&D activities associated with this LTP have no significant effect on the quality of the environment are bounded by the GEIS. The PSDAR states that the exposure and burial volumes will not exceed the values established in the GEIS.

As evaluated the proposed D&D activities will be accomplished with no significant adverse environmental effects. Continued and future decommissioning activities will be controlled by compliance with applicable environmental requirements (Federal, State and Local). The evaluations of decommissioning, or license termination activities conducted to date support this conclusion, in that:.

- MYAPC-specific activities and associated effects continue to be conservatively bounded by the conclusions of the GEIS in NUREG-0586 and in NUREG-1496.

- Projected radiation dose to the public continues to be minimal and well within the bounds of the GEIS.

- Projected burial volumes for LLW and highly activated (deep burial) waste do not exceed the values established in the GEIS.

- Projected average annual radiation dose to decommissioning workers has been consistently less than projected in the GEIS.

- Decommissioning activities continue to have no significant adverse effects on the environment.

- Non-radiological exposure (e.g. chemical or toxic materials) are consistently less than projections in the GEIS.

8.8 References

- MYAPC letter to NRC , MN-97-99, PSDAR dated August 27, 1997.

- MYAPS Environmental Report, Supplement One, dated April 19, 1972.
  (Earlier document dated October 1970 filed with the AEC,)
  (February 1971 addenda filed)

- April 19, 1972-Self Contained Supplement, fully replacing previous submittals.

- July 1972-Final Environmental Statement Related to Operation of the MYAPS

- NUREG-0586, Final Generic Environmental Impact Statement dated August , 1988

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

- NUREG/CR-0130, "Technology, Safety and Cost of Decommissioning a Reference Pressurized Water Reactor Power Station"

- State of Main Maine Chapter 851, Section 11.

- EPA. 1995. Environmental Protection Agency, "Compilation of Air Pollutant Emission Factors, Volume I: Stationary Point and Area Sources", 5th Edition, AP-42, 1995 (Sections 11.9 11.19, 13.2).

- EPA. 1996. Environmental Protection Agency, "User's Guide to MOBILE5 (Mobile Source Emissions Factor Model)". Office of Air Quality and Radiation, Office of Mobile Sources, Emission Planning and Strategies Division, Ann Arbor, MI., Report

no. EPA-AA-AQAB-94-01, revised September, 1996. 5th edition of EPA's AP-42 document (EPA 1995).

- RCRA Chapter 851, Section 11.
- The Robert G. Gerber report. " Summary of Geologic Information Covering The Maine Yankee Nuclear Power Plant Site and Vicinity" dated May, 1991
- Maine Erosion and Sediment Control Handbook: Best Management Practices, March 1991.

MAINE YANKEE

LTP SECTION 9

ACRONYMS

ACRONYMS

ALARA	As Low As Reasonably Achievable
ANI	American Nuclear Insurers
ANSI	American National Standards Institute
APS	American Protective Services
AR	Air Removal
AS	Auxiliary Steam
ASAP	As Soon As Possible
AUX	Auxiliary
BA	Boric Acid Storage Tank
BAMT	Boric Acid Mix Tank
BAST	Boric Acid Storage Tank
BOD	Board of Directors
BWST	Boron Waste Storage Tank
C of C	Certificate of Compliance
CAP	Community Advisory Panel
CEDM	Control Element Drive Mechanism
CFR	Code of Federal Regulations
CMP	Central Maine Power
COMM	Communications
CPU	Cavity Purification
CR	Condition Report
CST	Condensate Storage Tank
CTMT	Containment
CW	Circulating Water
CWPH	Circulating Water Pump House
D&D	Dismantlement and Decontamination
DAW	Dry Active Waste
DEP	Department of Environmental Protection
DFPP	Decommissioning Fire Protection Program
DOC	Decommissioning Operations Contractor
DOE	Department of Energy
DOT	Department of Transportation
DSAR	Decommissioning Safety Analysis Report
DWO	Decommissioning Work Order
DWST	Demineralized Water Storage Tank
EPA	Environmental Protection Agency
EVS	Earned Value Schedule
FHE	Fuel Handling Equipment
FP	Fuel Pool
FSAR	Final Safety Analysis Report
FSS	Final Site Survey
GPS	Global Positioning System
GTCC	Greater Than Class C
HAZMAT	Hazardous Materials
HEPA	High Energy Particulate Air
HLW	High Level Waste
INPO	Institute of Nuclear Power Operations
ISFSI	Independent Spent Fuel Storage Installation
ITDC	Important To the Defueled Condition
KV	Kilovolt
KW	Kilowatt
LD	Letdown
LLRW	Low Level Radioactive Waste
LLW	Low Level Waste
LLWB	Low Level Waste Building
LSA	Low Level Storage Area
LTP	License Termination Plan

MCC	Motor Control Center
MDC	Minimum Detectable Concentration
MDEP	Maine Department of Environmental Protection
MET	Meteorological
MY	Maine Yankee
MYAPC	Maine Yankee Atomic Power Company
MYQAP	Maine Yankee Quality Assurance Program
NAC	Nuclear Assurance Corporation
NCR	Nonconformance Report
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
ODCM	Offsite Dose Calculation Manual
OSHA	Occupational Safety and Health Administration
PAB	Primary Auxiliary Building
PCB	Poly Chloride Biphenyls
PCC	Primary Component Cooling
PCP	Process Control Program
PMP	Project Management Procedure
PPE	Personal Protective Equipment
PWST	Primary Water Storage Tank
PZR	Pressurizer
QA	Quality Assurance
RA	Restricted Area
RAI	Request for Additional Information
RC	Radiation Controls
RCA	Radiological Controlled Area
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REG	Regulation
REMP	Radiation Effluent Monitoring Program
RETS	Radiological Effluent Tech Spec
RMS	Radiation Monitoring System
RP	Radiation Protection
RPM	Radiation Protection Manager
RPV	Reactor Pressure Vessel
RW	Raw Water
RWP	Radiation Work Permit
RWST	Refueling Water Storage Tank
RX	Reactor
S&W	Stone & Webster
SG	Steam Generator
SAR	Safety Analysis Report
SCAT	Spray Chemical Addition Tank
SCC	Secondary Component Cooling
SCDHEC	South Carolina Department of Health and Environmental Control
SDP	Site Development Plan
SEC	Security
SEM	Site Engineering Manager
SER	Safety Evaluation
SERT	System Evaluation Reclassification Team
SFP	Spent Fuel Pool
SFPI	Spent Fuel Pool Island
SHE	Safety, Health, Environmental
SHE&R	Safety, Health, Environmental & Radiological
SRP	Savannah River Project
SSC	Systems, Structures & Components
SSI	Specific Safety Instructions

SSL	Secondary Seal & Leakage
STP	Sewerage Treatment Plant
STRU	Structures
SVH	Steam & Valve House
SW	Service Water
SW	Stone & Webster
SWEC	Stone & Webster Engineering Corporation
SYS	System
TE	Technical Evaluation
TEDE	Total Estimated Dose Equivalence
TK	Tank
TLD	Thermo Luminescent Dosimeter
TS	Technical Specifications
TURB	Turbine
UMS	Universal Multipurpose System (Cask)
USCQ	United States Coast Guard
USDOE	United States Department of Energy
USDOT	United States Department of Transportation
VCT	Volume Control Tank
VLV	Valve
WD	Waste Disposal
WMG	Waste Management Group
WO	Work Order
WPB	Wiscasset Planning Board
WT	Water Treatment
WTO	White Tagging Order
X	Transformer (X-16,24.....)
XFMR	Transformer

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