U.S. Department
of Transportation
**Maritime
Administration**

Office of Ship Operations

1200 New Jersey Ave., SE
Washington, DC 20590

Ref: 10 CFR 50.82 and 50.90

October 23, 2023

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Docket No. 50-238; License No. NS-1; N.S. *SAVANNAH*
License Amendment Request No. LAR 2023-01
Submittal and Request for Approval of the License Termination Plan

Pursuant to 10 CFR 50.90, the United States Maritime Administration (MARAD) hereby submits its License Termination Plan (LTP), and requests approval to amend the Nuclear Ship *SAVANNAH* (NSS) Facility Operating License, NS-1 (License) to incorporate the enclosed proposed change into the NSS license.

In accordance with 10 CFR 50.82(a)(10)

> *... the Commission shall approve the [License Termination] plan, by license amendment, subject to such conditions and limitations as it deems appropriate and necessary and authorize implementation of the license termination plan.*

MARAD's reason for this submittal is to request the NRC approve and authorize implementation of the LTP and revisions to it.

In accordance with 10 CFR 50.82(a)(9), MARAD

> *... must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval...*

MARAD has chosen to submit the LTP prior to submitting an application to terminate the license.

The proposed license amendment will modify the License to add a license condition approving the NSS LTP and revisions to it. Regarding revisions to the LTP, the proposed change will reference the criteria in LTP Chapter 10, *LTP Areas That Cannot Be Changed Without NRC Approval.*

In accordance with 10 CFR 50.82(a)(9)(i),

> *The license termination plan must be a supplement to the [Final Safety Analysis Report] FSAR or equivalent and must be submitted at least 2 years before termination of the license date.*

MARAD has chosen to submit the LTP as a supplement to the FSAR. MARAD affirms its understanding that the license cannot be terminated in fewer than two years from submittal of this LTP, in accordance with 10 CFR 50.82(a)(9)(i). MARAD anticipates requesting license termination to be effective in December 2025, provided all prerequisite actions are complete at that time.

The purpose of the LTP is to demonstrate compliance with NRC decommissioning requirements.

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The LTP satisfies the requirements of 10 CFR 50.82(a)(9) as it has been developed following the guidance in Revision 2 of Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, and in NUREG-1757, Volume 2, *Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria.*

The LTP satisfies applicable portions of the requirements of 10 CFR Part 20, Subpart E, *Radiological Criteria for License Termination.* Specifically, the LTP satisfies the requirements of the following:

- 10 CFR 20.1401 General provisions and scope; and,
- 10 CFR 20.1402 Radiological criteria for unrestricted use.

MARAD has no interest in pursuing license termination under restricted conditions or by using alternate criteria.

The License Amendment Request is provided in five (5) enclosures to this letter. Enclosure 1 is an evaluation of the request. Enclosure 2 provides the existing License marked up to show the proposed change. Enclosure 3 provides a retyped version of the proposed License change. Enclosure 4 is the LTP. Enclosure 5 is the Acceptance Criteria Review Matrix (ACRM). The ACRM documents each of the acceptance criteria found in RG 1.179 and NUREGs 1700 and 1757. The matrix acted as a QA measure for MARAD in preparing the LTP. MARAD hopes the matrix will be of assistance to NRC's reviewers.

MARAD has reviewed the proposed change comparing it to the current license basis in accordance with 10 CFR 50.92 and concludes that it involves no significant hazards consideration.

Pursuant to 10 CFR 50.91(b), a copy of this letter has been forwarded to the State of Maryland. The Safety Review Committee has reviewed this request.

MARAD requests approval of the proposed License Amendment by October 23, 2024 for implementation within 30 days from the date of approval.

This letter contains no regulatory commitments.

If there are any questions or concerns with any issue discussed in this submittal, please contact me at: O: (202) 366-2631, M: (410) 776-8268, and/or e-mail me at erhard.koehler@dot.gov.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 23, 2023.

Respectfully,

Erhard W. Koehler
Senior Technical Advisor, N.S. *SAVANNAH*
Office of Ship Operations

Enclosures (5)

Enclosures:

1. Evaluation of License Amendment Request

2. Proposed License Change (marked-up)

3. Proposed License Change (retyped)

4. License Termination Plan

5. Acceptance Criteria Review Matrix cc:

Electronic copy
NSS ESC
NSS SRC

MAR 610, 612, 615

Hardcopy, cover letter only
MAR-600, 640, 640.2

Hardcopy w/ all enclosures
MAR-100, 640.2 (rf)
USNRC (Tanya Hood, Andrew Taverna)
USNRC Regional Administrator - NRC Region I
MD Department of the Environment (Eva Nair)

EK/jmo

U.S. Department of Transportation
Maritime Administration

Office of Ship Operations

1200 New Jersey Ave., SE
Washington, DC 20590

Docket No. 50-238; License No. NS-1; N.S. *SAVANNAH*

ENCLOSURE 1 EVALUATION OF LICENSE AMENDMENT REQUEST

Subject: Add License Condition for the License Termination Plan

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License NS-1 for the Nuclear Ship *SAVANNAH* (NSS).

The Maritime Administration (MARAD) is a modal agency of the United States Department of Transportation (DOT). It is a Federal licensee as defined by the NRC. As such, funds for decommissioning and termination of the NSS license are provided by Federal appropriations.

MARAD proposes to revise the Operating License to add License Condition 2. C. (4) approving and authorizing implementation of the License Termination Plan (LTP).

MARAD requests approval of the proposed License Amendment by October 23, 2024 for implementation within 30 days from the date of approval.

2.0 PROPOSED CHANGE: ADD LICENSE CONDITION FOR LTP

In accordance with 10 CFR 50.82(a)(10)

> *... the Commission shall approve the [License Termination] plan, by license amendment, subject to such conditions and limitations as it deems appropriate and necessary and authorize implementation of the license termination plan.*

Therefore, MARAD proposes adding the following License Condition:

> 2. C. (4) MARAD shall implement and maintain in effect all provisions of the License Termination Plan (LTP), as approved in License Amendment No. xx. MARAD may make changes to the LTP without prior NRC approval provided the proposed changes are in accordance with LTP Chapter 10, *LTP Areas That Cannot Be Changed Without NRC Approval.*

MARAD proposes revising the License as shown in Enclosure 2, "Proposed License Change (marked-up)" and Enclosure 3, "Proposed License Change (retyped)."

3.0 TECHNICAL EVALUATION

In accordance with 10 CFR 50.82(a)(9)(ii), the LTP must address the following topics:

(ii) The license termination plan must include—

- *(a) A site characterization;*
- *(b) Identification of remaining dismantlement activities;*
- *(c) Plans for site remediation;*
- *(d) Detailed plans for the final radiation survey;*
- *(e) A description of the end use of the site, if restricted;*
- *(f) An updated site-specific estimate of remaining decommissioning costs;*
- *(g) A supplement to the environmental report, pursuant to § 51.53, describing any new information or significant environmental change associated with the licensee's proposed termination activities; and*
- *(h) Identification of parts, if any, of the facility or site that were released for use before approval of the license termination plan.*

The LTP satisfies the requirements of 10 CFR 50.82(a)(9) as it has been developed following the guidance in Revision 2 of Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors* (Reference a) and in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans* (Reference b).

The purpose of the LTP is to demonstrate compliance with NRC decommissioning requirements. Regarding 10 CFR 50.52(a)(10), the LTP demonstrates the following:

- Completing the remainder of decommissioning activities will be performed in accordance with the NRC regulations;
- Completing decommissioning will not be inimical to the common defense and security;
- Completing decommissioning will not be inimical to the health and safety of the public; and,
- Completing decommissioning will not have a significant effect on the quality of the environment.

The LTP satisfies applicable portions of the requirements of 10 CFR Part 20, Subpart E, *Radiological Criteria for License Termination.* Specifically, the LTP satisfies the requirements of the following:

- 10 CFR 20.1401 General provisions and scope; and,
- 10 CFR 20.1402 Radiological criteria for unrestricted use.

MARAD has no interest in pursuing license termination under restricted conditions or by using alternate criteria.

The LTP is arranged in ten (10) chapters, as described below. Eight (8) of the chapters address the specific requirements of 10 CFR 50.82(a)(9)(ii), i.e., those labeled (a) through (h), and two (2) chapters address general information and areas of the LTP that cannot be changed without NRC approval

1. Chapter 1, General Information

   LTP Chapter 1 describes the process used to meet the requirements for terminating the 10 CFR Part 50 license and to release the NSS for unrestricted use. The LTP has been prepared in accordance with the requirements in 10 CFR 50.82(a)(9) and is submitted both as an enclosure to support License Amendment Request (LAR) 2023-01 for the NRC to approve the LTP and as a supplement to the FSAR.

2. Chapter 2, Site Characterization

   LTP Chapter 2 discusses the Historical Site Assessment and characterization activities that have been conducted to determine the nature and extent of radioactive contamination on the ship prior to remediation.

   The information obtained from the characterization provides guidance for decontamination and remediation planning.

   Data from subsequent characterization may be used to change the original classification of an area, within the requirements of this LTP, up to the time of Final Status Surveys (FSSs), as long as the classification reflects the level of residual activity existing prior to any remediation in the area.

3. Chapter 3, Identification of Remaining Site Dismantlement Activities

   LTP Chapter 3 identifies the remaining site dismantlement and decontamination activities. The information includes those areas and equipment that need further remediation and an assessment of the potential radiological conditions that may be encountered. Estimates of the occupational radiation dose and the quantity of radioactive material to be released to unrestricted areas during the completion of the scheduled tasks are provided. The projected volumes of radioactive waste that will be generated are also included.

1.5.4 Chapter 4, Remediation Plans

LTP Chapter 4 discusses the various remediation techniques that may be used during decommissioning to reduce residual contamination to levels that comply with the release criteria in 10 CFR 20.1402. This chapter also discusses the ALARA evaluation and the impact of remediation activities on the Radiation Protection Program.

The selected remediation methods used are dependent upon the contaminated material and extent of contamination. The principal materials that may be subject to remediation are structural surfaces.

Note that there is no embedded piping, soils, surface water or groundwater at NSS. Remediation techniques that may be used for structural surfaces include standard and high pressure washing, wiping, grit blasting, chemical stripping and other methods.

5. Chapter 5, Final Status Survey Plan

   LTP Chapter 5 presents the final survey process used to demonstrate that the NSS complies with radiological criteria for unrestricted use specified in 10 CFR 20.1402. (e.g., annual dose limit of 25 mrem to AMCG plus ALARA). This chapter also describes proposed control mechanisms to ensure areas are not re-contaminated.

   The FSS Plan describes use of the Data Quality Objectives (DQO) process in designing surveys, survey methods and instrumentation, data collection and processing, and data assessment and compliance.

6. Chapter 6, Compliance with the Radiological Criteria for License Termination

   LTP Chapter 6 presents the radiological information and methods used to demonstrate compliance with the radiological criteria for license termination and to release the NSS for unrestricted use. Chapter 6 discusses radionuclides potentially present and mixture fractions, exposure pathways, computational models used for dose modeling, sensitivity analysis and deterministic parameter selection, Derived Concentration Guideline Level (DCGL) and Dose Factors, the basis for the selected reasonably foreseeable and less likely but plausible scenarios.

7. Chapter 7, Update of the Site-Specific Decommissioning Costs

   LTP Chapter 7 provides a detailed discussion of the NSSS, LLC Decommissioning Contract that satisfies the underlying intent of the 10 CFR 50.82(a)(9)(ii)(F) and how the current contract is sufficient to complete the remaining decommissioning activities to release the NSS for unrestricted use.

8. Chapter 8, Supplement to the Environmental Assessment

   LTP Chapter 8 describes the Environmental Assessment (Reference c) and Supplemental Environmental Assessment (Reference d) prepared by MARAD to evaluate new information and any significant environmental impacts associated with NSS decommissioning and license termination activities. MARAD is required by the National Environmental Policy Act of 1969 to evaluate the proposed action of decommissioning and terminating the license of the NSS.

   MARAD has determined that there are no significant environmental impacts for decommissioning the NSS.

9. Chapter 9, Portions of the Facility that were released prior to LTP Approval

   LTP chapter 9 describes the portions of the facility that were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H). No parts of the facility were released for use before approval of the LTP.

10. Chapter 10, Regulatory Notifications of Changes

    Chapter 10 lists the LTP areas that cannot be changed without NRC approval. Areas applicable to NSS are derived from Appendix B of NUREG-1700 (Reference b).

4.0 REGULATORY SAFETY ANALYSIS

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

MARAD started the decommissioning process in 1971 under the 10 CFR 50.82 rule that was completely revised in 1996.

The Statements of Consideration (SOCs) for the revised rule addressed the question - Are existing facilities grandfathered from any part of the proposed rule (Issue 14 - "Grandfathering" Considerations).

In response, the SOCs state (in part):

> *The Commission has reconsidered the issue of "grandfathering" and modified the language in the final rule to provide more specific guidance for nuclear power reactor licensees whose facilities are currently at certain stages of decommissioning. The Commission has decided to eliminate the provision in the proposed rule that would give those licensees that have an NRC approved decommissioning plan, before the date when a final rule became effective, the option of either complying with the final rule requirements or continuing with the requirements of the currently existing rule. All licensees will be required to comply with the decommissioning procedures specified in the provisions of the final rule, when it becomes effective ...*

MARAD understands they have been "grandfathered" as follows:

> MARAD holds a Possession-only license for the NSS nuclear utilization facilities that was modified by License Amendment 15 (Reference e) to allow dismantlement and disposal. As a result of License Amendment 15, the status of the facility is "Dismantlement."
>
> Dismantlement is defined in Regulatory Guide (RG) 1.86, "Termination of Operating Licenses for Nuclear Reactors," Reference (f). This 1974 RG describes the now outmoded Dismantling option of decommissioning. MARAD understands RG 1.86 was withdrawn as noticed in the Federal Register (81 FR 53507) on August 12, 2016 and that its withdrawal does not impact the NSS licensing basis.
>
> Dismantlement is characterized by removal of radioactive fluids, radioactive wastes and other materials having activities above accepted unrestricted activity levels. Mothballed activities continue to be performed. These include active surveillance, monitoring and maintenance of the nuclear facilities housed onboard the ship, and custody and maintenance of the ship as the primary physical boundary and protective barrier of the licensed site.
>
> Prior to the revision of 10 CFR 50.82 in 1996, MARAD would have been required to meet the following guidance of RG 1.86 to terminate the NSS license:

a. A dismantlement plan is submitted.

b. After NRC review and approval of the dismantling plan, a dismantling order is issued.

c. When dismantling is completed and the NRC has been notified by letter, the facility is inspected by NRC to verify the dismantlement plan has been completed.

d. If residual radiation levels do not exceed the values in Table I [of RG 1.86], the NRC may terminate the license.

The current 10 CFR 50.82 regulations share the underlying intent of the RG 1.86.

MARAD's review of the SOCs illuminated the fact that submittal of the LTP crosses the threshold where "grandfathering" ends. The new rule only allowed an existing NRC approved dismantlement plan to qualify as a Post-Shutdown Decommissioning Activities Report. All licensees would be required to submit an LTP [i.e., comply with 10 CFR 50.82(a)(9) and (10)]. Therefore, MARAD understands that with the submittal of the LTP for NRC approval, the following occurs:

- NSS license will remain as a Possession-only license.
- RG 1.86 remains as part of the NSS licensing basis to the extent appropriate to the NSS Possession-only license.
- NSS licensing basis changes to also include all aspects of the 1996 rule change and its associated Regulatory Guides and NUREGs.

In accordance with 10 CFR 50.82(a)(10)

> *... the Commission shall approve the [License Termination] plan, by license amendment, subject to such conditions and limitations as it deems appropriate and necessary and authorize implementation of the license termination plan.*

MARAD's reason for this submittal is to request the NRC approve and authorize implementation of the LTP and revisions to it.

In accordance with 10 CFR 50.82(a)(9), MARAD

> *... must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval...*

MARAD has chosen to submit the LTP prior to submitting an application to terminate the license.

The proposed license amendment will modify the License to add a license condition approving the NSS LTP and revisions to it. Regarding revisions to the LTP, the proposed change references the criteria in LTP Chapter 10, *LTP Areas That Cannot Be Changed Without NRC Approval.*

Chapter 10 lists the LTP areas that cannot be changed without NRC approval. In Section 1.2, NUREG-1700 (Reference b) states, in part:

> *In accordance with 10 CFR 50.82(a)(10), the LTP is approved by license amendment. Recognizing that there may be a need to make changes to the LTP following its approval by the NRC, the licensee should include a provision in the LTP that concerns such changes. Appendix 2 [(sic) B], "LTP Areas That Cannot Be Changed Without NRC Approval," sets out such a provision that the NRC finds acceptable.*

For the NSS, the LTP areas that cannot be changed without NRC approval are derived from Appendix B of NUREG-1700.

In accordance with 10 CFR 50.82(a)(9)(i),

> *The license termination plan must be a supplement to the [Final Safety Analysis Report] FSAR or equivalent and must be submitted at least 2 years before termination of the license date.*

MARAD has chosen to submit the LTP as a supplement to the FSAR and notes that the current schedule for completing License Termination is greater than 2 years. MARAD anticipates requesting license termination to be effective in December 2025, provided all prerequisite actions are complete at that time.

The LTP satisfies the requirements of 10 CFR 50.82(a)(9) as described in Section 3.0.

4.2 PRECEDENT

A recent precedent is established in the approval of the Lacrosse LTP which added a license condition similar to the one proposed in Section 2.0 (Reference g).

4.3 PROPOSED DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

MARAD has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change is administrative in nature and does not involve modification of any plant equipment or affect basic plant operation. The proposed change is requesting NRC approve a document, the LTP and revisions to it. The document is a detailed plan of how MARAD will satisfy the criteria to allow NRC to terminate the NSS license.

   The NSS reactor is not operational, all reactor fuel has been removed from the site since 1971 and the level of radioactivity in the NSS has significantly decreased from the levels that existed when the 1976 Possession-only License was issued. All safety-related systems are deactivated, disabled, drained and perform no active function. No aspect of the proposed change is an initiator of any accident previously evaluated. Consequently, the probability of an accident previously evaluated is not significantly increased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change is administrative in nature and does not involve physical alteration of plant equipment that was not previously allowed by the License or Technical Specifications. The proposed change does not change the method by which any safety-related system performs its function. All safety-related systems are deactivated, disabled, drained and perform no active function. No new or different types of equipment will be installed. The reactor will remain permanently shutdown and defueled.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change is administrative in nature. NRC approval of the proposed will have no effect on margins of safety relevant to the ship's defueled and partially dismantled primary and auxiliary reactor systems. As such, no change is being made to safety analysis assumptions, safety limits or safety system settings that would adversely affect plant safety as a result of the proposed change. The proposed change only involves requesting NRC approval of the LTP and revisions to it.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, MARAD concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

Based on the above, MARAD concludes that the proposed amendment(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment request is confined to (i) changes to surety, insurance, and/or indemnity requirements, or (ii) changes to recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

a. Regulatory Guide 1.179, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors, Rev. 2, July 2019

b. NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans, Rev. 2, April 2018

c. Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated October 3, 2008, - *Submittal of Finding of No Significant Impact and Environmental Assessment* (ML082810182)

d. CR-137, *Supplemental Environmental Assessment and Finding of No Significant Impact*, April 2019

e. Letter from Mr. John B. Hickman (NRC) to Mr. Erhard W. Koehler (MARAD), dated April 23, 2018, *Nuclear Ship SAVANNAH - Issuance Of Amendment 15 to revise the License to allow Dismantlement and Disposal*

f. Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*, June 1974

g. Letter from Ms. Marlayna Vaaler (NRC) to Mr. John Sauger (LaCrosse*Solutions*), dated May 21, 2019, La Crosse Boiling Water Reactor – Issuance Of License Amendment No. 75 To Approve the LaCrosse*Solutions*, LLC License Termination Plan

U.S. Department
of Transportation
**Maritime
Administration**

Office of Ship Operations

1200 New Jersey Ave., SE
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Docket No. 50-238; License No. NS-1; N.S. *SAVANNAH*

ENCLOSURE 2 PROPOSED LICENSE CHANGE (MARKED-UP)

Strikethrough indicates deletions. Text Boxes include insertions when needed.

U.S. MARITIME ADMINISTRATION

DOCKET NO. 50-238

N.S. *SAVANNAH*

Revise to Amendment 18

AMENDED FACILITY LICENSE

Amendment No. 15
License No. NS-1

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for renewal of Facility License No. NS-1 by the State of South Carolina Patriots Point Development Authority, and the U.S. Maritime Administration (the licensee[1]) dated August 20, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I

   B. The facility will be maintained in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amended license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The licensee is technically and financially qualified to engage in the activities authorized by this amended license in accordance with the rules and regulations of the Commission;

   E. The licensee has complied with the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   F. The issuance of this amended license will not be inimical to the common defense and security or to the health and safety of the public;

---

1 On June 29, 1994, the State of South Carolina Patriots Point Development Authority was deleted as a co-licensee by License Amendment 12 to License No. NS-1.

G. The possession and storage of the byproduct material as authorized by this amended license will be in accordance with the Commission's regulations in 10 CFR Part 30, including Section 30.33;

H. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility License No. NS-1 is hereby amended in its entirety to read as follows:

   A. This amended license applies to the facility owned by the U.S. Maritime Administration consisting of a pressurized water nuclear reactor (hereinafter "the reactor") and the associated components and equipment, which are located aboard the NS *SAVANNAH*, and are described in the application for license dated April 30, 1965, and amendments thereto.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the U.S. Maritime Administration:

      (1) Pursuant to Section 104b, of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, but not operate, the reactor as a utilization facility in accordance with the procedures and limitations set forth in this license; and

      (2) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to possess, but not to separate, such byproduct material as may have been produced by operation of the facility.

   C. This amended license shall be deemed to contain and is subject to the conditions specified in 10 CFR Chapter I, Part 20, Section 30.34 of Part 30 and Sections 50.54 and 50.59 of Part 50, and to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect and is subject to the following additional conditions:

      (1) The licensee shall not reactivate the reactor without prior approval of the Commission;

      (2) Deleted per Amendment 15;

      Revise to Amendment 17

      (3) <u>Technical Specification</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 14, are hereby incorporated in the license. The licensee shall possess the facility in accordance with the Technical Specifications.

INSERT:
(4) MARAD shall implement and maintain in effect all provisions of the License Termination Plan (LTP), as approved in License Amendment No. xx. MARAD may make changes to the LTP without prior NRC approval provided the proposed changes are in accordance with LTP Chapter 10, *LTP Areas That Cannot Be Changed Without NRC Approval.*

D. This amended license is effective as the date of issuance.

Insert the word "of".

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia, Director
Division of PWR Licensing-B

Enclosure:
Appendix A Technical
Specifications

Date of Issuance: July 15, 1986

U.S. Department
of Transportation
**Maritime
Administration**

Office of Ship Operations

1200 New Jersey Ave., SE
Washington, DC 20590

Docket No. 50-238; License No. NS-1; N.S. *SAVANNAH*

ENCLOSURE 3 PROPOSED LICENSE CHANGE (RETYPED)

U.S. MARITIME ADMINISTRATION

DOCKET NO. 50-238

N.S. *SAVANNAH*

AMENDED FACILITY LICENSE

Amendment No. 18
License No. NS-1

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for renewal of Facility License No. NS-1 by the State of South Carolina Patriots Point Development Authority, and the U.S. Maritime Administration (the licensee[1]) dated August 20, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I

   B. The facility will be maintained in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amended license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The licensee is technically and financially qualified to engage in the activities authorized by this amended license in accordance with the rules and regulations of the Commission;

   E. The licensee has complied with the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   F. The issuance of this amended license will not be inimical to the common defense and security or to the health and safety of the public;

1 On June 29, 1994, the State of South Carolina Patriots Point Development Authority was deleted as a co-licensee by License Amendment 12 to License No. NS-1.

G. The possession and storage of the byproduct material as authorized by this amended license will be in accordance with the Commission's regulations in 10 CFR Part 30, including Section 30.33;

H. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility License No. NS-1 is hereby amended in its entirety to read as follows:

   A. This amended license applies to the facility owned by the U.S. Maritime Administration consisting of a pressurized water nuclear reactor (hereinafter "the reactor") and the associated components and equipment, which are located aboard the NS *SAVANNAH*, and are described in the application for license dated April 30, 1965, and amendments thereto.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the U.S. Maritime Administration:

      (1) Pursuant to Section 104b, of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, but not operate, the reactor as a utilization facility in accordance with the procedures and limitations set forth in this license; and

      (2) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to possess, but not to separate, such byproduct material as may have been produced by operation of the facility.

   C. This amended license shall be deemed to contain and is subject to the conditions specified in 10 CFR Chapter I, Part 20, Section 30.34 of Part 30 and Sections 50.54 and 50.59 of Part 50, and to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect and is subject to the following additional conditions:

      (1) The licensee shall not reactivate the reactor without prior approval of the Commission;

      (2) Deleted per Amendment 15;

      (3) <u>Technical Specification</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 17, are hereby incorporated in the license. The licensee shall possess the facility in accordance with the Technical Specifications;

(4) MARAD shall implement and maintain in effect all provisions of the License Termination Plan (LTP), as approved in License Amendment No. xx. MARAD may make changes to the LTP without prior NRC approval provided the proposed changes are in accordance with LTP Chapter 10, LTP Areas That Cannot Be Changed Without NRC Approval.

D. This amended license is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia, Director
Division of PWR Licensing-B

Enclosure:
Appendix A Technical
Specifications

Date of Issuance: July 15, 1986

U.S. Department
of Transportation

**Maritime
Administration**

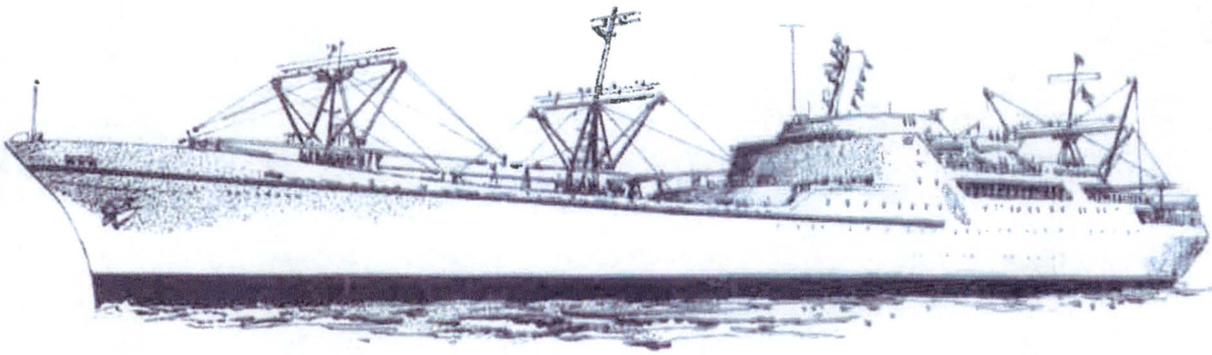
Office of Ship Operations

1200 New Jersey Ave., SE
Washington, DC 20590

Docket No. 50-238; License No. NS-1; N.S. *SAVANNAH*

ENCLOSURE 4 LICENSE TERMINATION PLAN

U.S. Department of Transportation
Maritime Administration
Office of Ship Disposal



N.S. SAVANNAH

LICENSE TERMINATION PLAN

STS-004-003

Revision 0

Approved: [signature] Date: 10/23/2023

Senior Technical Advisor

Prepared by:
NSSS, LLC

ABSTRACT

The Nuclear Ship *SAVANNAH* (NSS) is believed to be the first federally-designated historic property to undergo decommissioning under the provisions of a Nuclear Regulatory Commission (NRC) utilization facility license. It is also the only waterborne and mobile NRC-licensed site. For these and other reasons, this decommissioning project contains characteristics that uniquely differentiate it from conventional landside nuclear power facilities. This License Termination Plan (LTP), while following NRC prescribed format and content requirements, seeks to define, explain, and propose solutions to the challenges and opportunities that arise from *SAVANNAH*'s unique characteristics, while meeting the overarching objective to terminate the NS-1 license without restrictions.

Certain characteristics may simplify the nature of NSS decommissioning and license termination. These include the compact nature of the NSS site and its construction features that isolate it from much of the surrounding environment. Other characteristics may be considered complications, including the aforementioned compact and mobile (i.e., transient) site and also those resulting from NSS' designation as a National Historic Landmark (NHL). As a federally-owned NHL, it is incumbent upon the U.S. Department of Transportation, Maritime Administration (MARAD) to undertake such planning <u>and actions</u> (emphasis added) as are necessary to minimize harm to the landmark – which is a higher standard than is typically imposed on decommissioning projects, even when they involve properties with historic significance. Among the earliest agency stated goals for NSS decommissioning was a preference that the ship not be destroyed, but instead preserved if practical, following the license termination. This goal strongly influenced the design choices made by MARAD when planning decommissioning; however, in practice these choices were found to neatly complement and address some of the significant challenges that the compact and mobile site posed. The preservation goal also challenged decommissioning in that the disposition of NSS would necessarily come after the license termination by NRC. This left the eventual end state of the facility open and unknown during the decommissioning process. The LTP includes detailed discussions and analyses of three distinct end state scenarios, all of which are shown to meet the radiological criteria for unrestricted release. In this regard, MARAD has chosen to adopt the Environmental Protection Agency's 15 mrem per year standard, as described within the LTP.

Since 2018, MARAD has formally consulted with several parties under the provisions of Section 106 of the National Historic Preservation Act of 1966, as amended. This consultation ultimately resulted in a Programmatic Agreement (PA) covering both decommissioning and disposition as a single Undertaking. The PA, whose signatories include MARAD, the Advisory Council on Historic Preservation, Maryland State Historic Preservation Officer, and the NRC, was executed in March 2023. It is no longer uncommon for licensed facilities to enjoy extended operating periods which, when combined with some period of SAFSTOR, result in facilities that become historically significant with the passage of time. It seems probable that future decommissioning projects will involve facilities that possess significance to the extent that they become federally-designated historic properties (i.e., listed on the National Register of Historic Places). To that end, MARAD hopes that this LTP provides an example for what can be accomplished when two otherwise contradictory processes are thoughtfully managed and executed in a harmonious manner.

RECORD OF REVISIONS

Revision	Summary of Revisions
STS-004-003, LTP Rev. 0	Original

LIST OF EFFECTIVE PAGES

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FOREWORD

This License Termination Plan (LTP) has benefitted from the accrued experience of commercial industry and the Nuclear Regulatory Commission (NRC) since the first LTPs were submitted some twenty years ago. In preparing the LTP, MARAD carefully considered the LTP approval histories for several of the more recent commercial projects, including among others, LaCrosse, Humboldt Bay, and Zion; the Plum Brook federal site operated by the National Aeronautics and Space Administration was also considered. In general, Requests for Additional Information (RAI) generated during the LTP review and approval process tended to group into similar categories. Many, if not the preponderance, of the RAIs involved environmental affects that are not germane to N.S. *SAVANNAH* (NSS). Examples include the effects of groundwater and subsurface soil contamination, and other factors related to onsite and potential offsite terrestrial conditions. As described in the LTP, NSS is isolated from both the terrestrial and marine/aquatic environments, and so MARAD anticipates no such RAIs. Another common category of RAIs involved the location of information within the LTP itself. MARAD believes this category exists because the three NRC guidance documents, although consistent with themselves and the underlying regulations at 10 CFR 50.82, differ from each other in certain respects. The three guidance documents are:

1. Regulatory Guide 1.179, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors, Rev. 2, July 2019
2. NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans, Rev. 2, April 2018, including Appendix A
3. NUREG-1757, Volume 2, Consolidated Decommissioning Guidance – Characterization, Survey, and Determination of Radiological Criteria, Rev. 1, September 2006

Before actual writing of the LTP started, MARAD developed an internal working document that collected, in a single location, all the NRC LTP acceptance criteria. We arranged this document in chapter format, laying out the requirements from each of the guidance documents in sequential order, beginning with R.G. 1.179, and followed by NUREGs 1700 and 1757 in that order. No attempt was made to consolidate duplicate or duplicative content. The MARAD document became the working shell for the LTP. As LTP sections were written, they were inserted into the appropriate location(s) of the shell. Later, a tracking matrix was developed which served as a quality control / quality assurance mechanism for authors and reviewers. The matrix is called the **Acceptance Criteria Review Matrix (ACRM)**. As the LTP matured into a final narrative construct, the editors used the ACRM to carefully control the consolidation of duplicative content.

MARAD has included the final ACRM in the License Amendment Request that submits the LTP to NRC. It is our hope that the matrix will serve a similar purpose for NRC reviewers as it has for the MARAD authors and reviewers, and that it proves useful to NRC in the NSS LTP approval process.

ACKNOWLEDGMENTS

This License Termination Plan (LTP) is more than the collective work of several authors, editors, and reviewers. In a very real sense, it represents the culmination of decades of work and service by many individuals too numerous to name, who conceived, designed, constructed, operated, and finally decommissioned the world's first nuclear-powered merchant ship. It thus comes with no small irony that as the N.S. *SAVANNAH* (NSS) licensed lifecycle nears conclusion, the world maritime community is embracing with renewed interest and vigor, the potential use of commercial nuclear propulsion to significantly reduce carbon and other greenhouse gas emissions from merchant ships. While this objective was not part of the NSS program, meeting the objective must, of necessity, draw upon the experiences and precedents that NSS established, and which were further developed by West Germany's *Otto Hahn*. To that end, MARAD looks forward to preserving and disseminating the collected knowledge and experience of the NSS program, so that it can be used for its original purpose – to advance the development of nuclear-powered merchant ships for the benefit of humankind. In this way, we hope to continue and expand the legacy of the outstanding group of men and women who put the once outrageous idea of using the power of the atom to propel a peaceful, commercial ship into practical effect, who then explored all the many unforeseen challenges associated with the idea, and who ultimately safeguarded both the ship and its knowledge base through the long decades of protective storage and decommissioning. This LTP, as the culmination of their work and legacy, is dedicated to them.

I will take the liberty of exercising my prerogative as signatory to acknowledge the contributions of my contemporary staff, mentors and predecessors. It seems inevitable that some persons will be missed in the narrative; and in some cases, others may not be named out of narrative clarity; any such omissions are my own. In many respects, the NSS lifecycle is bracketed by two careers; my own, and that of Zelvin Levine. Zel graduated from Georgia Tech with a Doctorate in Chemical Engineering, and having completed the then-newly created School of Reactor Technology at Oak Ridge. He joined Babcock & Wilcox in 1955, and was one of the first employees assigned to the NSS project. In that sense, his career with NSS began virtually at the beginning of the program. Zel was responsible for significant engineering designs and safety analyses to support operating authorizations and licensing. He left B&W for the Martin Company on the STURGIS project before joining MARAD in 1969, and worked through several progressively more senior positions until he was named Assistant Administrator for Research and Development. He acted as MARAD's NSS licensee from about 1975 until his retirement at the end of 1998. I joined MARAD in 1991, and was assigned to the NSS in 1993 to supervise its removal from the Patriots Point Naval and Maritime Museum. I became the project engineer on a staff of about three persons, the other being Joseph Seelinger, at the time the Chief of the Division of Ship Maintenance and Repair, and a former NSS Reactor Operator from the 1963-1964 MARAD backup crew. I worked directly with Zel as MARAD formulated the license amendment request associated with the relocation of NSS, and the removal of Patriots Point as co-licensee. Zel, Joe and I continued to act as the MARAD licensee organization until Zel's retirement. Joe then became licensee, and I followed five years later when Joe retired in January 2004. By that time, the NSS DECON project had been authorized, and a small nucleus staff had been stood up. Baring anything unforeseen or untoward, I expect to complete my tenure as licensee upon the license termination.

Maritime Administrator Bill Schubert must be credited with the executive act of authorizing decommissioning in February 2002. He is followed immediately by Jim Caponiti, our Associate Administrator for National Security at the time, who championed the project and sought the resources necessary to develop it. Credit is also due to Bill Trost, the Director of the Office of Ship Operations, who had programmatic responsibility for NSS, and who endorsed the creation of the dedicated *SAVANNAH*

Technical Staff (STS) organizational unit in 2005. Their successors, and others in MARAD senior management have continued to support NSS operations, and to advance the cause of decommissioning after the late 2006 suspension described in the PSDAR. To that end, credit is due to Administrator Paul Jaenichen, who sought the resumption of decommissioning beginning in 2013, and Associate Administrator Kevin Tokarski who also championed the project and supported the staff.

The immediate STS organization has never numbered more than five direct MARAD employees at any time. In addition to myself, our original staff included John Wiegand, Decommissioning Program Manager, Greg Thornton, Facility Site Manager, and Rick Volkmann, Marine Surveyor, and technical writer. Our Contracting Officer was Gene Simmons from 2003 to his retirement in 2013. In the four short years of 2003 to 2007, we brought together contractors and federal partners to create a licensee organization virtually from scratch, design the DECON-LT project, and resurrect the ship itself from the doldrums of the James River Reserve Fleet. The strength of the initial planning is evidenced by the fact that the DECON-LT project started in 2017 is fundamentally unchanged from the 2006 design. Only the administrative and contracting details have changed. Other MARAD STS staff have included Robert Falk and Caleb Soeun as DPM, and Jim Brown as FSM.

By design, the STS is a blended organization that includes contractors in key roles, and partner organizations supporting decommissioning activities and providing oversight. We established long-term relationships with the Department of Energy's Argonne National Lab and the DOT Volpe Center in that period of 2003 to 2007. Larry Boing, Mike Buonopane, Bill Halloran, and Chris Zevitas were the principals from those organizations, along with Chuck Felhauer. Our initial decommissioning planning contractor, WPI, was represented by Jon Stouky, a pioneer in the decommissioning industry, and himself a former NSS program participant as the operator of the servicing barge *Atomic Servant*, and in the post of Nuclear Advisor. John Bowen and Franco Godoy were part of the WPI team. Jon and John would remain with the NSS program well beyond the completion of the WPI contracts in 2007. Both served in the contractor key positions of Nuclear Advisor, and John later served as the contract Project Manager during the TOTE Services bridge contract from 2018 – 2021.

We added Areva Federal Services as our Engineering and Management Oversight Services contractor in 2007, along with Radiation Safety and Control Services as our radiological protection and emergency response contractor. Keystone Shipping Company was assigned as our General Agent for ship husbandry, custodial care, conversion and modifications, and marine services. Areva was hired to perform the detail design for DECON, but that focus turned to SAFSTOR when the budgetary suspension struck. Harvey Story, Don McGee, Doug Roberson, Skip Litterer, Lars Flink and Trevor Nancarrow represented these companies, with Doug, Don and Trevor returning over the course of time. Doug and Don are both active with the project today.

In 2005 and 2006 MARAD acquired "license technical support" services from Sayres and Associates – a relationship which ran until September 2012, and which produced the most significant long-term staffing relationships, with John Osborne as Licensing & Compliance Manager, Art Paynter as Quality Assurance Manager, and Bob Sheranko as Business and Risk Manager. Both John and Art continue in their roles, while Bob retired in 2019. In the position of Nuclear Advisor were first Jon Stouky, and then Pat McConnell. Among them and together with the direct staff, and incumbent contractors, MARAD created a contemporary licensing program by reconstituting our fundamental basis documents, modernizing the Technical Specifications via license amendment requests, and crafting an administrative and technical procedures program based on industry standards and best practices. Our current decommissioning program could not exist without the hard work and dedication of these individuals and companies.

The non-profit N/S SAVANNAH Association was created as a successor to the informal crew reunion organization that had existed from the early 1980s. The NSSA provided considerable assistance to decommissioning planning efforts, including sponsoring a reunion onboard NSS in 2008 during which many interviews were conducted in support of the Historic Site Assessment. The NSSA continues to support NSS operations to this day, with our appreciation and gratitude.

Considering the importance and standing of NSS as a National Historic Landmark, and its place within the LTP, I would be remiss in omitting the persons who helped with our consultative efforts. Barbara Voulgaris, recently retired as MARAD's Senior Historian and Federal Preservation Officer, guided the consultation effort. She was aided by Thomas Shepherd, formerly of MARAD's Office of Chief Counsel, and Rand Pixa, MARAD's recently retired Deputy Chief Counsel. Ainsley Parrish, Justin Valentino, and Wendy Coble support the ongoing execution of the Programmatic Agreement. I am assisted by Anne Jennings, as a contractor with Sustainable Design Consortium. Argonne National Lab's Konnie Westcott and Dan O'Rourke also made significant contributions to MARAD's NHPA compliance effort. MARAD also appreciates the support provided by the PA signatories and concurring parties, especially with respect to the decommissioning activities portion of the NHPA Undertaking as described within the LTP.

In the period between 2008 and 2017, other contracts were awarded to support STS and NSS operations. The first integrated management contract was awarded in 2013 to Tote Services. This contract combined the formerly separate ship husbandry, radiological protection, and license technical support functions. The Phase I decommissioning effort began in the final year of this contract, and continued under a bridge contract from 2018 to 2021. Principals during this period included Tom McIntyre and John Bowen as Project Manager, Jim Byrne and John Bowen as Nuclear Advisor, Cliff Marks in several positions, and Herb Evans as Radiation Safety Officer.

Two engineering support contracts are in place to provide independent engineering review and oversight, field inspection services, and multidisciplinary regulatory support. Under the B2Z Engineering Contract are John Wiegand, retired from MARAD and drawing upon his many years of *SAVANNAH* experience, and Don McGee of Orano Federal Services (successor to Areva). Sustainable Design Consortium provides the field inspection services with Tony Margan and Robbie McCready, and independent review with Jim Reese and Bruce Reynolds, both via Tidewater.

Under the 2021 DECON-LT integrated services contract, the current key personnel include Matt Arsenault, Project Manager, Nick Waltz, Nuclear Advisor and BRM, Jerry Toumey, Decommissioning Manager, Scott Ginter, RSO, Art Paynter, QAM and John Osborne, LCM. Jay Tarzia, Ron Thurlow and Todd Eiler are the principal representatives of the joint venture partners RSCS and ESFS. The current MARAD senior management and headquarters staff with programmatic responsibility for NSS operations include Deputy Associate Administrator for Federal Sealift Doug Harrington, Director of Fleet Program Management Melinda Simmons-Healey, Program Manager Laila Linares, Director of Acquisitions Bruce Markman, Contracting Officer Ken Egbuna, and Contracting Specialist Lisa Miles. And of course, the MARAD senior leadership team including Maritime Administrator Ann Phillips, Deputy Administrator Tamekia Flack, and Executive Director Jack Kammerer, provide continuing leadership and support to NSS activities, including the license termination effort.

With respect to the LTP itself, I wish to thank the authors, contributors, and reviewers for their equally hard work and dedication. The LTP development team was led by Eric Darois of RSCS. John Osborne and Robert Falk developed the Acceptance Criteria Review Matrix. The principal LTP author / editor is John Osborne, and contributors include Robert Falk, Pete Hollenbeck, Heath Downey, Rob Grant, Art Paynter, and Caleb Soeun. Independent reviews were provided by Dave Fauver (RSCS), Don McGee (Orano via B2Z Engineering), John Wiegand (B2Z), James Reese (Tidewater), Robert Yetter (Energy*Solutions*), and Cameron Davis (ES). Three formal pre-submittal meetings were held and included among MARAD attendees not named above Jay Tarzia of RSCS.

For the entirety of the current NSS program, Cynthia Bearor has stood alongside me as a charter and integral member of the STS. Her organizational title of Documentation and Administrative Support Manager belies her importance; it is a holdover from the early days. Among many other things, she functions as the MARAD Site Safety Officer, and serves as either the Contracting Officer's Representative, or ACOR, for all our active contracts. For a long time, she and I were the only MARAD direct employees on the STS. To say that this project rests equally on her shoulders as mine would be an understatement because it fails to acknowledge how deeply I have come to rely on Cindy's support.

On November 8, 2022, we held a ceremony immediately before the Reactor Pressure Vessel lift. The ceremony commemorated our colleagues and program partners who did not live to see the day. A magnetic plaque was affixed to the RPV canister, and remains attached at Clive. Regrettably, we added another member to the list earlier this year. In their honor, I will end this acknowledgement by repeating their names herein.

Erhard W. Koehler
October 2023

SAVANNAH TECHNICAL STAFF
IN MEMORIAM
2002 – 2023

Helen Delich Bentley

Wayne Britz

Robert Falk

Lars Flink

Marvin Gordon

CAPT Moses Hirschkowitz

Fred Hoffman

Dr. Zelvin Levine

Walt Mathers

RADM Lauren S. McCready

Bob Moody

Joseph H. Seelinger

Gene Simmons

Harvey Story

Jon Stouky

Stanley D. Wheatley

ABBREVIATIONS AND ACRONYMS

Note to Reviewers: Acronyms will be redefined on first use in each chapter.

ACRM - Acceptance Criteria Review Matrix

AEC - Atomic Energy Commission

ALARA - As Low As Reasonably Achievable

c/d - counts per disintegration

CEQ - Council on Environmental Quality

CRD - Control Rod Drive

CV - Containment Vessel

DCF - Dose Conversion Factors

DCGL - Derived Concentration Guideline Level

DQA - Data Quality Assessment

DQAP - Decommissioning Quality Assurance Plan

DQO - Data Quality Objective

EA - Environmental Assessment

EAF - Electric Arc Furnace

EDE - Effective Dose Equivalent

EPA - Environmental Protection Agency

FSS - Final Status Survey

GEL - GEL Laboratories, LLC

HSA - Historical Site Assessment

HTD - Hard to Detect

IDC - Insignificant Dose Contributors

ISOCS - In Situ Object Counting System

LAR - License Amendment Request

LBGR - Lower Bound of the Gray Region

LLBP - Less Likely but Plausible

LLRW - Low Level Radioactive Waste

LTP - License Termination Plan

MARAD - U.S. Department of Transportation, Maritime Administration

MDA - Minimum Detectable Activity

MDC - Minimum Detectable Concentration

MDCC - Minimum Detectable Corrected Counts per minute

N/A - Not Applicable

NEPA - National Environmental Policy Act

NHL - National Historic Landmark

NHPA – National Historic Preservation Act

NRC - Nuclear Regulatory Commission

NRHP – National Register of Historic Places

NSS - Nuclear Ship *SAVANNAH*

NST - Neutron Shield Tank

OSHA - Occupational Safety & Health Administration

PPDA - Patriots Point Development Authority

PSDAR - Post Shutdown Decommissioning Activities Report

RASS - Remedial Action Support Survey

RCA - Radiologically Controlled Area

ROC - Radionuclide of Concern

SCM - Surface Contamination Monitor

SEA - Supplemental Environmental Assessment

SOF - Sum of the Fractions

USN - U.S. Navy

VSP - Visual Sample Plan

WRS - Wilcoxon Rank Sum

1 GENERAL INFORMATION

1.1 Identifying Information

Facility:	Nuclear Ship *SAVANNAH*; License NS-1; Docket No. 50-238
Location:	Baltimore City, Maryland
Address:	Pier 13 Canton Marine Terminal; 4601 Newgate Ave; Baltimore, MD 21224
Owner / Licensee:	Maritime Administration, U.S. Department of Transportation
Responsible Official:	Erhard W. Koehler, Senior Technical Advisor, N.S. *SAVANNAH*

1.2 Introduction

This License Termination Plan (LTP) is submitted by the Maritime Administration (MARAD) as licensee for the Nuclear Ship *SAVANNAH* (NSS). As described more fully in the Updated Final Safety Analysis Report (UFSAR) [Reference 1-1], NSS is the world's first nuclear-powered merchant ship. It was conceived and constructed during the Eisenhower Administration as a signature element of the *Atoms for Peace* Program and remains a powerful symbol of this nation's commitment to advance the peaceful uses of nuclear technology for the betterment of humankind. After successfully completing its programmatic goals and objectives, NSS was removed from service in 1970, defueled in 1971, and made permanently inoperable by 1976, as represented by issuance of a Possession-only License.

NSS was operated as a museum from 1981 to 1994, during which time the State of South Carolina, acting through the Patriots Point Development Authority (PPDA), was a co-licensee with MARAD.[1] During the museum period, NSS was first named to the National Register of Historic Places (NRHP, 1983), and then designated as a National Historic Landmark of the United States (NHL, 1991). In seeking to complete the NSS licensed lifecycle through decommissioning and license termination, MARAD has been sensitive to NSS' historic significance, and taken affirmative steps (see Section 1.4 of this chapter, and Section 3.1.1 of Chapter 3) to meet the objectives of the National Historic Preservation Act (NHPA) of 1966, as amended, with particular emphasis given to minimizing harm to NSS.[2] As should be apparent, the statutory requirement to minimize harm to the landmark does not clearly align with the traditional approaches taken in power reactor decommissioning and license termination. MARAD formally addressed this conflict in its 2008 Post Shutdown Decommissioning Activities Report (PSDAR) [Reference 1-2]. Whereas the PSDAR postulated methods by which the NHPA objectives might be met, this LTP uses detailed discussions of end-state scenarios and approaches towards meeting license termination radiological criteria to describe how MARAD will meet its NHPA objectives.

MARAD believes that the NSS decommissioning project is a first, in that the subject of the project is already a federally designated historic structure, as opposed to a structure deemed eligible for designation. MARAD and NRC are signatories, along with the Advisory Council on Historic Preservation (ACHP) and the Maryland State Historic Preservation Officer (MD SHPO), to a Programmatic Agreement (PA)

[1] A 1981 statute authorized MARAD to bareboat charter NSS to the State of South Carolina for use as a museum ship at the Patriots Point Naval and Maritime Museum in Mount Pleasant. A bareboat charter is a form of lease employed in the maritime industry. The museum was (and still is) operated by the Patriots Point Development Authority (PPDA), a state corporation. Because NSS was still licensed by NRC, the statute allowed for PPDA to become a co-licensee to exercise physical custody and control of the ship (as the NRC licensed site), and to manage NRC-required radiological protection and emergency response requirements. MARAD held title to NSS and was responsible for underwater hull maintenance. The charter and museum service ended in May 1994 when MARAD removed NSS for drydocking and underwater hull repairs. PPDA was removed as a co-licensee by Amendment 12 in July 1994.

[2] Section 110(f) of the NHPA provides as follows: "Prior to the approval of any Federal undertaking that may directly and adversely affect any National Historic Landmark, the head of the responsible Federal agency shall to the maximum extent possible undertake such planning and actions as may be necessary to minimize harm to the landmark."

[Reference 1-3] covering the combined decommissioning and disposition of NSS as an NHPA Undertaking. This LTP supports both aspects of the Undertaking, by describing how MARAD can meet the license termination criteria, and demonstrating how that license termination supports the potential NSS disposition end-states, including preservation. Reviewers should bear this in mind when reading the LTP.

This plan has been developed to address the license termination requirements of 10 CFR 50.82 and the radiological release criteria of 10 CFR Part 20, Subpart E. This plan is a supplement to the UFSAR [Reference 1-1] and is provided as an enclosure to License Amendment Request 2023-001. In preparing the LTP, MARAD followed the requirements of the Nuclear Regulatory Commission (NRC) guidance documents listed below:

1. Regulatory Guide 1.179, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors, Rev. 2, July 2019 [Reference 1-4]
2. NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans, Rev. 2, April 2018, including Appendix A [Reference 1-5]
3. NUREG-1757, Volume 2, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Rev. 1, September 2006 [Reference 1-6]

1.3 Purpose

The purpose of the LTP is to demonstrate compliance with NRC decommissioning and license termination requirements. MARAD has prepared the LTP in accordance with References [1-4], [1-5] and [1-6] to demonstrate that its decommissioning and license termination project meets the intent of the applicable NRC regulations. Regarding the regulations, the LTP:

- Satisfies the content requirements of 10 CFR 50.82(a)(9)(i) and (ii), as described in Chapters 1 through 9; and,
- Satisfies applicable portions of the requirements of 10 CFR Part 20, Subpart E, Radiological Criteria for License Termination. Specifically, the LTP satisfies the requirements of the following:
    - 10 CFR 20.1401 General provisions and scope; and,
    - 10 CFR 20.1402 Radiological criteria for unrestricted use.

Furthermore, MARAD affirms its understanding that the license cannot be terminated in fewer than two years from submittal of this LTP, in accordance with 10 CFR 50.82(a)(9)(i). MARAD anticipates requesting license termination to be effective in December 2025, provided all prerequisite actions are complete at that time.

By following References [1-4], [1-5] and [1-6], MARAD is confident that NRC will approve the LTP in accordance with 10 CFR 50.82(a)(10) after finding that:

- Completing the remainder of decommissioning activities will be performed in accordance with NRC regulations;
- Completing decommissioning will not be inimical to the common defense and security;
- Completing decommissioning will not be inimical to the health and safety of the public; and,
- Completing decommissioning will not have a significant effect on the quality of the environment.

1.4 Decommissioning Objectives

MARAD's principal decommissioning objective is to terminate license NS-1 without restrictions. Although postulated in the PSDAR [Reference 1-2], MARAD does not intend to request license termination under restricted conditions or by using alternate criteria.

MARAD will conduct remediation and survey operations such that it can submit a request to the NRC for unrestricted release of the site in accordance with Subpart E of 10 CFR Part 20 after meeting the requirements of 10 CFR 20.1402, Radiological Criteria for Unrestricted Use. The LTP documents the process that will be used to demonstrate that the dose from residual radioactivity that is distinguishable from background radioactivity does not exceed 25 mrem/year[3] to the Average Member of the Critical Group (AMCG) from all applicable pathways over a 1,000-year period and that residual radioactivity has been reduced to levels that are "As Low As Reasonably Achievable" (ALARA).

At the time of LTP submittal, MARAD has essentially completed all planned dismantlement activities (see Chapter 3), and can state with confidence that it has achieved its NHPA objectives to conduct decommissioning activities in a manner that respects (i.e., preserves, maintains, and minimizes impacts to) the historic fabric of the ship, such that decommissioning does not negatively impact potential future preservation uses. Subject to confirmation by NRC, the end-state condition of the nuclear power plant structures and components (See Section 3.1.1. of Chapter 3) also meets the NHPA objective to minimize harm to the character-defining features of the nuclear power plant, without preventing their radiological release or subsequent disposal via either shipbreaking or artificial reefing.

1.5 Plan Summary

The LTP is arranged in ten (10) chapters, as described below.

1.5.1 Chapter 1, General Information

LTP Chapter 1 describes the process used to meet the requirements for terminating the 10 CFR Part 50 license and to release the NSS for unrestricted use. The LTP has been prepared in accordance with the requirements in 10 CFR 50.82(a)(9) and is submitted both as an enclosure to support License Amendment Request (LAR) 2023-01 for the NRC to approve the LTP and as a supplement to the UFSAR.

1.5.2 Chapter 2, Site Characterization

LTP Chapter 2 discusses the Historical Site Assessment (HSA) and characterization activities that have been conducted to determine the nature and extent of radioactive contamination on the ship prior to remediation. The information obtained from the characterization provided guidance for decontamination and remediation planning.

Data from subsequent characterization may be used to change the original classification of an area, within the requirements of this LTP, up to the time of Final Status Surveys (FSSs), as long as the classification reflects the level of residual activity existing prior to any remediation in the area. This chapter satisfies the underlying intent of 10 CFR 50.82(a)(9)(ii)(A).

1.5.3 Chapter 3, Identification of Remaining Site Dismantlement Activities

LTP Chapter 3 identifies the remaining site dismantlement and decontamination activities. The information includes those areas and equipment that need further remediation and an assessment of the potential radiological conditions that may be encountered. Estimates of the occupational radiation dose and the quantity of radioactive material to be released to unrestricted areas during the completion of the scheduled tasks are provided. The projected volumes of radioactive waste that will be generated are also included. This chapter satisfies the underlying intent of 10 CFR 50.82(a)(9)(ii)(B).

[3] As described elsewhere in the LTP, MARAD has adopted a 15 mrem/year standard for NSS. Within the LTP, direct citations of regulations (i.e., 25 mrem/year) should be interpreted to mean 15 mrem/year unless stated otherwise. Additionally, MARAD is considering the post-LT life cycle for NSS to be 70 vice 1,000 years.

1.5.4 Chapter 4, Remediation Plans

LTP Chapter 4 discusses the various remediation techniques that may be used during decommissioning to reduce residual contamination to levels that comply with the release criteria in 10 CFR 20.1402. This chapter also discusses the ALARA evaluation and the impact of remediation activities on the Radiation Protection Program (RPP). It satisfies the underlying intent of 10 CFR 50.82(a)(9)(ii)(C).

The selected remediation methods used are dependent upon the contaminated material and extent of contamination. The principal materials that may be subject to remediation are structural surfaces.

Note that there are no embedded pipes, soils, surface water or groundwater at NSS. Remediation techniques that may be used for structural surfaces include standard and high pressure washing, wiping, grit blasting, chemical stripping, grinding, and other methods.

1.5.5 Chapter 5, Final Status Survey Plan

LTP Chapter 5 presents the final survey process used to demonstrate that the NSS complies with radiological criteria for unrestricted use specified in 10 CFR 20.1402 (i.e., annual dose limit of 25 mrem/year to AMCG plus ALARA). This chapter also describes proposed control mechanisms to ensure areas are not re-contaminated. This chapter satisfies the underlying intent of 10 CFR 50.82(a)(9)(ii)(D).

The FSS Plan describes use of the Data Quality Objectives (DQO) process in designing surveys, survey methods and instrumentation, data collection and processing, and data assessment and compliance.

1.5.6 Chapter 6, Compliance with the Radiological Criteria for License Termination

LTP Chapter 6 presents the radiological information and methods used to demonstrate compliance with the radiological criteria for license termination and to release the NSS for unrestricted use. Chapter 6 discusses the following:

- Radionuclides potentially present and mixture fractions;
- Exposure pathways;
- Computational models used for dose modeling;
- Sensitivity analysis and deterministic parameter selection;
- Derived Concentration Guideline Level (DCGL) and Dose Factors; and,
- Basis for the selected reasonably foreseeable and less likely but plausible scenarios.

This chapter satisfies the underlying intent of 10 CFR 50.82(a)(9)(ii)(E).

1.5.7 Chapter 7, Update of the Site-Specific Decommissioning Costs

LTP Chapter 7 provides a detailed discussion of MARAD's primary decommissioning services contract, other remaining costs, and available decommissioning funding that satisfies the underlying intent of the 10 CFR 50.82(a)(9)(ii)(F) to demonstrate that sufficient resources are available to complete the remaining decommissioning activities and release the NSS for unrestricted use.

1.5.8 Chapter 8, Supplement to the Environmental Report

LTP Chapter 8 describes MARAD's evaluations of the environmental impacts of decommissioning and license termination, including its 2008 Environmental Assessment [Reference 1-7] and 2019 Supplemental Environmental Assessment [Reference 1-8]. As a federal agency, MARAD is required by the National Environmental Policy Act of 1969 to evaluate the proposed action of

decommissioning and terminating the license of the NSS, independent of any such evaluation by NRC. MARAD has determined that there are no significant environmental impacts for decommissioning the NSS. Furthermore, MARAD affirms that there is no new information or significant environmental changes associated with the site-specific termination activities up to the submission of the LTP that is not contained in its 2008 and 2019 assessments; nor does MARAD expect there to be any new information or significant environmental changes in the period leading to license termination. This chapter satisfies the underlying intent of 10 CFR 50.82(a)(9)(ii)(G)

1.5.9 Chapter 9, Portions of the Facility that were Released prior to LTP Approval

LTP Chapter 9 describes the portions of the facility that were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H). No parts of the facility were released for use before approval of the LTP.

1.5.10 Chapter 10, Regulatory Notifications of Changes

LTP Chapter 10 lists the LTP areas that cannot be changed without NRC approval. The LTP areas applicable to NSS are derived from Appendix B of NUREG 1700 [Reference 1-5].

1.6 Facility and Site Description

The NSS was an 80 MW_{th} pressurized-water nuclear reactor. MARAD is owner and licensee of NSS. The MARAD Headquarters is located at 1200 New Jersey Ave., SE, Washington, DC. The NSS is located at Baltimore Harbor, Pier 13 in the Canton industrial district of the port, near the Seagirt Marine Terminal in Baltimore, Maryland. Note that the City of Baltimore is an independent jurisdiction surrounded by the similarly-named Baltimore County. The street address is 4601 Newgate Avenue, Baltimore, MD 21224.

The site is licensed under Possession-only License No. NS-1, Docket No. 50-238. The licensed site of the NSS is the boundary defined by the ship's hull (see Figures 1-1 through 1-7). There have been no changes to the original site boundary. The site (i.e., the ship) contains no soils or ground water. Given that the site is bounded by the ship's hull, there is no map that shows the detailed topography of the site.

Because the NSS is mobile and water borne, the off-ship characteristics of the site vary with the location of the vessel. The ship's principal characteristics are shown in Table 1-1 (see page 12).

Figure 1-1 NSS at Pier 13 in 2023

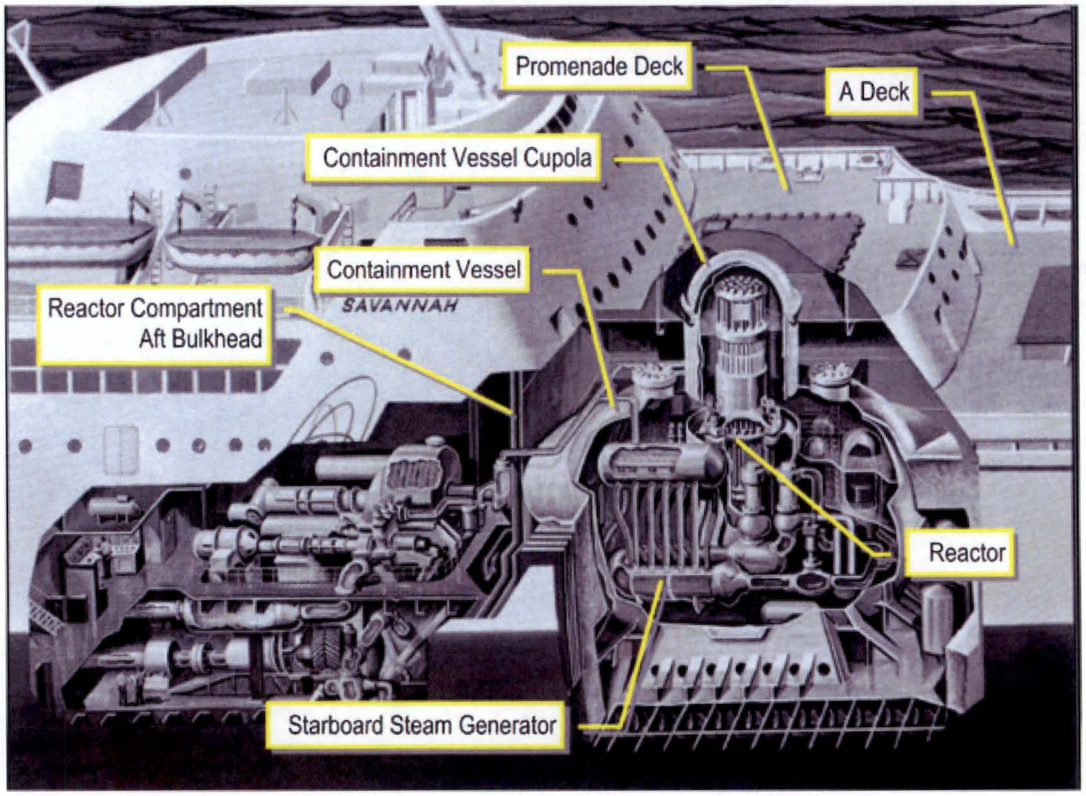

Figure 1-2 Containment, Reactor Compartment and Engine Room (Frames 99-148)

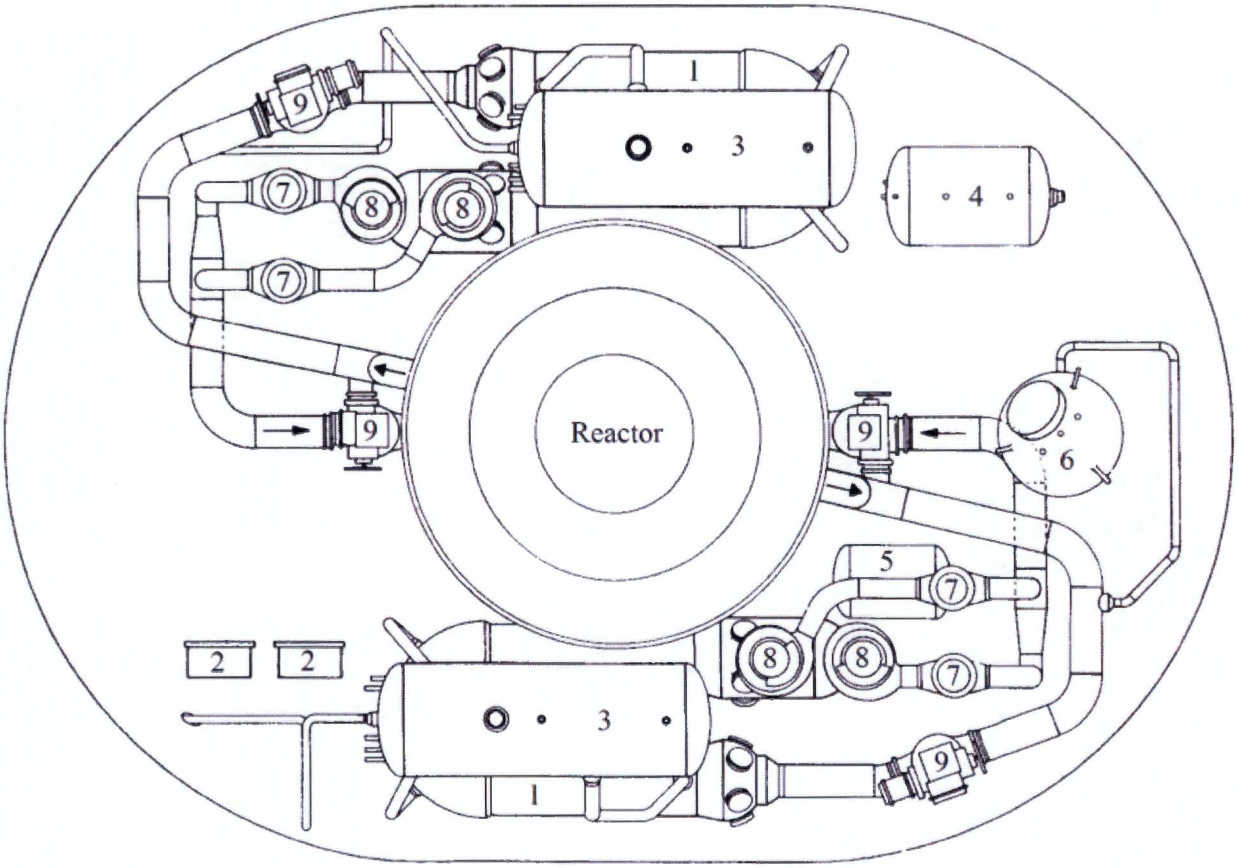

Figure 1-3 Arrangement of Principal Components in Containment

1	Heat Exchanger (Steam Generator U-tube bundle)	2	Let Down Cooler	3	Steam Drum (Steam Generator)
4	Effluent Condensing Tank	5	Containment Drain Tank	6	Pressurizer
7	Check Valve	8	Primary Pump	9	Gate Valve

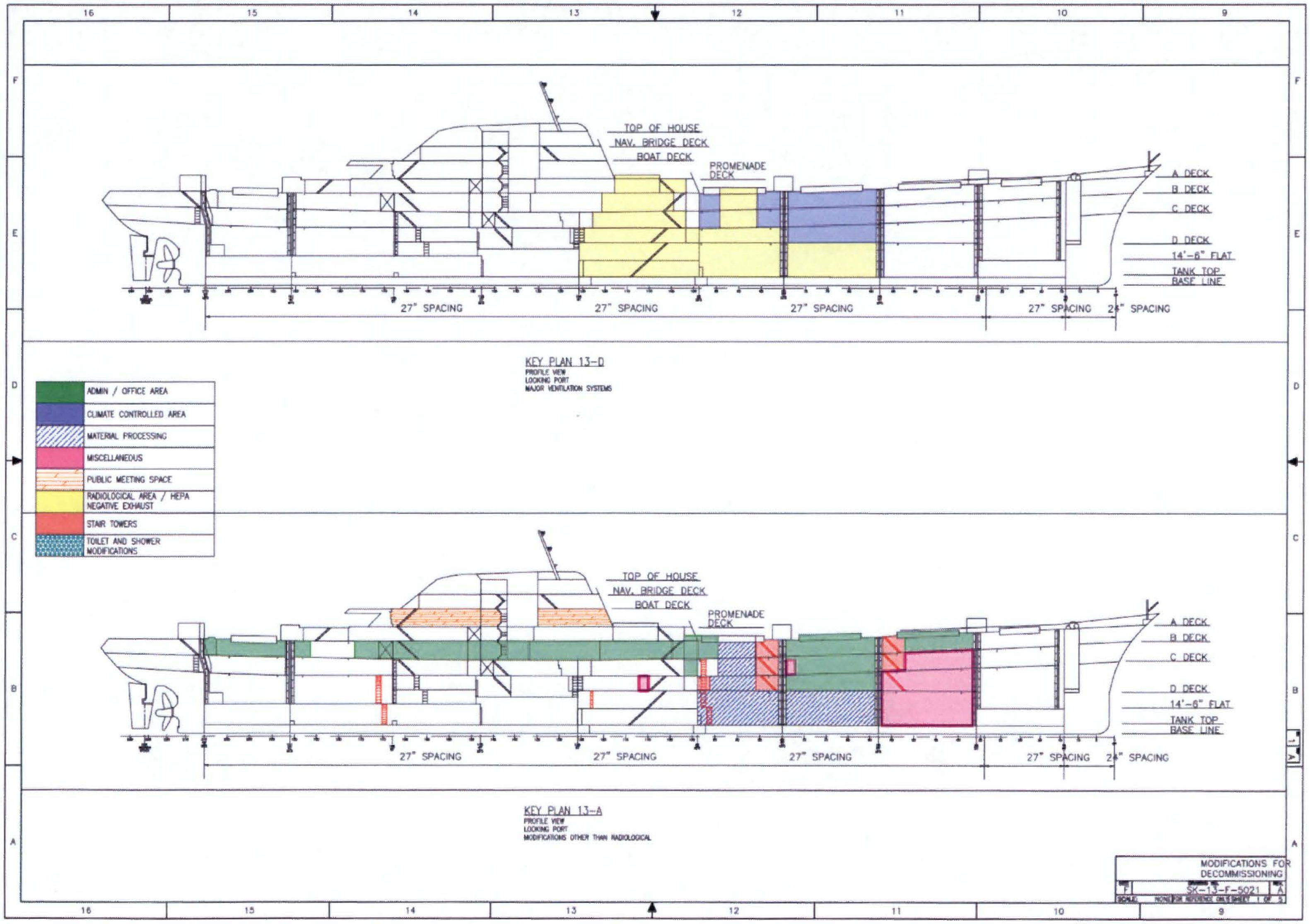

Figure 1-4 General Arrangement – Inboard Profile

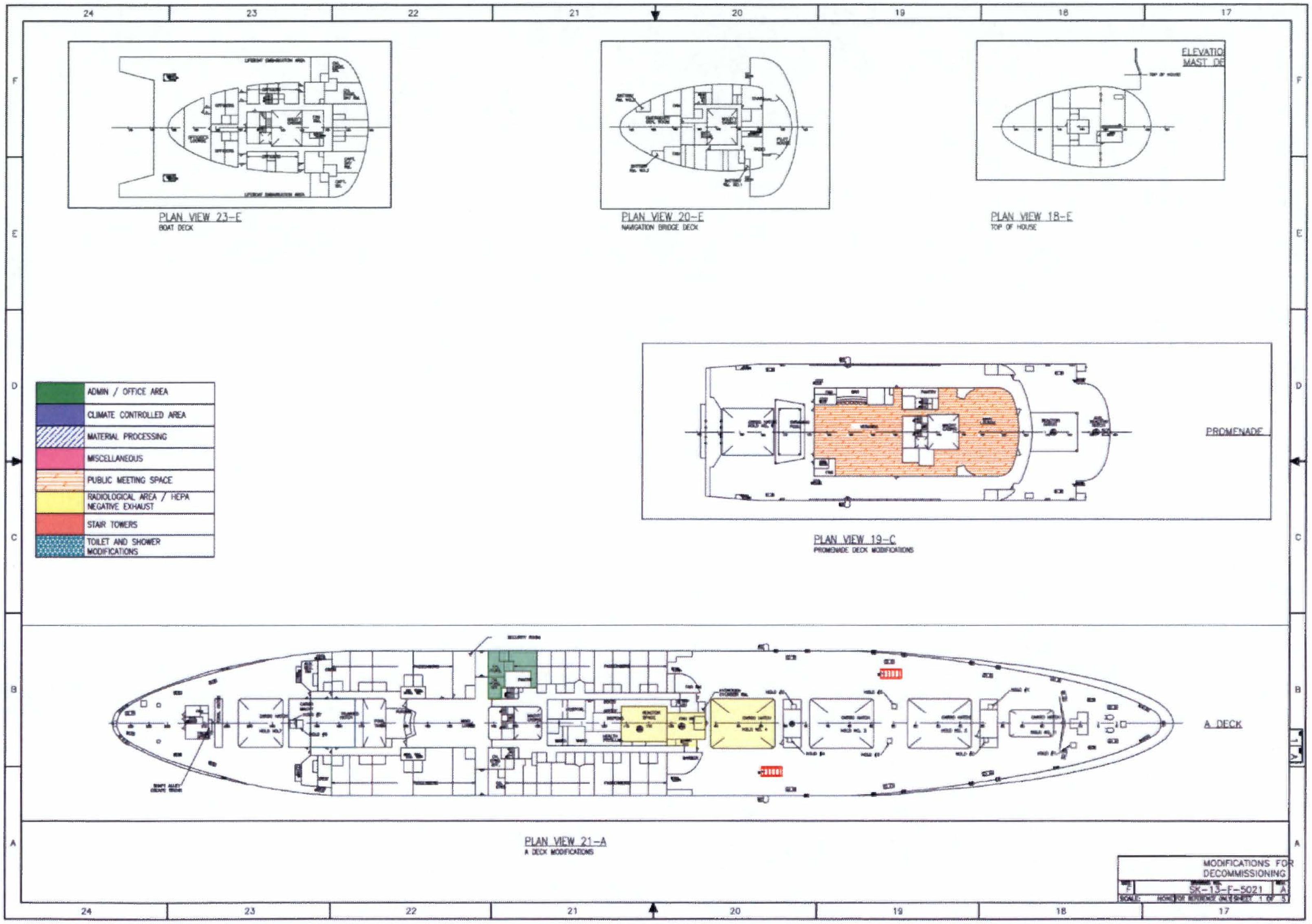

Figure 1-5 General Arrangement - Decks - Top of House to A Deck

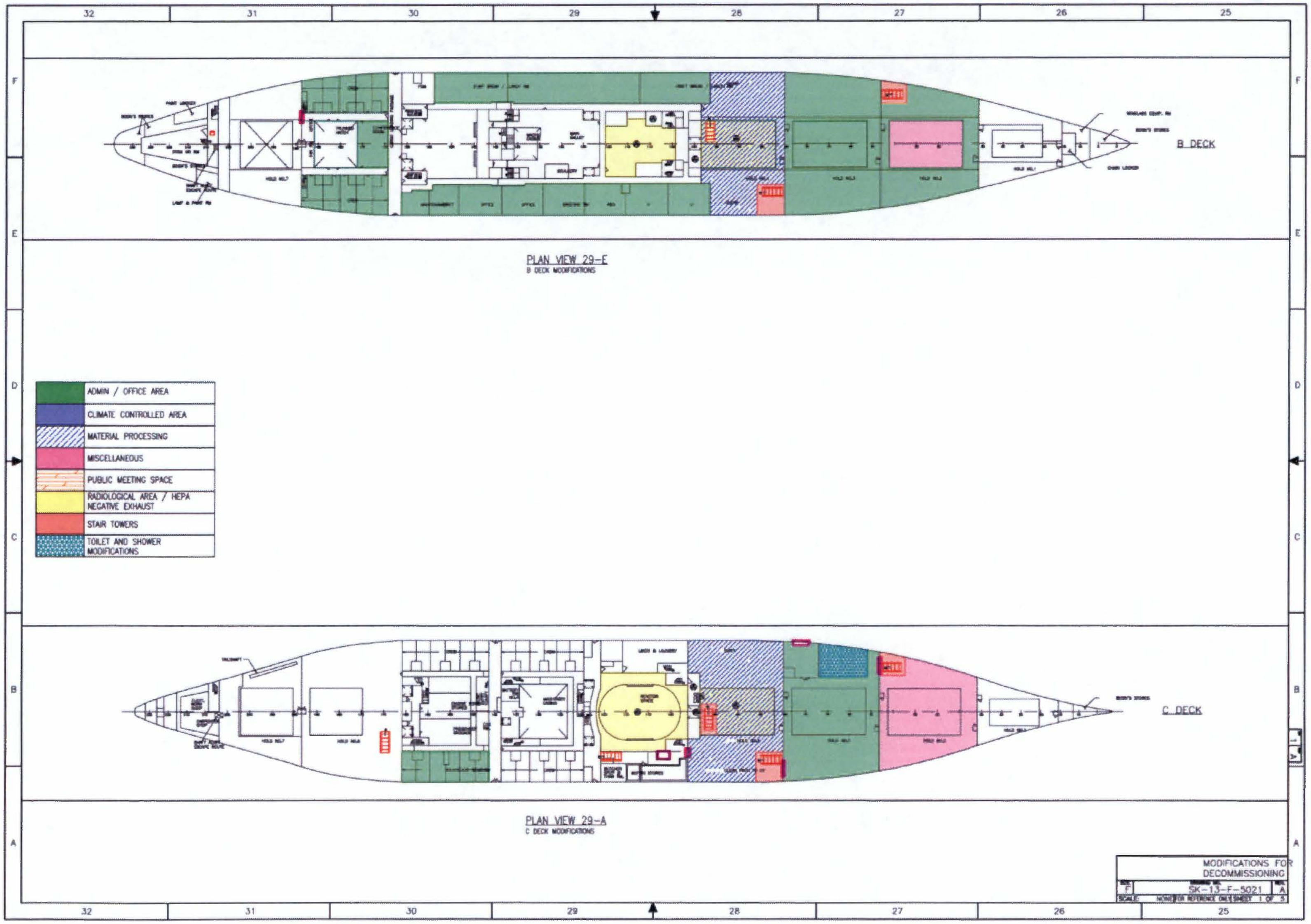

Figure 1-6 General Arrangement - Decks - B and C

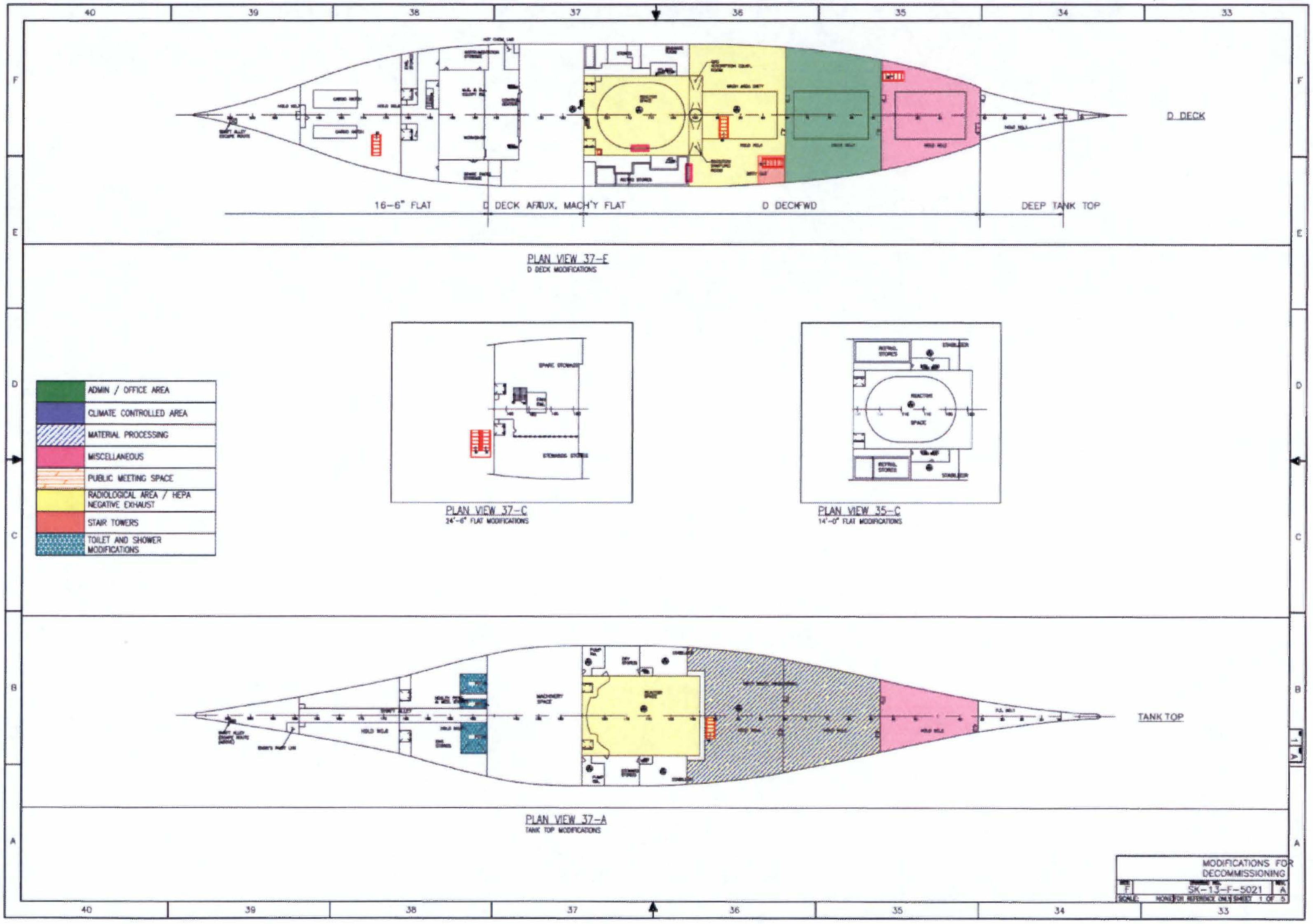

Figure 1-7 General Arrangement - Decks - D and Tank Top

Table 1-1 Principal Characteristics of the N.S. *SAVANNAH*

Dimensions	Feet
Length Overall	595.5
Length Between Perpendiculars	545
Beam, Maximum	78
Height, baseline to weather deck (A-deck)	50
Height, baseline to top of house	85
Draft, design	29
Draft, light ship (approximate)	18.33
Displacement and Tonnage in Long Tons	
Net carrying capacity, total deadweight tons	9,656
Full load displacement, tons	21,990
Light ship, tons	12,334

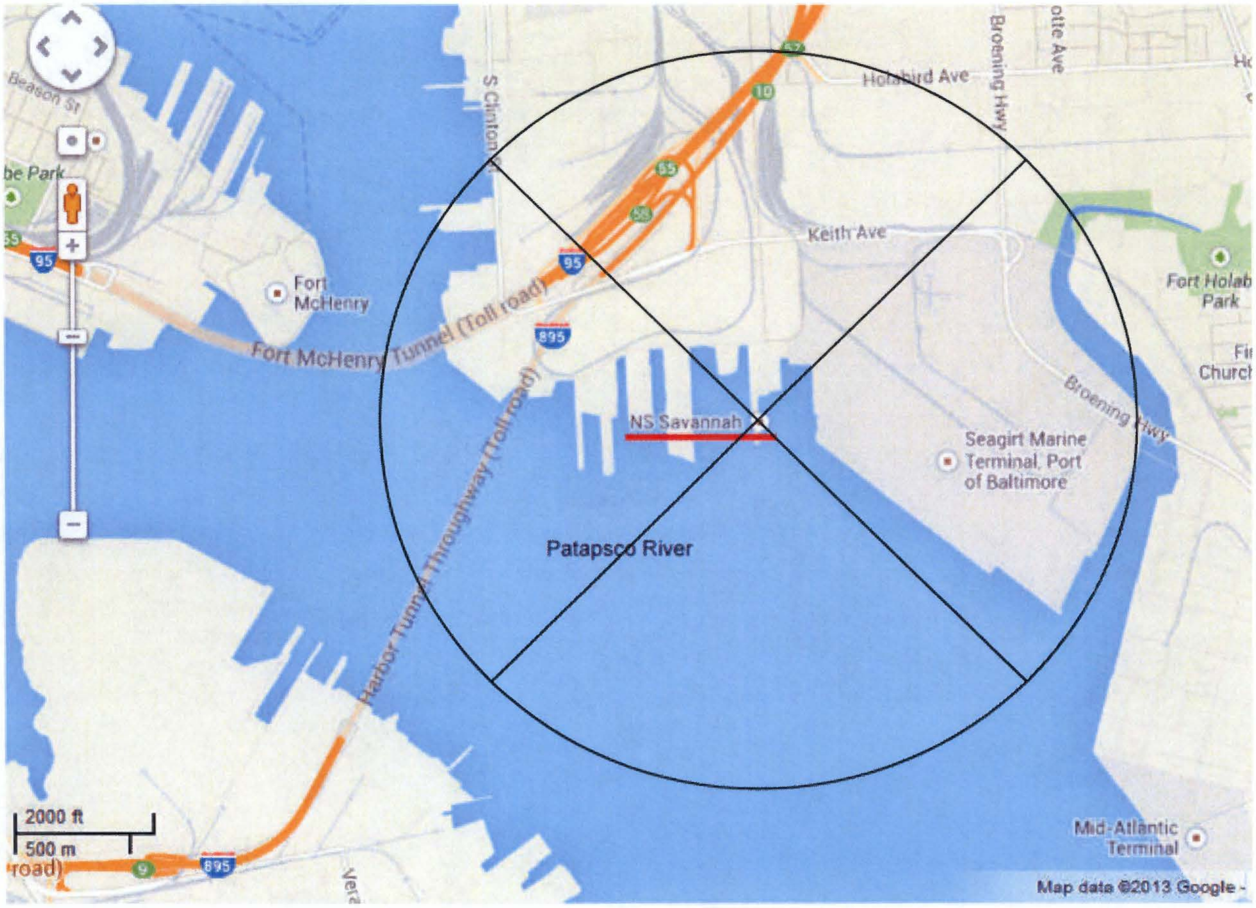

Figure 1-8 NSS Location from Google Maps, with an approximate one-mile radius overlay

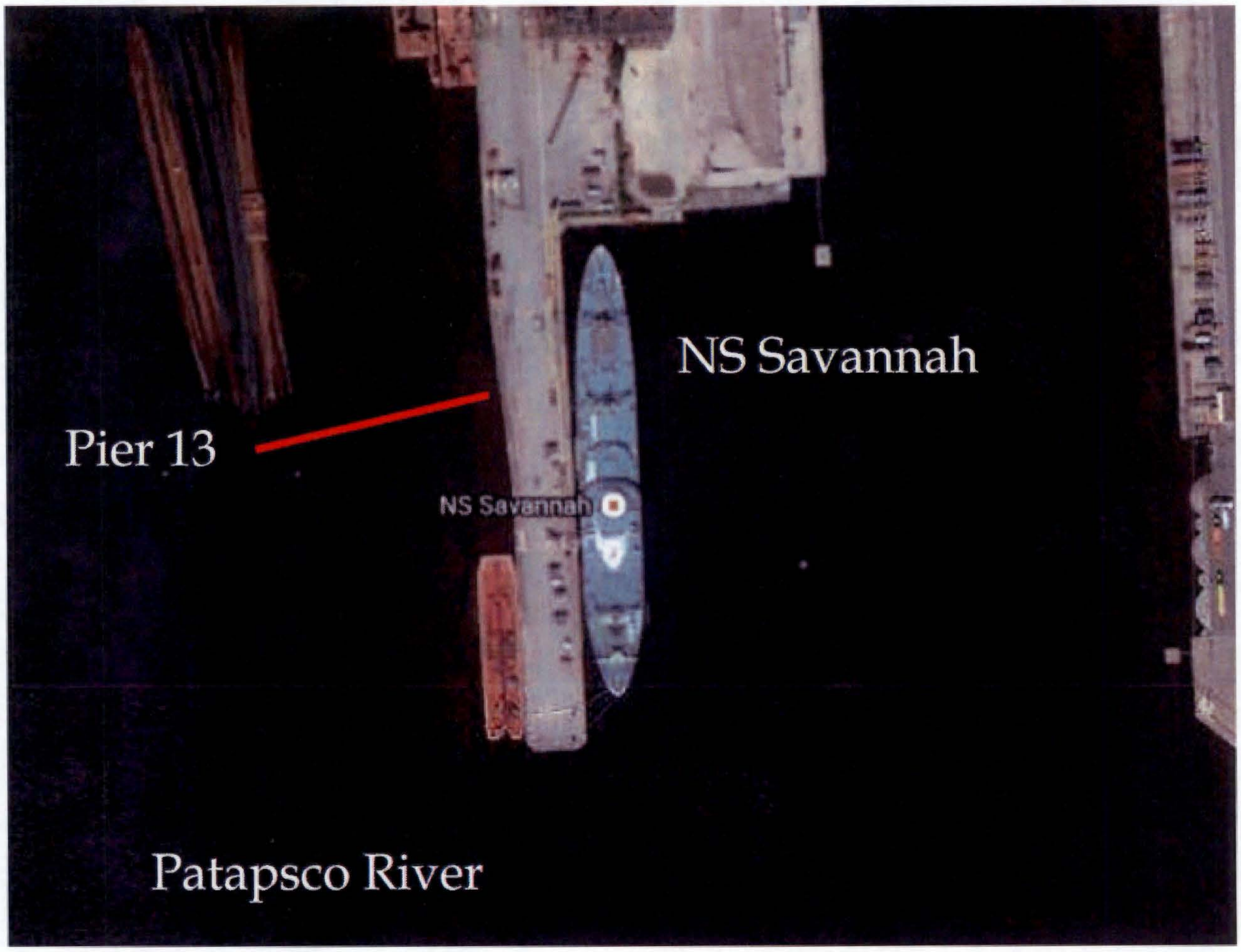

Figure 1-9 NSS at Pier 13 from Google maps satellite view, circa 2009

The west wall of the Seagirt Marine Terminal appears at the right margin of the image.

1.6.1 Port of Baltimore General Description

Baltimore, one of the major ports of the United States, is at the head of tidewater navigation on the Patapsco River. The mid-harbor point, at the intersection of Fort McHenry and Ferry Bar Channels 0.6 mile southeast of Fort McHenry, is 8 miles from the mouth of the river, 150 miles above the Virginia Capes, and 62 miles from the Delaware River.

The terrain in the Baltimore vicinity is relatively low and flat, with no wind channeling effects. There are no natural hazards in the Chesapeake Bay enroute to Baltimore. The route is void of man-made obstructions such as lift bridges, locks and traffic control points. The following information has been taken from Coast Pilot 3 - Chapter 15 - Edition 52, 2019.

1.6.1.1 Anchorages

General, dead ship and small-craft anchorages are in Baltimore Harbor. (See 33 CFR 110.1 and 110.158 for limits and regulations.)

1.6.1.2 Tides and Currents

The mean range of tide is 1.1 feet at Baltimore; daily predictions are given in the Tide Tables. Prolonged winds of constant direction may cause substantial variation in the tide. Currents in the harbor are 0.8 knot on the flood and ebb. (See the Tidal Current Tables for daily predictions.)

1.6.1.3 Weather

Baltimore is in a region about midway between the rigorous climates of the North and the mild climates of the South. It is adjacent to the modifying influences of the Chesapeake Bay and Atlantic Ocean to the east and the Appalachian Mountains to the west. January is the coldest month, and July, the warmest. Winter and spring have the highest average wind speeds.

Baltimore Harbor is frozen over during severe winters, but the icebreakers and the larger power-driven vessels keep the dredged channels open so that self-propelled vessels seldom have difficulty entering the harbor. Ice conditions in the main channel are most severe in the vicinity of Seven Foot Knoll Light, where ice moving from the northern end of Chesapeake Bay tends to collect in packs.

1.6.1.4 Repairs

Baltimore has extensive facilities for wrecking and salvage. In addition to equipment especially designed for salvage operations, heavy hoisting facilities are available.

1.6.2 Surrounding Area of the NSS Site

As noted in Section 1.4, the site itself is the NSS, with its boundary formed by the exterior surface of its hull, superstructure, and tophamper.[4] Everything exterior to the ship itself is an offsite area.[5] This includes the water in which the ship floats, the structure to which the ship is attached when at berth (i.e., Pier 13 in the current condition), and the land to which that structure is attached. There have been no changes to the site boundary since original construction.

As shown in Figure 1-1, NSS is berthed at an existing industrial facility in the Canton Industrial District of the Port of Baltimore. The current facility was constructed as a grain elevator, primarily for export service, by the Northern Central Railway, a subsidiary of the Pennsylvania Railroad (PRR), in 1922. Portions of the current facility were constructed in 1905, and similar facilities were located on the site beginning in the 1880s. The elevator ceased operations in 1994. Elevated portions of the grain discharge equipment were removed from the finger pier (Pier 13) at an indeterminate date. During the late 1990s, MARAD temporarily layberthed two general cargo ships at Pier 13. Immediately to the east of Pier 13 is the bulk unloading facility of the National Gypsum (dba Gold Bond) Company's wallboard plant on Newkirk Street, and beyond that is the Maryland Port Administration's Seagirt Marine Terminal. To the west are the remains of the PRR bulk ore pier (Pier 12; only the pier head remains in use, with remnants of the removed finger pier still attached), and the active general cargo and coal export facilities of the Canton Marine Terminal (Pier 11) and the CNX Marine Terminal. Other commercial facilities and extensive rail trackage lie to the north of Pier 13, with Baltimore Harbor to the south. In 2019 the Pier 13 complex was sold to Berg Demolition, who proceeded to demolish the remaining structures of the former grain elevator; that work was completed in the summer of 2020. In 2021, the facility was purchased by National Gypsum, who expanded the bulk gypsum storage onto portions of the former grain elevator footprint. The remaining footprint was converted into a parking area, which is leased to Hildebrandt Trucking for temporary storage of flatbed trailers serviced by dray trucks. Other tenants on Pier 13 include Moran Towing, with three (3) ship-assist tugs stationed near the head of the pier, and Project Liberty Ship which berths the historic S.S. *John W. Brown* along the west wall. Figures 1-10 through 1-15 depict NSS at Pier 13 in context with its surrounding area.

[4] Masts, spars, booms, rigging, boats, etc. In general, all the fixed items of outfitting that are not part of the ship's hull or superstructure.

[5] The only exception to this rule occurs when the ship is drydocked. Current procedures define the drydock floor and its access points to be owner-controlled areas for purposes of access and work control; however, this does not involve a literal expansion of the licensed site boundary. Similar access controls are employed when divers work on the underwater hull with the ship afloat.

The layberth facility is constructed on fill and contains no ground water. There are no off-site wells used by nearby communities; potable water for residential and commercial use is supplied by municipal systems fed by distant reservoirs in northwest Baltimore City and further beyond in Baltimore County.

1.6.2.1 Within a One Mile Radius (all Baltimore, MD 21224)

There are numerous industrial and commercial facilities within an approximately one-mile radius of the ship including (arranged alphabetically):

- A.H. Gardner & Son Trucking, 2207 Newkirk Street
- B&E Storage, 2500 Broening Highway
- BWI2 Amazon Fulfillment Center, 2010 Broening Highway
- Canton Railroad Co., 1841 South Newkirk Street
- Canton Stevedoring Inc. (Pier 11 Canton Marine Terminal), 3800 Newkirk Street
- CNX Marine Terminal Inc. (Consol Energy – export coal), 3800 Newgate Avenue
- George Hilderbrandt Trucking Inc., 4600 Newgate Avenue
- Gold Bond Building Products, Baltimore Gypsum Board Plant (and adjacent marine terminal), 2301 South Newkirk Street
- Heidelberg Materials Corp. (formerly Lehigh Cement Co.), 3100 Mertens Street
- Maryland Port Administration, Seagirt Marine Terminal, 2600 Broening Highway
- Maryland Transportation Authority, 2310 Broening Highway; also 3990 Leland Avenue
- Moran Towing Corp. (Ship Assist Tugs and Maintenance Facility), Pier 13, 4601 Newgate Avenue
- Moran Towing Corp. (Operations Center and Dispatch Office) 4616 Newgate Avenue
- MTC Logistics, 4851 Holabird Avenue
- PGT Trucking / CT Transportation, 2150 S. Newkirk Street
- Point Breeze Business Center, 2200 Broening Highway
  - U.S. State Department - US DESPATCH
  - All State Career Center
  - IDEMIA North America
- Project Liberty Ship SS *John W Brown*, Pier 13, 4601 Newgate Avenue
- Sherwood Lumber Co. (formerly Middle Atlantic Wholesale Lumber, Inc.), 2150 Newkirk Street
- Ruckert Terminals Corp, 2300 Broening Highway

Transportation

- Interstates 895 (Harbor Tunnel) and 95 (Ft McHenry Tunnel) pass within 4000 feet of the ship

- CSX and Norfolk Southern Railroads have extensive trackage within the one-mile radius. CSX maintains a ready track for locomotives inside the Seagirt Marine Terminal at the foot of New Vail Street

There are no residences within 1 mile.

1.6.2.2 Within a Two-Mile Radius

Within the two-mile radius to the east are significant residential areas of Canton and Dundalk that include:

Medical

- Johns Hopkins Community Physicians - Greater Dundalk, 2112 Dundalk Avenue, Dundalk, MD 21222
- Johns Hopkins Home Care Group, 5901 Holabird Avenue Baltimore, MD 21224

Parks

- Saint Helena Playground, corner of Rails Avenue and Parnell Avenue, Dundalk MD 21222
- Fort McHenry National Monument and Historic Shrine, 2400 East Fort Avenue, Baltimore, MD 21230
- Latrobe Park, East Fort Avenue, Baltimore, MD 21230

Schools

- Fort Holabird Park, at Oak Avenue Baltimore, MD 21224
- Dundalk Elementary School, 2717 Playfield Street, Dundalk MD 21222
- Saint Rita's School, 2907 Dunleer Road, Dundalk MD 21222
- Canton Middle School, 801 South Highland Avenue, Baltimore, MD 21224
- Archbishop Borders School, 3500 Foster Avenue, Baltimore, MD 21224
- Holabird Elementary, 1500 Imla Street, Baltimore, MD 21224
- John Ruhrah Elementary School, 701 Rappolla Street, Baltimore, MD 21224
- Graceland Park/O'Donnell Heights Elementary School, 6300 O'Donnell St., Baltimore, MD 21224

Senior Care

- Baltimore County, 101 Center Place, Dundalk MD 21222
- Park View At Dundalk, 103 Center Place, Dundalk, MD 21222
- CSI Support & Development Services 3600 O'Donnell St Baltimore MD 21224
- Future Care Canton Harbor, 1300 South Ellwood Avenue. Baltimore, Maryland 21224
- Heritage Center, 7232 German Hill Road, Baltimore, MD 21222

Just Outside of 2 Miles

Schools

- Dundalk Middle School, 7400 Dunmanway, Dundalk MD 21222
- Dundalk High School, 1901 Delvale Avenue, Baltimore, MD 21222

- Community College of Baltimore County, 7200 Sollers Point Road, Dundalk, MD 21222
- Logan Elementary School, 7601 Dunmanway, Dundalk MD 21222
- Highlandtown Elementary School, 3223 East Pratt Street, Baltimore, MD 21224
- Patterson High School, 100 Kane Street, Baltimore, MD 21224
- Francis Scott Key Elementary/Middle School, 1425 East Fort Avenue, Baltimore, MD 21230

Senior Care

- Olf Senior Housing Inc., 6424 E Pratt Street. Baltimore MD 21224
- Cove Point Apartments, 7801 Peninsula Expressway, Dundalk, MD 21222

Hospital

- Johns Hopkins Bayview Medical Center 4940 Eastern Avenue. Baltimore MD 21224

Downtown Baltimore is 4 miles north northwest of the ship.

Figure 1-10 NSS at Pier 13, September 2023, looking northward

From left to right appear the head of the Canton Marine Terminal (Pier 11), storage piles for export coal, the remnant head of Pier 12, the Pier 13 Canton Marine Terminal centered in foreground showing S.S. John W. Brown and NSS, the National Gypsum (Gold Bond) terminal, and ending with the west wall of the Seagirt Marine Terminal at far right. An expanded image appears below.

Figure 1-11 NSS at Pier 13, September 2023, looking northward

Figure 1-12 NSS at Pier 13, September 2023, looking northwest

At the top left in background is the coal loading pier of the of the CNX Marine Terminal (vessel at berth marked "TRANSBULK"), closer to NSS is the Pier 11 Canton Marine Terminal facility operated by Canton Stevedoring, Inc. Downtown Baltimore City appears in the center background.

Figure 1-12 NSS at Pier 13, September 2023, looking westward

Like Figure 1-12, this image shows the CNX Marine Terminal finger pier with a vessel loading at berth, and the Canton Stevedoring project cargo facility immediately beyond NSS. At upper left is the Fairfield section of Baltimore, while the middle harbor area appears to the left.

Figure 1-13 NSS at Pier 13, September 2023, looking northeast

The National Gypsum (Gold Bond) wall board plant appears in upper left, with the expanded gypsum rock storage area below. To the right is the west boundary of the Seagirt Marine Terminal and its container storage yard. The warehouse building in the background is the Amazon Fulfillment Center on Broening Highway.

Figure 1-14 NSS at Pier 13, September 2023, looking eastward

The berth area in left center is part of the National Gypsum facility. To the east is the Seagirt Marine Terminal operated by PortsAmerica Chesapeake for the Maryland Port Administration. The Dundalk area of Baltimore County appears beyond the terminal.

Figure 1-15 NSS at Pier 13, September 2023, looking southward

The broad expanse of the Baltimore Harbor appears astern of NSS, with the I-695 Francis Scott Key memorial bridge in the upper-left corner.

1.6.3 Operational Background

The NSS was designed, constructed and operated as a joint research and development project of the U.S. Department of Commerce, Maritime Administration and the U.S. Atomic Energy Commission (AEC). In general, MARAD's contribution was the ship while the AEC's was the nuclear fuel, reactor and related nuclear systems. The program was managed by a Joint Group established by MARAD and AEC in 1956, which also included the U.S. Coast Guard and U.S. Public Health Service. The Joint Group was disbanded in 1965 effective with the issuance of operating license NS-1; from that date forward MARAD managed the NSS alone.

Table 1-2 presents a chronology of significant licensing events. As noted in the table, the NSS was no longer operational after 1971. Since then, it has been moved by towing.

Table 1-2 Chronology of Significant Licensing Events

Date	Event Description
April 1955	President Eisenhower announced a proposal to build a nuclear-powered merchant ship to demonstrate peaceful uses of the atom.
October 1956	Contract to Babcock & Wilcox to design and construct the reactor.
January 22, 1957	Memorandum of Understanding (MOU) established between MARAD and AEC to perform tasks related to design development, testing, initial operation and international acceptance of nuclear merchant ships.
April 1957	Contract to George Sharp to design hull.
December 10, 1957	Contract to New York Ship building to construct ship.
May 22, 1958	Keel laid.
July 21, 1959	Ship launched, crew training began.
July 24, 1961	Authorization of AEC fueling and operation for test and demonstration purposes.
December 21, 1961	Initial criticality.
March 1962	Initial sea trials at Camden, NJ and Yorktown, VA.
August 1962	Commencement of initial voyage. First voyage to commercial port Yorktown, VA to Savannah, GA. Demonstration phase of operations began with voyage from Savannah, GA to Norfolk, VA.
August 1965	AEC Operating License (NS-1) issued.
August to October 1968	Commercial marine refueling at MARAD Refueling Facility, Todd Shipyards, Galveston, TX. During the last voyage before the Fuel Shuffle Outage, evidence of a minor fuel failure was detected. It appeared that small amounts of fission products were released to the primary coolant whenever there was a significant change in reactor power level. Post shuffle operation indicated that the situation still existed; however, it did not limit operation or access anywhere on the ship.
July 25, 1970	Commercial Operations ended.
November 6-8, 1970	Final Voyage to Pier "E" Todd Shipyards, Galveston, TX from annual

Date	Event Description
	drydocking 10/30 through 11/06 at Todd Shipyards, New Orleans, LA.
November 8, 1970	Final Reactor shutdown at 5:50 PM and established Cold Iron condition at Todd Shipyard, Galveston, TX per FAST-21.
December 3, 1971	Permanent cessation of operations established by completing Layup Procedure LU-9. As part of defueling, primary pump motors and impellers were removed.
December 21, 1972	Thirty-six spent fuel elements (32 plus the four replaced during the fuel shuffle) (Core I and Ia) were shipped from Galveston, TX to AEC - Savannah River Plant, Aiken, SC for reprocessing. (September 21, 1973, Operations Report).
May 19, 1976	Possession-only License issued (License Amendment 8) and recognized the ship was in a state of protective storage. Total estimated residual activity 1.09E+5 Ci.
August 14, 1981	Patriots Point Development Authority (PPDA) becomes a co-licensee (License Amendment 9) and the ship was bareboat chartered for public display at the Patriots Point Naval and Maritime Museum, Mt. Pleasant, SC from 1981 through 1994.
July 17, 1991	The National Park Service designated the ship as a National Historic Landmark.
June 29, 1994	License Amendment 12 removed PPDA as co-licensee. After drydocking, MARAD places the ship in protective storage at the James River Reserve Fleet, Ft. Eustis, VA.
December 11, 2006	Post Shutdown Decommissioning Activities Report, Revision 0 was submitted but withdrawn on January 27, 2007.
January 31, 2007	License Amendment 13 was issued. It included six administrative changes. The most significant change is following 30 day notification to NRC, the ship can be located at any appropriate domestic location with a MARAD approved Port Operating Plan.
May 8, 2008	NSS was moved to Pier 13 Canton Marine Terminal at 4601 Newgate Ave., Baltimore, MD.
October 3, 2008	MARAD submits to NRC its Environmental Assessment and Finding of No Significant Impact regarding NSS Decommissioning.
December 11, 2008	Post Shutdown Decommissioning Activities Report, Revision 1 submitted.
May 2017	Consolidated Appropriations Act for FY 2017 (Public Law 115-31) provides $24 million to fund SAFSTOR activities and start decommissioning.
March 2018	Consolidated Appropriations Act for FY 2018 (Public Law 115-141) provides $107 million. This amount is in addition to the $24 million provided by the Consolidated Appropriations Act for FY 2017. The sum equals the $131 million estimate to complete decommissioning and terminate the NSS license

Date	Event Description
April 23, 2018	Amendment 15 allows dismantlement and disposal of the facility.
June 12, 2018	Amendment 16 revises Technical Specifications to allow creating D-deck Containment Vessel (CV) door.
April 22, 2019	MARAD publishes its Supplemental Environmental Assessment and Finding of No Significant Impact regarding NSS DECON.

1.7 References

1-1 *Updated Final Safety Analysis Report*, STS-004-002, Revision 13, July 20, 2023

1-2 STS-100, *Post Shutdown Decommissioning Activities Report*, Revision 1, December 11, 2008

1-3 *Programmatic Agreement among the U.S. Department of Transportation, Maritime Administration, the U.S. Nuclear Regulatory Commission, the Advisory Council on Historic Preservation, and the Maryland State Historic Preservation Officer for the Decommissioning and Disposition of the Nuclear Ship Savannah*, Baltimore, Maryland dated March 17, 2023.

1-4 Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, Rev. 2, July 2019

1-5 NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, Rev. 2, April 2018

1-6 NUREG-1757, Volume 2, Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria, Rev. 1, September 2006

1-7 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated October 3, 2008, - *Submittal of Finding of No Significant Impact and Environmental Assessment* (ML082810182)

1-8 CR-137, *Supplemental Environmental Assessment and Finding of No Significant Impact*, April 2019

2 SITE CHARACTERIZATION

2.1 Introduction

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(A), the guidance of *Regulatory Guide 1.179, Standard Format and Content for License Termination Plans for Nuclear Power Reactors* [Reference 2-1] and the guidance in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, [Reference 2-2], this chapter provides a summary of MARAD's site-specific characterization activities performed over the period from CY 2003 through CY 2020. The chapter also includes a summary description of the Historical Site Assessment (HSA) [Reference 2-3].

The purpose of site characterization is to ensure that the Final Status Surveys (FSSs) will be conducted in all areas where contamination existed, remains, or has the potential to exist or remain. With respect to NSS, the results of the characterization surveys, including the HSA, demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected. As described in later chapters, remediation and FSS efforts will address any currently undetected residual radioactivity.

References 2-1 and 2-2 describe the content required for this chapter. Summarizing from these references, the LTP should include the following items, which are not meant to be all-inclusive:

- The LTP should describe historic events (including dates, types of occurrences, and locations inside and outside the facility), such as radiological spills, onsite disposals, or other radiological accidents or incidents, that resulted or could have resulted in the contamination of structures, equipment, letdown areas, or soils and ground water beneath buildings and in outside areas (see 2.2);
- The LTP describes, in summary form, the original shutdown (see 2.1.1) and current radiological and non-radiological status of the site (see 2.1.2);
- The LTP describes the survey instruments and supporting quality assurance (QA) practices used in the site characterization program. The LTP should discuss how the licensee applied the data quality objectives discussed in NUREG-1575 during site characterization;
- The LTP identifies the background levels used during scoping or characterization surveys;
- The LTP site characterization is sufficiently detailed to allow the Nuclear Regulatory Commission (NRC) staff to determine the extent and range of radiological contamination of structures, systems (including sewer systems and waste management systems), floor drains, ventilation ducts, piping and embedded piping, rubble, ground water and surface water, components, residues, and environment, including maximum and average contamination levels and ambient exposure rate measurements of all relevant areas (structures, equipment, and soils) of the site (including contamination on and beneath paved parking lots) (see 2.1.4 for summary information); and,
- The LTP site characterization should contain sufficiently detailed data to support planning for all remaining decommissioning activities and the final status survey program.

MARAD's site characterization efforts have been iterative, beginning roughly in 2003 and continuing to present. From 2018 – 2020, MARAD completed detailed characterization surveys throughout the ship to support decommissioning and license termination planning and the acquisition of contract services to perform the related dismantlement, waste disposal, remediation and survey work. The 2018 characterization included radiological spaces outside the Reactor Compartment (RC) and Containment Vessel (CV). The 2019 characterization was inside the RC and CV only. These surveys are documented as reports CR-104 [Reference 2-4] and CR-109 [Reference 2-5].

Note to Reviewers: As provided in Reference 2-1, a licensee can submit its entire site characterization package separately at any time before submitting the LTP and reference it in the LTP, or the licensee can submit the site characterization as an integral part of the LTP. MARAD has summarized its site characterization efforts in its *Updated Final Safety Analysis Report* (UFSAR) [Reference 2-6] beginning with Revision 11. This chapter of the LTP contains an expanded discussion of the NSS site characterization activities. Reviewers should be aware that these expanded discussions are drawn from contemporary reports, often verbatim, with some dating back nearly twenty years. With minor exceptions, these discussions have not been edited to account for the passage of time. Furthermore, there are instances where parent reference documents, such as NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* [Reference 2-7] are quoted within the body of the text. These reference documents typically do not include language applicable to a ship, e.g., a floor should be a deck; a wall should be a bulkhead; and a ceiling should be an overhead in most cases. For some survey areas and units, these words may be used in reference to complex internal ship structure. The LTP may include footnotes or other explanatory material where the parent reference appears confusing. Such explanations may also be found in subsequent chapters.

2.1.1 Operations and Final Shutdown Summary

The commercial operation of the NSS ended in July 1970. In November 1970, NSS made a final round trip voyage from Galveston, TX to New Orleans, LA for drydocking. Upon the ship's return to Galveston, the reactor was shutdown, for what later proved to be the final time. This was a normal controlled shutdown in accordance with operating procedures and was not the result of any accident condition.

Over the course of 1971, the reactor was defueled and prepared for long-term layup. At the time MARAD expected that NSS would, at some later date, be refueled and returned to service. As described in References [2-3] and [2-6], the spent fuel was transferred to the Atomic Energy Commission (AEC) in 1972 for reprocessing. By early 1973, it was evident that the expected return to service would not occur. MARAD then discontinued the maintenance activities that would have permitted the reactor to be refueled. Decades later, when the decommissioning rulemaking required a declaration of permanent cessation of operations, MARAD determined that the completion of defueling activities on December 3, 1971, would serve that purpose.

2.1.2 Current Site Radiological and Non-Radiological Conditions

After completing the 2018 – 2020 characterization surveys, the following radiologically controlled areas (RCAs) were remediated and are no longer RCAs:

- Port Stabilizer Room
- Starboard Stabilizer Room
- Port Charging Pump Room
- Starboard Charging Pump Room
- Forward Control which includes on C-Deck the Cold Water Chemistry Lab (port) and Radiation Monitoring Room (starboard) plus on D-deck the Gas Adsorption Equipment Room (port) and Radiation Sampling Room (starboard).
- Health Physics Laboratory
- Hot Chemistry Laboratory
- D Deck Nuclear Electronics Workshop and Storeroom (NEWS)

- Bulk Stores (including Special Stores #1)

2.1.3 Chronology of Decommissioning Planning and Characterization

In its December 2008 revised *Post Shutdown Decommissioning Activities Report* (PSDAR) [Reference 2-8], MARAD provides a summary description of the planning activities associated with decommissioning and license termination. Other documents on the NS-1 docket provide additional context and detail such as the MARAD annual reports for CY 2002 – 2007 and the records of public meetings between MARAD and NRC in 2003 and 2004. To an extent, MARAD benefitted from detailed investigations and surveys performed in 1998-1999 by the Army Corps of Engineers to support decommissioning planning for their nuclear power barge STURGIS, a plant of similar characteristics and history. At the outset, MARAD believed that it would be necessary to dispose of the NSS Reactor Pressure Vessel (RPV) at the commercial Low Level Radioactive Waste (LLRW) repository at Barnwell, SC, on 1) the basis that the RPV was not expected to meet the waste acceptance criteria for the commercial LLRW repository at Clive, UT,[6] and 2) MARAD was uncertain if the Department of Energy (DOE) would accept the NSS waste at one of its facilities. Because Barnwell would not accept out-of-compact waste after 2008, and because it was unlikely that the RPV could be removed before that date, it appeared that decommissioning would be geographically limited to Mid-Atlantic Compact states.[7] This was perceived to be too limiting from a competitive acquisition perspective, resulting in a need for more detailed and discrete information. The progression of characterization efforts, particularly regarding the disposition of the RPV, is more apparent within this context. Eventually, the competitive pressures relaxed when it was shown that the RPV could meet the Clive waste acceptance criteria. Those pressures were essentially eliminated in 2009 when the additional commercial LLRW repository was opened in Andrews, TX.[8] In 2019, MARAD also received a determination from the DOE that NSS waste was of AEC origin and eligible for disposal at certain of their sites; however, in consultation with DOE, MARAD determined to pursue a commercial alternative for waste disposal.

In 2003, MARAD awarded a decommissioning planning contract to WPI, Inc., operating from offices in Richmond, VA. WPI's initial efforts are described in Section 2.4.1 of the PSDAR [Reference 2-8] and included the 2004 RPV activation analysis described in Section 2.3.1 of this chapter. This activation analysis is documented as report CR-142 [Reference 2-9] and was the technical basis for the conclusion that the RPV would not meet the Clive waste acceptance criteria.

MARAD awarded a follow-on contract to WPI in late 2004 to perform the characterization scoping surveys described in Section 2.3.2 of this chapter. This effort was performed in the first half of CY 2005. It extended throughout the ship and is documented as report CR-038 [Reference 2-10]. The effort included radiological and environmental sampling and involved the first entries into the CV since 1975. Although the overall effort met the pre-survey expectations, one significant finding did not support the 2004 RPV activation analysis. This led directly to the destructive sampling of the RPV and surrounding Neutron Shield Tank described in Section 2.3.3 of this chapter. The activities were performed in the latter half of CY 2005 and are documented as report CR-056 [Reference 2-11] dated January 2006.

The above activities, together with a draft Environmental Assessment, supported MARAD decommissioning planning up to the submittal of PSDAR Rev. 0 in December 2006. As described in the

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[6] Based largely on commercial decommissioning experience up to that time, and later supported by the 2004 activation analysis described in this chapter.

[7] The Mid-Atlantic Compact, consisting of the states of South Carolina, New Jersey, and Connecticut. For NSS waste to be disposed at Barnwell after July 2008, decommissioning would have to take place in one of those three states. Connecticut was not considered a viable option for a variety of reasons, and New Jersey would not permit NSS waste to be included in its allowed site volume. This apparently restricted decommissioning to South Carolina, which had only one known suitable site available (i.e., Barnwell).

[8] With two LLRW sites available, and no restriction on the location where decommissioning activities could be performed, there was adequate competition for services under the Federal Acquisition Regulations.

PSDAR, Rev. 1 [Reference 2-8], that document was withdrawn soon after submittal. The decommissioning project itself was rescoped, with a near-term focus on returning NSS to protective storage, albeit with significant administrative and technical upgrades to meet contemporary SAFSTOR criteria. Although the full scope of the SAFSTOR effort was ultimately not funded, MARAD was able to complete the research and investigations necessary to prepare and complete the HSA. The HSA was revised in 2023. MARAD completed an Environmental Assessment (EA) and Finding of No Significant Impact (FONSI) in 2008 and submitted these to NRC for its information and use on October 3, 2008 [Reference 2-12]. Following the submittal of the PSDAR, Rev.1 [Reference 2-8], planning to support the SAFSTOR technical upgrades continued into 2009; however, as mentioned earlier, funding to implement these upgrades was not appropriated, and the upgrades were not carried out. The administrative efforts were completed. The uncompleted SAFSTOR technical program included a MARSSIM-based detailed characterization campaign.

Decommissioning funds were appropriated in two tranches, one in FY 2017 for $24M, and the other in FY 2018 for $107M. With resources available, MARAD tasked its integrated support contractor to implement the characterization plan. This campaign was divided into two separate efforts; the first was performed in August-September 2018 and involved areas of the ship outside of the RC and CV. This effort is described in Section 2.3.4 of this chapter and is documented in report CR-104 [Reference 2-4]. The second effort took place after asbestos insulation was removed from components in the CV. This effort was conducted in 2019-2020, is described in Section 2.3.5 of this chapter, and is documented in report CR-109 [Reference 2-5].

Two other characterization efforts are described in this chapter. In 2019, a MARSSIM-based survey was conducted on the fully exposed exterior surfaces of the hull, while the ship was on drydock in Philadelphia, PA. This effort is described in Section 2.3.6 of this chapter and is documented in report CR-143 [Reference 2-13]. Finally, in 2021, the exterior lead shielding on the Neutron Shield Tank (also known as the Primary Shield Tank) was sampled and analyzed to support the release of the material. This effort is described in Section 2.3.7 of this chapter and is documented in report CR-144 [Reference 2-14].

In addition to the above-described characterization efforts, the site's operational Radiation Protection Program provided valuable historical data and continues to provide input regarding site radiological conditions. Measurements and samples beyond the scope of the operational survey program have been conducted in areas recognized as needing additional information to assess the type, magnitude, and extent of contamination. These measurements and samples have been considered in decommissioning planning.

2.1.4 Other Considerations Regarding Site Characteristics and Characterization

The NSS site possesses a number of unique or unusual features and characteristics that distinguish it from typical land-based facilities. These are described in Section 5.2 of Chapter 5 of the LTP. Some of these features and characteristics have direct bearing on the decommissioning planning and the past characterization efforts described in this chapter. Among these are the following:

- The site (i.e., the ship) contains no soil;
- The site contains no surface or ground water;
- The site contains no embedded pipe, no rubble, no buried or surface paved parking lot;
- Topside deck drains gravity drained overboard;
- Welded hull blanks prevented any auxiliary or secondary system from discharging overboard after operations ceased, and,
- The NSS "sewer systems" operated as follows (see also Chapter 9 of the UFSAR):
    - Both sampling sinks and the decontamination shower drain in the Cold Water Chemistry Lab (C deck Port) gravity drain to Contaminated Water Tank Starboard TD-6;

- The decontamination shower at frame 125 gravity drained overboard. The hull opening was welded closed in 1976;
- There are no deck drains other than the "shower drain" in the cold chemistry laboratory;
- Sinks and showers gravity drained overboard. The hull opening was welded closed in 1976. These are not contaminated systems; and,
- Toilets drained to the sewage tank in the engine room where it was pumped overboard. The hull opening was welded closed in 1976. This is not a contaminated system.

The overall objectives of the NSS characterization program were:

- To identify potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information;
- To identify spaces in the ship that need further remediation as opposed to those posing no threat to human health;
- To provide an assessment of the likelihood of contaminant migration;
- To provide information useful for scoping and characterization surveys; and,
- To establish appropriate survey units and provide initial classification of these survey units as impacted or non-impacted by radioactivity.

Note that there were no significant radiological events or activities reported in the HSA or during the museum period where the "passenger and stateroom" shower, sink or sewer systems could have become contaminated.

Characterization efforts for the NSS decommissioning project were an iterative process spanning all aspects of the remediation activities. The information developed during the characterization program represents a radiological and hazardous material assessment based on the knowledge and data available at the end of 2019. This information was sufficient to satisfy the objectives listed above. Additional measurements and samples may be obtained during the remediation process to continue to ensure adequacy of area classifications and effectiveness of the FSSs to show compliance with the established Derived Concentration Guideline Levels (DCGLs), in accordance with the guidelines of MARSSIM.

The site characterization incorporates the results of investigations and surveys conducted to quantify the extent and nature of contamination at the NSS. In addition, the results of site characterization surveys and analyses have been and continue to be used to identify areas of the site that will require remediation, as well as to plan remediation methodologies, develop waste classification and volumes, and estimate disposal costs.

The information obtained from the characterization provides guidance for decontamination and remediation planning. Materials which were shown to be contaminated - such as sludges, liquids, residues and system components - at concentrations greater than the unrestricted release criteria have been and will continue to be removed and properly packaged for shipment and disposal. Extensive characterization and monitoring have been performed. Measurements and samples were taken in each accessible area, along with the historical information, to provide a clear picture of the residual radioactive materials and its vertical and lateral extent at the ship.

2.2 *Historical Site Assessment*

2.2.1 Introduction

The HSA for the NSS was completed in 2008 by AREVA Federal Services. The report documenting the HSA is CR-003 [Reference 2-3]. The process for conducting the HSA was established in accordance with MARSSIM guidelines. Assessment activities included reviews of records, inspections of the ship, evaluation of existing characterization-type data, and personal interviews with former crewmembers. Many of these interviews were conducted during a reunion of the former crew held in May 2008.

The HSA focused on historical events and routine operational processes that resulted in contamination of the plant systems and rooms within the Radiologically Controlled Area (RCA). The HSA, as part of the initial characterization program, was conducted to support the objectives detailed in Section 2.1.

2.2.2 Methodology

The HSA was designed to evaluate input from two separate sources - plant records and personnel interviews. The review of plant records included review of the following:

- Routine radioactive effluent release reports;
- Non-routine reports submitted to the NRC under provisions of the technical specifications, 10 CFR 20, or 10 CFR 50;
- Plant incident reports;
- Corrective action reports; and,
- Findings documented in accordance with other assessment processes such as the Decommissioning Quality Assurance Plan (DQAP) and oversight activities.[9]

The information obtained through this process forms the input data for the records that are maintained on site to satisfy the requirements of 10 CFR 50.75(g)(1). The objective of the document reviews was to identify events that caused the contamination of systems, buildings (i.e., ship compartments and structures), external surfaces, subsurface areas, or waterways, via atmospheric releases, liquid releases, or release of solid radioactive material. For each event, available supporting documentation regarding event description, facility and system design, radiological surveys and analysis, remediation efforts, and post remediation surveys was collected and reviewed.

Assessment activities included:

- Reviews of documents related to the ship;
- Inspections of various areas of the ship;
- Evaluation of existing data on radioactive, hazardous, and toxic contaminants; and,
- Personal interviews with former crew members.

In addition to the review of plant records, telephone interviews with individuals involved in nuclear operations at the NSS were conducted. Personnel interviewed included selected present and former employees and contractors involved in operations, maintenance, and radiation protection activities at the ship. Additional interviews were conducted during the crew reunion that took place during the week of May 19, 2008. Many former crewmembers participated and several of them were interviewed, covering the same topics as discussed in the telephone interviews.

2.2.3 Results

Information reviewed during the HSA identified several events that involved atmospheric releases, unplanned liquid releases, facility contamination and release of radioactive material. The following list

[9] An NRC approved quality assurance plan was not required until 1983 - many years after the NSS received a Possession-only License in 1976.

describes the major events in each of these categories and the site areas impacted by the events. Representative documented spills of radioactivity include:

- A 1962 incident at the sampling station where a Tygon tube failed and sprayed the operator with contaminated water with activity of 3.5E-04 μCi/mL;
- A case in 1962 where a desurger ruptured in the charge pump room, spraying an engineer with contaminated water with activity of 2.3E-06 μCi/mL;
- A case in 1962 where air samples in passenger areas showed 5.5E-10 $\mu Ci/cm^3$, above background levels which were typically in the range of 1.0E-11 to 1.0E-12 $\mu Ci/cm^3$;
- A 1966 primary coolant leak in the containment vessel that necessitated extensive decontamination;
- A 1966 radioactive spill of a few drops of contaminated water on the linoleum deck in Hot Chemistry Laboratory;
- A 1968 primary system leak from a pressure relief system isolation valve;
- A 1969 primary system leak from a control rod drive buffer seal;
- Another 1969 primary system leak in vent piping from a primary coolant pump that released 3500 gallons inside the containment vessel;
- A case in 1969 where contamination was found in the engine room on the steps leading to one of the buffer seal charge pump rooms during maintenance on one of the charge pumps; and,
- An instance in 1971 where low-level contamination of 20-60 $pCi/100cm^2$ was detected on the Promenade Deck in an area where casks had been transferred.

2.3 Characterization Activities

2.3.1 Activation Analysis in 2004

An activation analysis program was completed in 2004 and documented in CR-142 [Reference 2-9].

The objective of that assessment was the analytical determination of nuclide activation levels in the NSS RPV, reactor internals and neutron shield tank (NST) for the purpose of waste classification. That effort involved a dual approach – manual calculations using a simplified reactor model for isotopic irradiation/decay, and a detailed irradiation analysis using the ORIGEN-ARP 2.00 computer code. Oak Ridge National Laboratory developed the code for the NRC and DOE to satisfy a need for an "easy-to-use" standardized method of isotope activation and depletion/decay analysis for radioactive material.

The principal isotopes contributing to activity levels include Fe-55, Co-60, Ni-63, Nb-94, C-14, and Ni-59. Other activation products present at shutdown (1970) either have half-lives less than a year and have now decayed to insignificant Curie levels or are present in insignificant quantities resulting in negligible contributions to the current Curie inventory. As shown in Table 4 of CR-142, insignificant quantities of Nb-94 and C-14 are less than 0.001 Ci and less than 0.1 Ci for Fe-55, Co-60, Ni-59, and Ni-63. The decay time used for this evaluation was 37 years, which assumed that the RPV, reactor internals, and neutron shield tank would be packaged and shipped in November 2007. This assumption provided additional conservatism for assessing the packaging and shipment of the RPV and reactor internals when they were removed in 2022.

Activity levels were determined analytically and then averaged over the waste volume in the RPV, reactor internals and NST to obtain Curie concentrations for each nuclide in accordance with calculation techniques acceptable to Barnwell and in accordance with 10 CFR Part 61. The averaged Curie concentration levels of each nuclide were compared with Barnwell's waste acceptance criteria and presented in Table 2-1.

Table 2-1 Barnwell Waste Acceptance Criteria Classification of the RPV and Reactor Internals

Radio-nuclide	NSS Total Curies (November 2007) in RPV and internals (Curies)	NSS Concentration in RPV and internals (Curies/m^3)	Barnwell Waste Acceptance Criteria		NSS Concentration Relative to the Waste Acceptance Classification Limit
			Concentration (Curies/m^3)	Classification	
Fe-55	17.8	1.39	< 700	A	0.002
Co-60	1108	86.74	< 700	A	0.124
Ni-59	30.6	2.40	< 22	A	0.109
C-14 in metal	7.32	0.57	< 8	A	0.072
Nb-94	0.100	0.0078	< 0.02	A	0.391
Sum of the Fractions – Class A Waste					**0.698**
Ni-63 in metal	2902	227.2	$\leq$ 700	B	0.324

2.3.2 WPI Characterization in 2005

This program involved extensive radiological surveys and sampling as described in the WPI detailed report, CR-038, *Radiological and Non-Radiological Spaces Characterization Survey Report* [Reference 2-10]. The NSS Characterization Project was intended to provide MARAD with a profile of radiological and non-radiological contaminants on the ship in radiological spaces. The scope of work was to perform a radiological and environmental hazard characterization program of the radiological spaces to document the location and extent of radiological and environmentally hazardous materials within these spaces preceding the decommissioning effort. In addition, a number of smears and samples were taken in non-radiological spaces to facilitate future analyses. MARAD used the information obtained from this project to develop appropriate decommissioning strategies and to estimate associated costs.

Only those locations and equipment/structures that were expected to be radioactive were surveyed in depth to determine the extent and types of radioactive materials present. The remaining areas (principally aft of the engine room, forward of the reactor compartment, and in the mid-ship-house and public areas) were surveyed less rigorously than radiological areas but in sufficient detail to confirm that no radioactive materials reside in those locations. The characterization program was conducted from March 20 to April 25, 2005. As this characterization activity aged, MARAD determined that it effectively devolved into a scoping characterization that, at best, would provide insight for developing a new characterization performed much closer to actual decommissioning.

Included were 1,423 smears, 26 paint samples, 14 metal samples, six core bores in the concrete secondary containment, 10 crud samples from the primary system, four water samples from the primary side of the steam generators, one water sample from the essentially empty Neutron Shield Tank (NST), and 11 air samples.

In-situ gamma spectroscopy was also performed in 16 locations using the Berkeley Nucleonics SAM 935® portable surveillance and measurement system, which uses a 3 inch by 3 inch NaI (Tl) detector to provide isotopic identification.

Each portable instrument was checked daily for proper background. This background value was established when the instrument was first put into service on this project. A source count value using an appropriate check source was established initially for the portable instruments. From this initial count, a ±20% range was established for each instrument. On a daily basis or more frequently if appropriate, the appropriate check source was counted with each portable instrument. The daily source count was entered on the Instrument Source Check Log for each instrument and verified to be within this ±20% range. Table 2-2 presents the portable instrumentation used for the project.

Table 2-2 2005 Characterization Project Instrument List

Instrument model	Serial number	Probe model	Probe serial number	Radiation detected	Readout units
Ludlum 3	97416	44-9 pancake	NA	Beta/gamma	cpm
Ludlum 12	75809	44-9 pancake	NA	Bata/gamma	cpm
Ludlum 12	91037	44-9 pancake	NA	Beta/gamma	cpm
Ludlum 19	42972	Internal scintillator	NA	Gamma	μR/hour
Ludlum 19	95499	Internal scintillator	NA	Gamma	μR/hour
Ludlum 19	95469	Internal scintillator	NA	Gamma	μR/hour
Ludlum 2221	197766	43-5 scintillator	127385	Alpha	cpm
Ludlum 2221	94954	44-9 pancake	NA	Beta/gamma	cpm
Ludlum 2929	102001	43-10-1	103276	Beta/gamma	cpm
Ludlum 2929	160019	43-10-1	167229	Beta/gamma	cpm
Teletector 6112D	28991	NA	NA	Gamma	mR/hour
Radeco H-810DC	0864	Air sampler	NA	Air particulate	NA
Radeco H-810DC	0865	Air sampler	NA	Air particulate	NA
Canberra high resolution gamma spectrometer*	S/N 96-5740	Base unit/detector	SAM 935 (90163/SSR593)	Gamma	Kev/Mev

The non-radiological areas were clear of detectable radiological contamination. The non-radiological areas evaluated are summarized in Table 2-3.

Table 2-3 2005 Non-Radiological Area Summary

Deck/compartments	Number of areas evaluated	Dose rate found	Contamination found
Navigation Bridge Deck	8	Background	All < background
Boat Deck	10	Background	All < background
Promenade Deck	2	Background	All < background
A Deck	20	Background	All < background
B Deck	44	Background	All < background
C Deck	31	Background	All < background
D Deck	11	Background	All < background
Weather Deck (A Deck) fwd and aft	15	Background	All < background
14 foot – 0 inch Flat	5	Background	All < background
Hold Deck (i.e., Horseshoe Tank Top)	6	38 μR/hour*	All < background
Cargo Hold Number 4 (aft)	5	250 μR/hour**	All < background
Machinery Casing, boat to C Deck	4	Background	All < background
Machinery Space and Control Center	8	Background	All < background
Hold Number 5, M.S. & D.A. Equip Rm and Workshop Rm	2	Background	All < background

*Hold Deck (i.e., Horseshoe Tank Top) had a pipe running under the deck plate in the passageway that read 38 μR/hour on contact. This pipe ran through a portion of the crossover area also. The pipe is for the waste transfer system. Lower dose rates were recorded at various areas of the passageway above the deck plates. (Note that piping in the Tank Top area was removed in 2019. The piping removed included suction and discharge piping for the Buffer Seal Charging Pumps as well as piping of the Equipment Drain and Waste Collection system.)

**Hold Number 4, "D" Deck Starboard, had readings on the aft bulkhead up to 250 μR/hour. This appears to be shine from the Cold Water Chemistry Lab (Radiation Monitoring Room).

Table 2-4 provides a summary of radiological conditions found during the evaluation of radiological areas excluding containment. Note that both Hold Deck (i.e., Horseshoe Tank Top) and Cargo Hold Number 4 (aft) were subsequently designated as Radiological Areas. The values listed are maximums.

Table 2-4 2005 Radiological Area Summary

Deck/Compartments	Dose Rate Found	Contamination Found (dpm/100cm^2)
Hot Chemistry Lab, D Deck port side of the Control Center room	Background	< 1000
Port Forward Stabilizer Room, upper level, 14 foot – 0 inch Flat	8 μR/hour	All < background
Port Forward Stabilizer Room, lower level	150 μR/hour	All < background
Fan Room B Deck, starboard side	Background	All < background
Stateroom B-1 (radiological waste storage)	Background	All < background
Cold Water Chemistry Lab, C Deck	50 μR/hour	All < background
Gas Adsorption Equip Room (enter through Cold Water Chemistry Lab, C Deck), lower level	2000 μR/hour	Max 3904
Hold #4, D Deck, starboard	250 μR/hour	All < background
Charge Pump Rooms, port and starboard	180 μR/hour	All < background
Health Physics Lab, A Deck	5 μR/hour	Max 1221
Hold Deck, aft of containment, port-to-starboard crossover passage (i.e., Horseshoe Tank Top)	25 μR/hour	All < background
Hold Deck, outside containment, port and starboard passages (i.e., Horseshoe Tank Top)	38 μR/hour	All < background

Table 2-5 provides a summary of radiological conditions found during the evaluation of Reactor Compartment areas outside of the Containment Vessel. The values listed are maximums.

Table 2-5 2005 Reactor Compartment Radiological Summary

Deck/Compartments	Dose Rate Found	Contamination Found (dpm/100cm^2)
B Deck, access area aft of reactor	Background	All < background
B Deck, area forward of reactor	Background	All < background
C Deck forward, access from B deck	Background	All < background
A Deck around Cupola	Background	All < background
Top of Cupola	4 μR/hour	All < background
Aft Mezzanine, mid-level between C Deck and D Deck	3–5 μR/hour	All < background
Lower level of Secondary Containment	221 mR/hour	All < background

Reactor Compartment Lower Level, access down ladder tube only.

General area starboard side 300–1000 μR/hour, 1400–1600 μR/hour head high.

Overhead yellow line emits 221 mR/hour on contact. These lines are posted:

- Starboard forward general area is 400–500 μR/hour; and,
- Port side general area was 60–80 μR/hour.

No radiological contamination was found.

Table 2-6 provides a summary of radiological conditions found during the evaluation of CV. The values listed are maximums.

Table 2-6 2005 CV Radiological Summary

Deck/Compartments	Dose Rate Found	Contamination Found ($dpm/100cm^2$)
CV, upper hatch closed	15 μR/hour	All < background
CV, upper hatch open	400 μR/hour	All < background
CV, 1^{st} level	500 μR/hour	1200
CV, inside shield tank upper ring	7 mR/hour	All < background
CV, 2^{nd} level	3 mR/hour	All < background
CV, 3^{rd} level	10 mR/hour	< 1000
CV, 3^{rd} level, area over U-tube steam generator	35 mR/hour	< 1000
CV, 4^{th} level	3 mR/hour	All < background

2.3.3 RPV, Reactor Internals and NST Sampling in 2005

In 2005, a project was conducted on the NSS to determine the Curie content and isotopic inventory of the RPV, Reactor Internals, and NST by extracting metal samples at selected locations in the RPV and internals, and subsequent radiochemical analysis. The objective of this project was to refine the 2004 activation analysis performed by WPI and obtain a more accurate set of nuclide activation measurements for waste classification of the RPV for disposal purposes. These measurements are based on current RPV and internals conditions as observed from actual metal sampling in the reactor internals. The results of the activity are documented in WPI report number CR-056 [Reference 2-11].

Using a heavy metal boring system, a 4 inch access hole was drilled through the external lead shield and the outer and inner annuli of the NST, and the thermal insulation layer adjacent to the RPV shell. The 4 inch bore was sleeved with PVC pipe. A 1.0625 inch hole was bored in the center through the carbon steel RPV shell, the 0.5 inch stainless steel cladding on the RPV shell inner surface, and through the outer thermal shield. A 0.5 inch hole was drilled through the middle thermal shield.

All metal samples were taken in the form of chips by extending a drill bit with an extension shaft, operating inside of a sleeve, through the metal to be sampled. The sleeve forced the chips up the drill bit flute from which samples were obtained. A new drill bit and sleeve was used for each sample to eliminate cross contamination. Each sample was packaged separately and marked to preserve a chain of custody.

A total of eight metal samples, one insulation sample, and two liquid samples from the secondary steam generator loops were collected, bagged, packaged for transportation and shipped to GEL Laboratories, LLC (GEL), a QA certified laboratory in Charleston, SC. A 10 CFR Part 61 analysis of seven metal samples was performed. The lead shield and thermal insulation were analyzed by gamma scan only.

Table 2-7 below presents the results of total RPV nuclide activation levels based on actual radiochemistry data from the samples and analysis using the ORIGEN code for the Part 61 analyses, respectively. The concentration of each radionuclide was averaged over the entire volume of metal in the RPV and internals. As shown, all nuclides are within the Waste Classification Class A limit both individually per isotope and when combined using the sum of the fractions for Class A Waste, which is 0.89. These results satisfy the WAC criteria and averaging methodology for burial at Chem-Nuclear Systems (now Energy*Solutions* - Barnwell, SC) and Envirocare (now Energy*Solutions*) of Utah (Clive, UT).

Table 2-7 Summary of the RPV and Internals Activities

Nuclide	Metal Sample Analysis		ORIGEN		WAC Class A Limit / Ratio*
	Curies	Curies/m^3	Curies	Curies/m^3	Curies/m^3
Ni-59***	4.1	0.3	3.9	0.3	22 / 0.014
Nb-94***	< MDA**	--	<0.0001	--	--
C-14***	<0.01	--	<0.0001	--	--
Ni-63***	385	30.1	356	27.9	35 / 0.86
Co-60	62	4.9	80	6.3	700 / 0.007
Fe-55	1.1	0.09	0.9	0.07	700 / 1.3E-4

*Ratio of Curie concentration from metal sample analysis to Class A limit

** Minimum detectable Activity Level

*** in activated metal

2.3.4 Characterization Activities in 2018

Radiation Safety and Control Services Inc. (RSCS) performed an NSS radiological characterization in specific areas outside of the reactor compartment and containment vessel in August and September 2018. . The results of the characterization effort are documented in CR-104 [Reference 2-4]. The intention of this work was not to repeat previous characterization efforts, but to fill in the gaps and expand the characterization data in certain areas. The areas surveyed are presented in Table 2-8.

Table 2-8 2018 Areas Surveyed

Survey Unit	Description	Location (Deck)
103A1	Port Stabilizer Room	Hold
104A1	Port Charging Pump Room	Hold
105A1	Starboard Charging Pump Room	Hold
106A1	Cold Water Chemistry Lab/ Radiation Monitoring Room	C
107A1	Gas Absorption Equipment Room/ Radiation Sampling Room	D
108A1	Health Physics Lab	A
109A1	Hot Chemistry Lab	D
201A1	State Room B1	B
202A1	Horseshoe area	Hold
203A1	Hold Deck (below deck plates)	Hold
308A1	NEWS Room bulk inventory	System
111A1	CRDM Pump Room	B
301A1	Navigation Bridge Deck: exterior surfaces and deck	Navigation
302A1	Boat Deck – exterior surfaces and deck	Boat
303A1	Starboard Stabilizer Room	Hold
304A1	B Deck Fan Room	B
305A1	Engine Room Machinery Space	D
306A1	Cargo Hold No.4 C Deck (Starboard)	C
307A1	Cargo Hold No.4 D Deck (Starboard)	D
309A1	Cold Chemistry Lab	D
401B1	Background (A3 + Barber & Beauty Shop)	A
105D1	Buffer Seal (System SL) Booster Pumps & Charge Pumps	System
113D1	Radioactive Waste and Dilution System (WD)	System
119D1	Radiation Monitoring System (RM)	System
119D2	Installed Check Source locations	System

The primary focus of radiological characterization was a MARSSIM based approach using scans, static measurements, smears and dose rate measurements in twenty identified survey units. In addition, several systems/items were also included in this characterization program. CAD drawings of the survey units were created specifically for documentation – including floors, walls, and ceilings. Seventeen static/smear locations were designated in each survey unit using Visual Sample Plan (VSP) software. A total of ten tritium samples were part of this characterization and survey units with the highest potential for tritium were selected for these samples. The MARSSIM survey class included specific instructions for collection of measurements and data logging, figures with locations, and documentation for recording survey activities/results. These surveys were designed to collect the specified radiological samples and analysis for these areas (scan, static, smear, dose rate, sample).

For the survey design, the Co-60, H-3, and Am-241 screening values, taken from NUREG/CR-5512, Vol. 3, Table 5.19 [Reference 2-15], were used as investigation levels. The investigation levels for beta and gamma scans were 500 cpm and 5,000 cpm respectively.

A characterization survey package was created for each survey unit, which included specific instructions for collection of measurements and data logging, figures with locations and documentation for recording survey activities/results.

Radiological surveys were performed with the following instrumentation:

- Thermo RadEye SX with Ludlum Model 43-89 detector (alpha/beta directs and beta scans);
- Thermo RadEye SX with Ludlum Model 44-10 (gamma scans);
- Ludlum Model 3 with Ludlum Model 44-9 (beta scans);
- Bicron MicroRem (dose rate); and,
- Ludlum Model 19 (dose rate).

Instruments were properly calibrated, and beta efficiencies were determined with Tc-99, alpha efficiencies with Th-230 and gamma calibrations with Cs-137. Operational checks were performed each day prior to use. With a 3-minute count time and 1-minute background, direct measurement minimum detectable concentrations were in the range of 10 to 39 dpm/100cm^2 alpha and 98 to 338 dpm/100cm^2 beta. Measurements were collected with the detector approximately 0.5 inches from the surface.

In Situ Object Counting System (ISOCS) measurements were performed with a 2x2 inch stabilized and ISOCS characterized sodium iodide detector with a lead collimator that was portable for access to the various locations within the ship. The ISOCS sensitivity varied by count time, the object being counted, and the distance from the object.

Smears were counted at the RSCS laboratory on a Tennelec gas proportional counting system. The minimum detectable activity for alpha smear measurements was approximately 16 dpm/100cm^2 and approximately 43 dpm/100cm^2 for beta smears. In accordance with the DQAP and the Radiation Protection Program (RPP), GEL analyzed tritium smears (method GL-RAD-A-002) and also counted 48 smears for quality assurance by gas proportional counting (method GL-RAD-A-001).

The radiological data generated was intended only to provide guidance for future decontamination and remediation activities for these areas. Use of properly calibrated instruments with operational checks and duplicate measurements as part of the survey process provided data quality indicators. The data quality was acceptable with no deviations from measurement protocols and reasonable agreement with duplicate measurements.

The following was performed during this characterization program:

- 396 alpha/beta direct measurements;
- 412 alpha/beta removable measurements;
- 10 tritium measurements;
- 403 dose rate measurements;
- Beta scans;
- Gamma scans; and,
- 69 ISOCs measurements.

Table 2-9 provides the maximum results for each type of survey conducted.

Table 2-9 2018 Maximum Results by Measurement Type

Measurement	Criteria	Maximum	Location (Survey Unit)	Description
Beta Scan	500 cpm	30,000 cpm	105D1 (Buffer Seal System)	Valve SL-57V (internal contamination)
Gamma Scan	5,000 cpm	73,166,400 cpm	105D1 (Buffer Seal System)	buffer seal coolers (internal contamination)
Alpha Static	27 dpm / $100cm^2$	264 dpm / $100cm^2$	306A1 (Cargo Deck 4 C Deck Starboard)	deck
Beta Static	7,100 dpm / $100cm^2$	722,172 dpm / $100cm^2$	105D1 (Buffer Seal System)	booster pump (internal contamination)
Alpha Removable	27 dpm / $100cm^2$	14 dpm / $100cm^2$	109A1 (Hot Chemistry Lab)	box
Beta Removable	7,100 dpm / $100cm^2$	7,576 dpm / $100cm^2$	105D1 (Buffer Seal System)	booster pump
Tritium (H-3)	1.2E8 dpm / $100cm^2$	6,638 dpm / $100cm^2$	107A1 (Gas Absorption/Radiation Sampling Room)	sample sink
Dose Rate	None (information only)	400 μR / h	119D2 (Sources)	steering gear sources

Tables 2-10 and 2-11 summarize the results by survey unit. Not Applicable is indicated by N/A.

Table 2-10 NSS 2018 Characterization Summary Table A

Survey Unit	Beta Scan Maximum CPM	Alpha Static >27 DPM/100cm^2	Alpha Static Maximum DPM/100cm^2	Beta Static >7,100 DPM/100cm^2	Beta Static Maximum DPM/100cm^2	Alpha Removable >27 DPM/100cm^2	Alpha Removable Maximum DPM/100cm^2
103A1	8,389	0	20	1	72,811	0	6
104A1	772	0	15	0	3,502	0	6
105A1	527	0	15	0	847	0	3
106A1	979	0	17	0	898	0	3
107A1	3,703	0	11	2	49,441	0	6
108A1	1,100	0	18	0	560	0	3
109A1	1,543	0	4	0	2,247	0	14
111A1	356	1	57	0	558	0	10
201A1	8,093	0	18	0	2,900	0	3
202A1	289	0	6	0	657	0	3
203A1	319	0	9	0	1,475	0	3
301A1	238	1	41	0	627	0	6
302A1	2,413	0	3	0	47	0	1
303A1	645	0	3	0	449	0	13
304A1	803	0	18	0	373	0	4
305A1	742	0	11	0	880	0	3
306A1	532	13	264	0	359	0	3
307A1	11,036	0	6	0	1,099	0	3
309A1	805	0	23	0	1,727	0	3
401B1	587	4	76	0	3,213	0	3

Table 2-10 NSS 2018 Characterization Summary Table A (continued)

Survey Unit	Beta Scan Maximum CPM	Alpha Static >27 DPM/100cm^2	Alpha Static Maximum DPM/100cm^2	Beta Static >7,100 DPM/100cm^2	Beta Static Maximum DPM/100cm^2	Alpha Removable >27 DPM/100cm^2	Alpha Removable Maximum DPM/100cm^2
105D1	30,000	0	17	9	722,172	0	3
113D1	30	N/A	N/A	0	0	0	3
119D2	N/A	N/A	N/A	N/A	N/A	0	3
308A1	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table 2-11 NSS 2018 Characterization Summary Table B

Survey Unit	Beta Removable>7,100 DPM/100cm^2	Beta Removable Maximum DPM/100cm^2	Dose Rate Maximum µrem/h	Tritium Removable >1.6E8 DPM/100cm^2	Tritium Removable Maximum DPM/100cm^2
103A1	0	453	9	-	-
104A1	0	481	12	0	3,352
105A1	0	69	3	0	41
106A1	0	16	7	0	16
107A1	0	4,022	120	0	6,638
108A1	0	14	3	0	63
109A1	0	1,667	2	0	855
111A1	0	23	3	-	-
201A1	0	14	3	-	-
202A1	0	27	4	-	-
203A1	0	27	3	-	-

Table 2-11 NSS 2018 Characterization Summary Table B (continued)

Survey Unit	Beta Removable>7,100 DPM/100cm^2	Beta Removable Maximum DPM/100cm^2	Dose Rate Maximum µrem/h	Tritium Removable >1.6E8 DPM/100cm^2	Tritium Removable Maximum DPM/100cm^2
301A1	0	13	4	-	-
302A1	0	4	3	-	-
303A1	0	20	2	-	-
304A1	0	66	2	0	17
305A1	0	20	3	-	-
306A1	0	23	2	-	-
307A1	0	27	4	-	-
309A1	0	16	1.5	0	2
401B1	0	27	2	-	-
105D1	1	7,576	10	N/A	N/A
113D1	0	34	1.5	N/A	N/A
119D2	0	65	400	N/A	N/A
308A1	N/A	N/A	N/A	N/A	N/A

2.3.5 Characterization Activities in 2019

A detailed profile of radiological and non-radiological contaminants on the ship in radiological spaces was performed in 2019, again by RSCS. The results of the characterization efforts are documented in CR-109 [Reference 2-5]. This scope of work was to perform a radiological and environmental hazard characterization for the RC and CV. The intention of this work was not to repeat previous characterization efforts, but to fill in the gaps and expand the characterization data in certain areas.

Radiological surveys were performed with the following instrumentation:

- Ludlum Telepole;
- Thermo RadEye SX with Ludlum Model 43-93 detector (alpha/beta directs and beta scans);
- Thermo RadEye SX with Ludlum Model 44-10 (gamma scans);
- Ludlum Model 3 with Ludlum Model 44-9 (beta scans);
- Bicron MicroRem (dose rate);
- Ludlum Model 19 (dose rate);
- AMP-100 dose rate probe;
- Ludlum 3030E with a 43-10-1 probe above a fixed position smear counter; and,
- NaIs and OSPREY ISOCS Characterized.

Instruments were properly calibrated, and beta efficiencies were determined with Tc-99, alpha efficiencies with Th-230 and gamma calibrations with Cs-137. Source check responses were established for each instrument with plus and minus two standard deviation values determined. Operational checks were performed each day prior to use. Control charts for background and daily source checks were established for the counting instruments such as the RadEye with 43-83 alpha beta probe and the Ludlum 3030 with the 43-10-1 gross alpha/beta probe.

For the survey design, the Co-60, H-3, and Am-241 screening values, taken from NUREG/CR-5512, Vol. 3, Table 5.19 [Reference 2-14], were used as investigation levels.

Equation 2-1 was used to calculate the Minimum Detectable Activity (MDA):

$$MDA\ (dpm) = \frac{3+3.29\sqrt{R_b t_s\left(1+\frac{t_s}{t_b}\right)}}{eff * t_s} \qquad \text{Equation 2-1}$$

Where:

R_b = *background count rate (c/m)*

t_s = *sample count time (min)*

t_b = *background count time (min)*

eff = *counting efficiency in counts per disintegration (c/d)*

Baseline radiation and contamination surveys were performed in the Reactor Compartment and Containment Vessel between April 5 and April 30, 2019. The surveys were conducted to evaluate area conditions for end state and decommissioning planning. MARSSIM scans were not performed since component removals or remediations during decommissioning were deemed likely to change the locations of surface contamination.

The surveys consisted of:

- General Area, contact, and 30 cm dose rates (not background corrected);
- 100cm^2 removable contamination smears and gross wipes/smears;
- Biased scans with alpha-beta or beta-gamma detectors; and,
- Fixed position counts with alpha-beta detectors.

The quantity of smear samples and measurements for the baseline surveys are summarized in Table 2-12. In addition to the 273 100cm^2 smears documented on the surveys, several dozen large area smears were obtained to get sufficient general area surface contamination activity for a reliable radionuclide mix to be generated.

Table 2-12 2019 Baseline Survey Measurement Summary (Quantity Collected)

Location	100cm^2 Smear Count	Fixed Position Scans	Surface Scans	GA Dose Rate	Contact/30 cm
CV Level 1	30	6	14	12	6
CV Level 2	30	5	12	14	11
CV Level 3	30	4	12	12	9
CV Level 4	26	5	16	14	4
RC Upper Level A	14	3	12	8	1
RC Upper Level B	27	7	20	15	2
RC Mid-Level C & D	54	1	32	16	2
RC Lower Level Forward	42	6	30	17	17
RC Lower Level Aft	20	1	12	10	6
Total	**273**	**38**	**160**	**118**	**58**

Scans and fixed position measurements are reported in corrected counts per minute (ccpm) with background levels in the areas ranging from 0 – 0.5 cpm alpha and 100 – 20,000 cpm beta. The highest alpha and beta backgrounds reported in each area in Table 2-12 were used along with Equation 1 to calculate the minimum detectable corrected counts per minute (MDCC) and MDAs for fixed position readings that used a 1-minute background (tb) and a 3-minute count time (ts). Comparison of the MDAs to the screening DCGLs in Table 2-13 demonstrates that adequate sensitivity was maintained even in the highest background locations with worst case MDAs of 49% of the Co-60 [7,100 dpm/100cm^2 DCGL] and 12% of the Cs-137 [28,000 dpm/100cm^2 DCGL].

Table 2-13 Fixed Position Alpha (α) Beta (β) Sensitivity in Highest α β Background per Survey Area

Location	Alpha Bkg cpm	Beta Bkg cpm	Alpha MDCC ccpm	Beta MDCC ccpm	Alpha MDA dpm	Beta MDA dpm	% Co-60 DCGL	% Cs-137 DCGL
CV Level 1	0.5	2000	4	171	9	1098	15%	4%
CV Level 2	0.5	15000	4	466	9	2997	42%	11%
CV Level 3	0.5	15000	4	466	9	2997	42%	11%
CV Level 4	0.5	20000	4	538	9	3459	49%	12%
RC Lower Level AFT	0	350	1	72	3	463	7%	2%
RC Lower Level FWD	0	1500	1	148	3	952	13%	3%
RC Mid-Level C Deck	0	130	1	44	3	285	4%	1%
RC Mid-Level D Deck Aft	0	130	1	44	3	285	4%	1%
RC Mid-Level D Deck	0	130	1	44	3	285	4%	1%
RC Upper Level B Deck	0	110	1	41	3	262	4%	1%
RC Upper Level A Deck	0	110	1	41	3	262	4%	1%
RC Upper Level A Deck	0	110	1	41	3	262	4%	1%
RC Upper Level Ventilation	0	100	1	39	3	251	4%	1%
CV RC Systems	0.5	20000	4	538	9	3459	49%	12%

The system characterization was focused on obtaining representative samples of interior contaminants. This included smears as well as samples of any sludges, liquids, or other materials present. It also focused on obtaining interior beta and gamma dose rates to supplement dose to curie estimates of the source terms present. System access and sampling was performed in accordance with a detailed Work Order and daily Job Hazards Analysis briefs. Area surveys, system surveys, and sampling were performed using project-specific procedures for the instruments used and sampling performed.

On-site gamma spectroscopy measurements were performed with a 2x2 stabilized and characterized sodium iodide detector with a lead collimator that was also portable for access to the various locations within the ship. The geometry composer software was used to create a smear composite, paint chip sample, as well as liquid and sludge sample container geometries for on-site analysis.

In accordance with the DQAP and the RPP, GEL also analyzed baseline survey smears with positive activity for quality assurance by gas proportional counting (method GL-RAD-A-001) and tritium smears (method GL-RAD-A-002). System smear sample composites were also analyzed by gamma spectroscopy at the same laboratory. These results were reviewed and five of the smear composites were selected for hard-to-detect analysis to provide C-14, Ni-63, Sr-90 and Tc-99 results. These Radionuclides of Concern (ROCs) were obtained during characterization planning. The bolded and italicized ID cells in Table 2-14 were selected for hard-to-detect analysis.

Results of characterization surveys on external surfaces of system components and decks show that there is very little removable and total activity. The results of internal system component surveys only established removable levels and not total activity or whether radioactivity has penetrated the material.

The required MDA for gamma spectrometry was set at 25 pCi/filter for Cs-137. The analysis method was by DOE HASL 300, 4.5.2.3/Ga-01-R.

Table 2-14 Off-site Laboratory Smear Composite Gamma Spectroscopy Results

Sample ID	Location	Co-60 pCi	Ag-108m pCi	Cs-137 pCi	Co/Cs Ratio
RCCV-RS-006	Smears from PRT PR-T1 Effluent Condensing Tank	6.02E+02			
RCCV-RS-010	Smears from Steam generator	3.58E+04		1.39E+02	2.58E+02
RCCV-RS-002	Smears from PD-T1 Lab Waste Tank	7.67E+03	1.79E+01	3.68E+01	2.08E+02
RCCV-RS-007	Smears from Pressurizer	4.26E+04		3.39E+02	1.26E+02
RCCV-RS-001	Smears from IX piping	8.21E+04	2.52E+02	9.09E+02	9.03E+01
RCCV-RS-003	Smears from PD-T2	1.05E+03	1.73E+01	2.33E+01	4.51E+01
RCCV-RS-004	Smears from PD-T3	2.40E+03		2.27E+02	1.06E+01
RCCV-RS-011	Smears from SLT-1 Buffer Seal Surge Tank	1.79E+03	1.07E+01	6.57E+02	2.72E+00
RCCV-RS-005	Smears from PD-T4	3.96E+03	3.94E+01	3.05E+03	1.30E+00
RCCV-RS-009	Smears from Emergency Cooling Heat Exchanger	4.06E+02	1.45E+02	7.72E+02	5.26E-01
RCCV-RS-008	Smears from SC-E1 pipe Shutdown Circ. sys.	1.95E+02	4.27E+01	8.31E+02	2.35E-01
RCCV-RS-012	Smears from Ventilation	2.15E+02	1.65E+01	9.81E+03	2.19E-02

Note: The bolded and italicized Sample ID cells in Table 2-14 were selected for hard-to-detect analysis.

Table 2-15 presents the off-site laboratory results for the steam generator composite with the hard-to-detect radionuclides. The required MDA for H-3 analysis was 150 pCi/filter. The analysis method was by EPA 906.0 Modified. The required MDA for C-14 analysis was 10 pCi/filter. The analysis method was by EPA EERF C-01 Modified. The required MDA for Ni-63 analysis was 20 pCi/filter. The analysis method was by DOE RESL Ni-1, Modified. The required MDA for Tc-99 analysis was

10 pCi/filter. The analysis method was by DOE EML HASL-300, Tc-02-RC Modified. The required MDA for Sr-90 analysis was 2 pCi/filter. The analysis method was by EPA 905.0 Modified /DOE RP501 Rev. 1 Modified.

The composite activities in Tables 2-15 through Table 2-19 were compared to the 90th percentile building screening values from NUREG/CR-5512, Volume 3, Table 5.19 [Reference 2-15]. There is no value for Ag-108m in the screening table; therefore, the value for Ag-110m was used as a substitute for the calculations.

Table 2-15 Steam Generator Composite Smear Results Compared to Screening DCGLs

Nuclide	Total Composite pCi	Total Composite DPM	Screening DCGL DPM/100cm^2	Activity/ Screening DCGL Fraction	Normalized Fraction
C-14	3.50E+02	7.77E+02	3.70E+06	2.10E-04	3.97E-06
Co-60	3.58E+04	7.95E+04	7.10E+03	1.12E+01	2.12E-01
Ni-63	3.38E+07	7.50E+07	1.80E+06	4.17E+01	7.88E-01
Sr-90	<MDA	<MDA	8.70E+03	0.00E+00	0.00E+00
Tc-99	7.90E+01	1.75E+02	1.30E+06	1.35E-04	2.55E-06
Ag-108m	<MDA	<MDA	1.00E+04	0.00E+00	0.00E+00
Cs-137	1.39E+02	3.09E+02	2.80E+04	1.10E-02	2.08E-04
Total				**5.29E+01**	**1.00E+00**

Table 2-16 presents the off-site laboratory results for the Pressurizer composite with the hard-to-detect radionuclides.

Table 2-16 Pressurizer Composite Smear Results Compared to Screening DCGLs

Nuclide	Total Composite pCi	Total Composite DPM	Screening DCGL DPM/100cm^2	Activity/ Screening DCGL Fraction	Normalized Fraction
C-14	2.23E+03	4.95E+03	3.70E+06	1.34E-03	8.60E-05
Co-60	4.26E+04	9.46E+04	7.10E+03	1.33E+01	8.56E-01
Ni-63	1.79E+06	3.97E+06	1.80E+06	2.21E+00	1.42E-01
Sr-90	1.39E+00	3.09E+00	8.70E+03	3.55E-04	2.28E-05
Tc-99	2.74E+01	6.08E+01	1.30E+06	4.68E-05	3.01E-06
Ag-108m	<MDA	<MDA	1.00E+04	0.00E+00	0.00E+00
Cs-137	3.39E+02	7.53E+02	2.80E+04	2.69E-02	1.73E-03
Total				**1.56E+01**	**1.00E+00**

Table 2-17 presents the off-site laboratory results for the Containment Drain Tank (PD-T4) composite with the hard-to-detect radionuclides.

Table 2-17 Containment Drain Tank Composite Smear Results Compared to Screening DCGLs

Nuclide	Total Composite pCi	Total Composite DPM	Screening DCGL DPM/100cm^2	Activity/ Screening DCGL Fraction	Normalized Fraction
C-14	5.10E+01	1.13E+02	3.70E+06	3.06E-05	1.77E-05
Co-60	3.96E+03	8.79E+03	7.10E+03	1.24E+00	7.16E-01
Ni-63	1.95E+05	4.33E+05	1.80E+06	2.41E-01	1.39E-01
Sr-90	<MDA	<MDA	8.70E+03	0.00E+00	0.00E+00
Tc-99	1.42E+01	3.15E+01	1.30E+06	2.42E-05	1.40E-05
Ag-108m	3.94E+01	8.75E+01	1.00E+04	8.75E-03	5.06E-03
Cs-137	3.05E+03	6.77E+03	2.80E+04	2.42E-01	1.40E-01
Total				**1.73E+00**	**1.00E+00**

Table 2-18 presents the off-site laboratory results for the RC Exhaust Ventilation composite with the hard-to-detect radionuclides.

Table 2-18 RC Exhaust Ventilation Composite Smear Results Compared to Screening DCGLs

Nuclide	Total Composite pCi	Total Composite DPM	Screening DCGL DPM/100cm^2	Activity/ Screening DCGL Fraction	Normalized Fraction
C-14	6.43E+02	1.43E+03	3.70E+06	3.86E-04	4.47E-04
Co-60	2.15E+02	4.77E+02	7.10E+03	6.72E-02	7.79E-02
Ni-63	1.20E+04	2.66E+04	1.80E+06	1.48E-02	1.72E-02
Sr-90	<MDA	<MDA	8.70E+03	0.00E+00	0.00E+00
Tc-99	<MDA	<MDA	1.30E+06	0.00E+00	0.00E+00
Ag-108m	1.14E+01	2.53E+01	1.00E+04	2.53E-03	2.93E-03
Cs-137	9.81E+03	2.18E+04	2.80E+04	7.78E-01	9.02E-01
Total				**8.63E-01**	**1.00E+00**

Table 2-19 presents the off-site laboratory results for the Primary Purification RC IX piping composite with the hard-to-detect radionuclides.

Table 2-19 Primary Loop RC IX Piping Composite Smear Results Compared to Screening DCGLs

Nuclide	Total Composite pCi	Total Composite DPM	Screening DCGL DPM/100cm^2	Activity/ Screening DCGL Fraction	Normalized Fraction
C-14	1.48E+04	3.29E+04	3.70E+06	8.88E-03	1.03E-02
Co-60	8.21E+04	1.82E+05	7.10E+03	2.57E+01	2.98E+01
Ni-63	5.24E+06	1.16E+07	1.80E+06	6.46E+00	7.49E+00
Sr-90	1.59E+00	3.53E+00	8.70E+03	4.06E-04	4.70E-04
Tc-99	1.67E+02	3.71E+02	1.30E+06	2.85E-04	3.31E-04
Ag-108m	2.52E+02	5.59E+02	1.00E+04	5.59E-02	6.48E-02
Cs-137	9.09E+02	2.02E+03	2.80E+04	7.21E-02	8.35E-02
Total				**8.63E-01**	**1.00E+00**

The results of the RPV, Internals and NST Sampling in 2005 were also re-evaluated in 2019. The 2005 activated metal sample results were decayed to 2019. The decayed activity concentrations were converted from µCi/g to µCi/cc by multiplying by the average density of steel at 8 g/cc to determine the waste classification in accordance with 10 CFR Part 61.55. Table 2-20 presents the total activity concentrations and classification of the activated components.

Table 2-20 Component Activity and Classification Summary

Component	Total µCi/cc	Class
NST Outer Diameter (OD)	2.71E-02	A
NST Inner Diameter (ID)	2.38E-02	A
RPV OD	2.82E-02	A
RPV ID	5.23E-01	A
Outer Thermal Shield (OTS) ID/OD	1.96E+01	A
Middle Thermal Shield (MTS) OD	7.58E+01	B
MTS ID	1.40E+02	B
Total	**2.36E+02**	

2.3.6 Survey of Exterior Hull in 2019

A MARSSIM-based radiological survey was performed by RSCS on the NSS exterior hull in September and October of 2019 while the ship was on drydock in Philadelphia, PA. The results of this survey are documented in CR-143 [Reference 2-12]. This survey was performed while the ship was on drydock because this was the only time that the exterior hull would be readily accessible to perform these surveys. This survey was performed following the methodology of a FSS, as described in the decommissioning activities guided by MARSSIM.

As part of the drydock maintenance activities, the underwater portion of the hull was stripped of paint. These surveys were performed after paint stripping (on the bare hull metal) to document the radiological conditions of the hull structural materials directly where available.

The survey design was a MARSSIM based approach using scans, static measurements, smears, and dose rate measurements in nine identified survey units. *Visual Sample Plan (VSP)* [Reference 2-16] software was used to create maps and to randomly select sample locations within each survey unit.

The hull was designated as a MARSSIM Class 3 area. The exterior hull of the ship contained a total of nine (9) survey units. Table 2-21 presents the nine survey units; Figure 2-1 depicts the starboard side and Figure 2-2 the port side.

Table 2-21 Hull Survey Units

FSS Unit Number	MARSSIM Survey Unit Class	Description of Hull Area Surveyed
FSS-310A1	3	Starboard Fore
FSS-311A1	3	Starboard Mid
FSS-312A1	3	Starboard Aft
FSS-313A1	3	Port Fore
FSS-314A1	3	Port Mid
FSS-315A1	3	Port Aft
FSS-316A1	3	Starboard Above Boot Stripe
FSS-317A1	3	Port Above Boot Stripe
FSS-318A1	3	Rudder

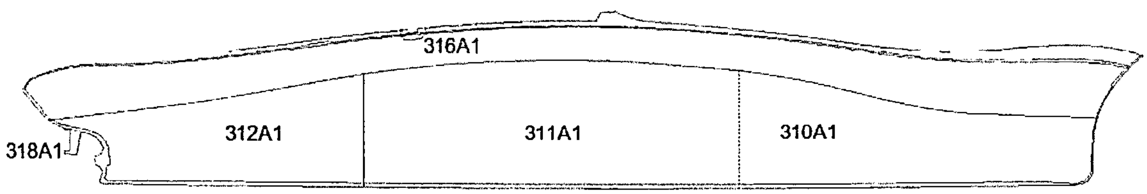

Figure 2-1 NSS Starboard Survey Units

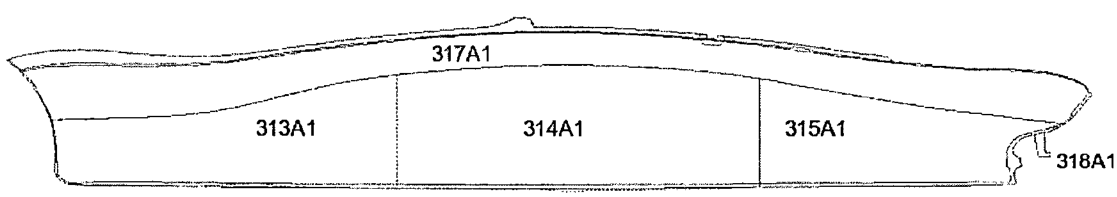

Figure 2-2 NSS Port Survey Units

The screening DCGL for Co-60 (most restrictive) of 7,100 dpm /100cm^2 for beta (NUREG 1757, Vol 2, MARSSIM, App H, Table H.1) [Reference 2-15] was used for design of these surveys. It should be noted that this value is less than the Co-60 DCGL presented in Chapter 6. Alpha emitting radionuclides were not included in the screening DCGLs based upon the results of the recent characterization of the CV/RC showing no significant alpha were present; therefore, the planned approach did not include the scanning or static measurements for alpha on the hull.

These surveys were designed to collect the specified radiological samples and analysis for these areas (scan, static, smear, dose rate) as laid out in MARSSIM. For Class 3 survey areas, surface scan surveys have judgmental coverage, and given the degree of difficulty to access the survey areas, scan surveys are planned to be performed in the vicinity of the static locations.

The following measurements were performed during this MARSSIM Type Survey:

- 180 beta direct measurements;
- 180 beta removable measurements;
- 180 gamma dose rate measurements; and,
- Beta scan surveys.

Smears were counted at the RSCS laboratory on a low background gas proportional counting system.

Radiological surveys were performed with the following instrumentation:

- Thermo RadEye SX with Ludlum Model 43-89 dual-phosphor scintillation detector (beta scans);
- Thermo RadEye SX with 43-93 dual-phosphor scintillation detector (direct static beta measurements and beta scans); and,
- Bicron MicroRem (dose rate measurements).

The surveys were performed with calibrated instrumentation following the requirements of STS-005-008, *Radiological Instrumentation and Survey Documentation*. Swipes were stored in a safe condition following STS-005-019, *Chain of Custody of Samples*. All survey instruments were properly calibrated, and beta efficiencies were conservatively determined with Tc-99 and gamma efficiencies with Cs-137. Instruments were operationally checked daily.

A MARSSIM survey package was created for each survey unit, which included specific instructions for collection of measurements and data logging, figures with locations, and documentation for recording survey activities/results.

A summary of the survey results is provided in Tables 2-21 and 2-22.

Table 2-22 Descriptive Statistics, Beta Static Measurement Data, Background Corrected

Beta – β Statistic	Survey Unit (FSS)								
	310A1	311A1	312A1	313A1	314A1	315A1	316A1	317A1	318A1
Number of Measurements	20	20	20	20	20	20	20	20	20
Arithmetic Mean	-402.60	3.89	-22.83	-26.96	-280.22	-351.96	-146.74	-90.65	30.00
Standard Deviation (sample)	151.50	341.28	265.74	201.24	133.41	104.37	94.07	106.45	80.74
Standard Error of the Mean	33.88	78.29	59.42	45.00	29.83	23.34	21.03	23.80	18.05
Coefficient of Variation	-0.38	87.73	-11.64	-7.47	-0.48	-0.30	-0.64	-1.17	2.69
Maximum	-143.48	847.83	460.87	365.22	21.74	-156.52	-13.04	100.00	182.61
Median	-432.61	-139.13	-143.48	-19.57	-297.83	-323.91	-134.78	-86.96	34.78
Minimum	-665.22	-460.87	-291.30	-452.17	-439.13	-560.87	-343.48	-321.74	-121.74
Range	521.74	1308.70	752.17	817.39	460.87	404.35	330.43	421.74	304.35
UCL95 (median)	66.40	153.45	116.46	88.19	58.47	45.74	41.23	46.65	35.38
LCL95 (median)	2.12	4.91	3.73	2.82	1.87	1.46	1.32	1.49	1.13

Table 2-23 NSS Hull MARSSIM Survey Summary

Survey Unit	Beta Scan Average CPM	Beta Static >7,100 DPM / 100cm^2	Beta Static Maximum DPM / 100cm^2	Beta Removable >7,100 DPM / 100cm^2	Beta Removable Maximum DPM / 100cm^2	Net Dose Rate Maximum µrem/h
FSS-310A1	-34.20	0	-143.48	0	11.41	2
FSS-311A1	27.00	0	847.43	0	14.97	5
FSS-312A1	19.80	0	460.87	0	11.41	2
FSS-313A1	12.42	0	365.22	0	14.97	2
FSS-314A1	7.80	0	21.74	0	11.41	3
FSS-315A1	-3.18	0	-156.52	0	14.97	3
FSS-316A1	1.06	0	26.09	0	11.41	1
FSS-317A1	5.70	0	100.00	0	14.97	1
FSS-318A1	10.20	0	182.60	0	14.97	1

The MARSSIM survey of the exterior hull demonstrates that:

- No unexpected results or trends are evident in the data;
- The sampling and survey results demonstrate that residual radioactivity in the survey areas are indistinguishable from background levels;
- The data quality meets the necessary requirements and is deemed to be acceptable for its intended purpose;
- The amount of data collected from each survey unit is adequate to provide the required statistical confidence needed to decide that the DCGLs are met; and,
- All measurements were below the screening DCGL.

2.3.7 Sampling NST lead in 2021

In October 2021, preparations were made for removal and release of the NST lead shielding. Lead samples were collected by drilling six ¼ inch diameter holes into the lead shielding at approximately 60 degree intervals. These holes were drilled below the shield tank cooling coils in the region of the highest neutron flux. Figure 2-3 shows the active core region by the black rectangle in the center of the figure. Two samples were collected at each location, an inner and outer sample. The purpose of the sampling was to verify whether any neutron activation occurred in the lead. In accordance with the DQAP and the RPP, samples were sent to GEL for gamma spectrometry and analysis of Hard-to-Detect radionuclides. Only one sample had a positive result at 0.15 pCi/g Cs-137. This result was attributed to cross contamination. The laboratory results are described in CR-144 [Reference 2-14].

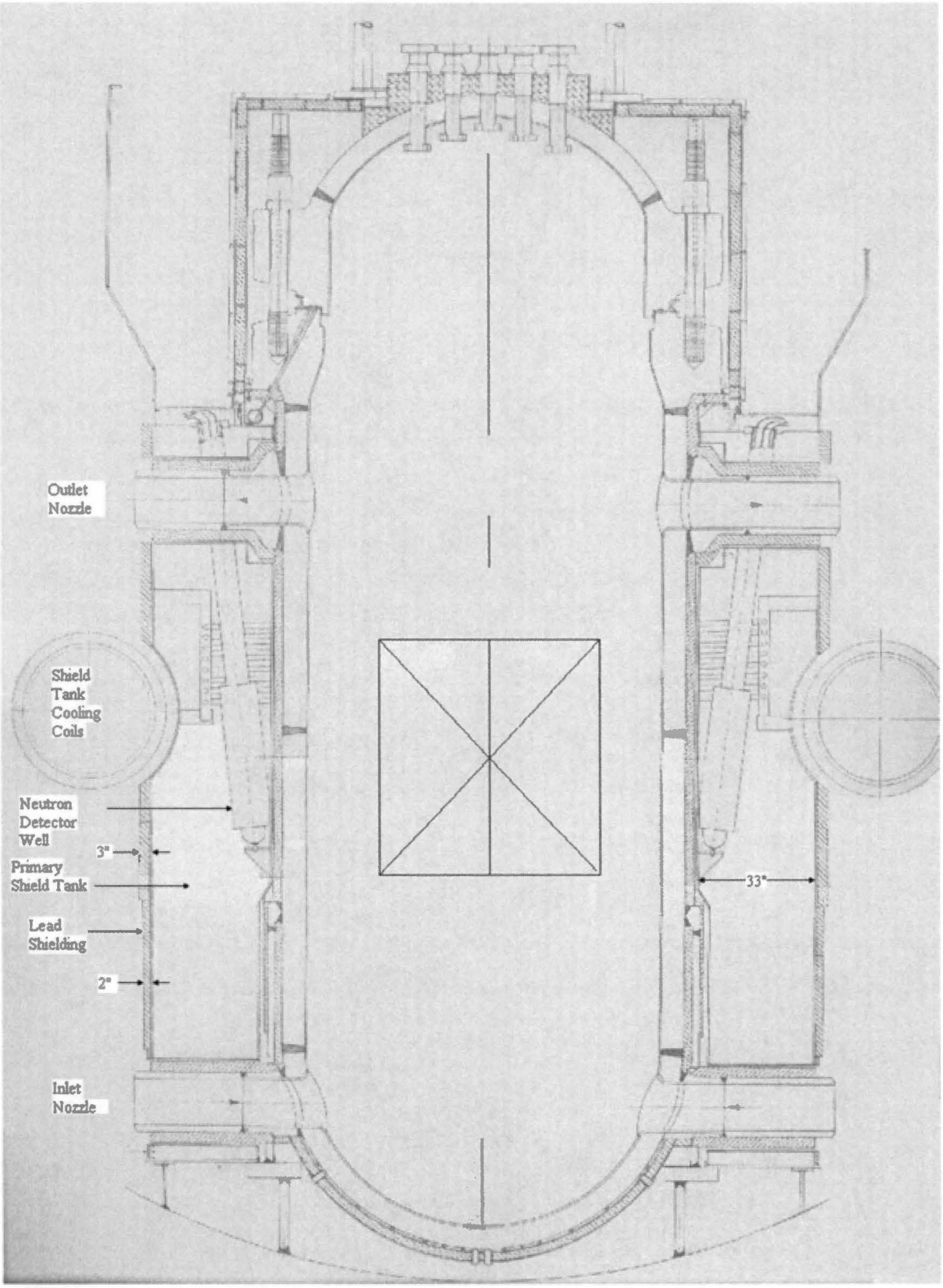

Figure 2-3 Neutron Shield Tank

2.3.8 Engine Room Survey in 2022

In November 2021, MARAD determined that a more detailed in-depth survey of the secondary systems contained in the engine room would be prudent to support license termination planning, especially with respect to the shipbreaking end state condition. The effort included removal of asbestos-containing insulation materials as a prerequisite activity. With the ACM removed, a survey package was developed and executed in 2022.

The survey package identified eighteen (18) survey points of the internal surfaces of the Engine Room steam, condensate, and feedwater system components for measurements. For all but one survey point, either two (2) or four (4) locations for measurements were specified; for survey point, number eighteen (18), eight (8) locations for measurements were specified. The technicians were required to collect a three (3) minute Total Surface Activity (TSA) measurement at the survey location with a Ludlum model 43-93 detector, collect a 100 cm^2 smear at the TSA location and perform a beta scan survey of 100% of the accessible system internals. The technicians were also required to collect a three (3) minute local area instrument background (beta and alpha) with the model 43-93 detector at or in the general area of the survey location. The Minimum Detectable Concentrations (MDCs) ranged from 311 to 377 dpm/100 cm^2. Six smears were sent offsite for gamma isotopic analysis. Only one smear had positive Co-60 activity of 23 dpm/filter. This smear was from the Main Steam Moisture Separator. Based upon this analysis, the Main Steam system has been classified as Impacted.

2.4 Initial Classifications

The initial classifications were obtained from the preliminary classifications presented in the HSA. Those preliminary classifications were based upon the screening values for residual surface radioactivity for building surfaces presented in NUREG-1757, Volume 2, Table H-1. During development of the LTP, MARAD completed reviews of 1) radiological survey data (routine and non-routine) over the ship's operating history since 2008 and 2) the 2018 and 2019 characterization survey results compiled for the decommissioning.

2.4.1 Systems

During development of the LTP, an extensive review of systems was conducted by MARAD to determine those systems that contain radioactive materials or in which radioactive material was detected at some time during the operating history of the plant. Table 2-24 provides a listing of plant systems and their status relative to the potential for radioactivity. The assessment considers the internal portions of the systems. Systems that might be assessed as non-impacted and are located in contaminated areas may themselves be externally contaminated and may be considered for remediation or disposal as radioactive waste.

As described in Section 3.1.1 of this LTP, several components of the nuclear power plant are planned for retention, subject to confirmation before license termination. See Sections 3.1.1, and 5.9.2 through 5.9.5 for additional information.

Table 2-24 Initial Classification of Systems

Survey Unit	System Name	Status
SYS-101	Reactor Space Ventilation System (RSV)	Impacted
SYS-102	Emergency Cooling System (DK)	Impacted
SYS-103	Soluble Poison Shutdown System (SP)	Impacted
SYS-104	Primary Loop Purification System (PP)	Impacted
SYS-105	Buffer Seal System (SL)	Impacted
SYS-106	Hydrogen Addition System (HA)	Impacted
SYS-107	Primary Relief System (PR)	Impacted
SYS-108	Primary Sampling System (SA)	Impacted
SYS-109	Intermediate Cooling System (CW)	Impacted
SYS-110	Containment Cooling System (CC)	Impacted
SYS-111	Shutdown Circulation (SC)	Impacted
SYS-112	Primary Pressurizer System (PE) including retained portions of the pressurizer	Impacted
SYS-113	Radioactive Waste and Dilution System (WD)	Impacted
SYS-114	Equipment Drain and Waste Collection System (PD) The remaining components are the Fresh Water Shield Tank, PD-T6 Contaminated Water Tank Port, and PD-T5 Contaminated Water Tank Starboard	Impacted
SYS-115	Gaseous Waste Collection and Disposal System (WL)	Impacted
SYS-116	Main Steam System including retained portions of the Steam Generators	Impacted
SYS-117	Neutron Shield Tank and Fuel Transfer Tank (retained walls)	Impacted
SYS-401	Ships Ventilation System	Non-impacted
SYS-402	Service Water System	Non-impacted
SYS-403	Potable Water System	Non-impacted
SYS-404	Service Air System	Non-impacted
SYS-405	Lubrication Oil System	Non-impacted
SYS-406	Auxiliary Steam System	Non-impacted
SYS-407	Main Condensate and Feedwater System	Non-impacted

2.4.2 Structures

Classification is the process by which an area or survey unit is described according to its radiological characteristics. The significance of classification is that this process determines the FSS survey design and the procedures used to develop this design. In classifying areas, those that have no reasonable potential for residual contamination are classified as non-impacted areas. These areas have no radiological impact from site operations and are typically identified early in decommissioning. Areas with some potential for residual contamination are classified as impacted areas. Impacted areas are further divided into one of three classifications as defined by MARSSIM:

- Class 1 areas: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) above the DCGL. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or unplanned releases are known to have occurred, 3) waste storage sites, and 4) areas with contaminants in discrete solid pieces of material with high specific activity. Note that areas containing contamination in excess of the DCGL prior to remediation would generally be classified as Class 1 areas unless ample evidence exists to show that a lower classification is justified;
- Class 2 areas: These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGL. To justify changing the classification from Class 1 to Class 2, there should be measurement data that provides a high degree of confidence that no individual measurement would exceed the $DCGL_W$. Other justifications for reclassifying an area as Class 2 may be appropriate based on sites-specific considerations. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (e.g., process facilities), 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper bulkheads and overheads of some buildings or rooms subjected to airborne radioactivity, 5) areas where low concentrations of radioactive materials were handled, and 6) areas on the perimeter of former contamination control areas; and,
- Class 3 areas: Any impacted areas that are not expected to contain any residual radioactivity or are expected to contain levels of residual radioactivity at a small fraction of the DCGL, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

Class 1 areas have the greatest potential for contamination and therefore receive the highest degree of survey effort for the final status survey, followed by Class 2 areas, and then by Class 3 areas. Non-impacted areas do not require any level of survey coverage because they have no potential for residual contamination. As a survey progresses, reevaluation of classifications may be necessary based on newly acquired survey data. The FSS plan includes a process by which measurements that approach pre-defined action levels (fractions of the DCGLs) are investigated to see if reclassification of an area(s) is necessary.

The initial classifications were obtained from the preliminary classifications presented in the HSA. Those preliminary classifications were based upon the screening values for residual surface radioactivity for building surfaces presented in NUREG-1757, Volume 2, Table H-1. During development of the LTP, MARAD completed reviews of 1) radiological survey data (routine and non-routine) over the ship's operating history since 2008 and 2) the 2018 and 2019 characterization survey results compiled for the decommissioning. The 2018 and 2019 radiological characterization survey data in particular represents a substantial volume of information.

The survey results will be notated in the FSS packages with the survey unit number shown in Table 2-25. The preliminary survey unit numbers assigned in the 2018 characterization effort will not be used (i.e., the numbers in Tables 2-10 and 2-11).

Based upon the nature and extent of contamination detected and the areas to be surveyed, some of the survey unit decks and bulkheads below 2 meters high will be treated as either a Class 1 or Class 2 survey unit. Depending on the area, the bulkhead above 2 meters and the overhead will be treated as either a Class 2 or Class 3 survey unit.

Table 2-25 Initial Classification of Structures or Rooms

Survey Unit #	Survey Unit Description	Deck or Elevation	Initial Classification
STR-101	Containment Vessel (CV) – 1st Level (Tanktop)	5'	1
STR-102	Containment Vessel (CV) – 2nd Level (Flat)	14'	1
STR-103	Containment Vessel (CV) – 3rd Level (D Deck)	23'	1
STR-104	Containment Vessel (CV) – 4th Level (C Deck)	32'	1
STR-105	Reactor Compartment – Lower Level	5' – 23'	1
STR-106	Port Stabilizer Room and Port Booster Pump Room	Hold	1
STR-107	Port Charging Pump Room	Hold	1
STR-108	Starboard Charging Pump Room	Hold	1
STR-109	Auxiliary Access Trunk, C-Deck the Cold Water Chemistry Lab (Port) and Radiation Monitoring Room (Stbd)	A - C	1
STR-110	Gas Absorption Equipment Room/ Radiation Sampling Room	D	1
STR-201	Reactor Compartment – Mid Level D Deck	23'	2
STR-202	Reactor Compartment – Mid Level C Deck	32'	2
STR-203	Reactor Compartment – Upper Level B Deck	41'	2
STR-204	Reactor Compartment – Upper Level A Deck	50'	2
STR-205	Starboard Stabilizer Room	Hold	2
STR-206	Engine Room Machinery Space	Hold – Boat Deck	2
STR-207	Health Physics Lab	A	2
STR-208	Horseshoe area	Hold	2
STR-209	Hot Chemistry Lab	D	2
STR-210	Cargo Hold No. 4	Tanktop - D	2

Table 2-25 Initial Classification of Structures or Rooms (continued)

Survey Unit #	Survey Area Description	Deck or Elevation	Initial Classification
STR-301	Navigation Bridge Deck: interior surfaces Pilot House, Gyro. Room, Emergency Generator Room, Fan Rooms, Battery Rooms, Chart and Radio Rooms, Stairwells (ladders) down to Boat Deck	Navigation Bridge	3
STR-302	Navigation Bridge Deck: exterior surfaces, top of the house and radar mast	Navigation Bridge	3
STR-303	Boat Deck: interior surfaces Officers P/S Rooms, Officers Lounge, Fan Room, Passageways, Capt.'s Stateroom and Day Room, Chief Eng.'s Stateroom and Day Room, Stairwells (ladders) going down to Promenade Deck	Boat	3
STR-304	Boat Deck: exterior surfaces	Boat	3
STR-305	Promenade Deck: interior surfaces of Main Lounge, Veranda, Pantry, Novelty Shop, Men's Room, Powder Room, Card Room, Writing/Library Room, Projection Booth, Stairwells (ladders) going down to A Deck	Promenade	3
STR-306	Promenade Deck: exterior surfaces	Promenade	3
STR-307	A Deck: interior surfaces of Passengers Staterooms, Crew Spaces, Barber Shop, Hospital Spaces, Main Lobby, Pantry, Fan Rooms, Hydrogen Room, Passageways, Stairwell (ladders) going down to B Deck (See footnote at end of table.)	A	3
STR-308	A Deck Forward exterior surfaces forward of Frame 99 including Mast Houses, Masts, Deck Machinery and Cargo Gear.	A	3
STR-309	A Deck Aft exterior surfaces aft of Frame 168: including Docking Bridge, Masts, Deck Machinery and Cargo Gear.	A	3
STR-310	B Deck: interior surfaces of Crew Staterooms, Main Galley, Dining Room, Crew's Mess, Officers Mess, Officers Pantry, Crew Lounges, Crew's Pantry, Scullery, Passageways, Stairwells (ladders) going down to C Deck (See footnote at end of table.)	B	3

Table 2-25 Initial Classification of Structures or Rooms (continued)

Survey Unit #	Survey Area Description	Deck or Elevation	Initial Classification
STR-311	C Deck: interior surfaces of Main Laundry, Soiled Linen Room, Clean Linen Room, Food Freezers and Refrigerators, Butcher Shop, Crew Staterooms and Bathrooms, Respiratory Equipment Rooms, Viewing Gallery Area, Garbage Room, Nuclear Battery Room, Mechanical Space, Stewards Stores, Engine Room Stores, CO2 Room, Music Entertainment Equipment Room, Electric Work Shop, Carpenter Shop, Passageways (all), Stairwells (ladders) down to D Deck[10]	C	3
STR-312	D Deck: Electronics Work Shop, Bulk Stores Rooms, Special Store Rooms #1, 2, and 3, Food Freezers and Refrigerators, Machinery Space Equipment Room, Work shop Room, Engineering Stores Room, Spare Parts Storeroom, Passageways (all) (See footnote 10)	D	3
STR-313	Refrigerator rooms - Vegetables and Dairy (Port and Starboard, Passageways and Stairwells (ladders) going down to Horseshoe area and up to D-deck	14 foot flat	3
STR-314	Dry Stores (port) and Steward (starboard) outboard of horseshoe area	Tanktop	3
STR-315	Cargo Hold No. 3	C Deck	3
STR-316	Cargo Hold No. 4	C Deck	3
STR-317	Elevators and Elevator Shafts	All Decks	3
FSS-310A1 through FSS-318A1	Exterior surfaces of the Hull	All	3
STR-401	Port, Starboard, Fore Peak, and Aft Ballast Tanks		Non-impacted
STR-402	Shaft Alley and Shaft Alley Recess		Non-impacted
STR-403	Cargo Holds 1 and 2	All levels	Non-impacted
STR-404	Cargo Holds 5 through 7 (See footnote at end of table.)	All levels	Non-impacted
STR-405	Chain Locker and Windless Equipment Room	All levels	Non-impacted
STR-406	Steering Gear Room, Paint Locker, Bos'n Storerooms	All levels	Non-impacted

[10] These spaces have been converted to other uses since the Possession-only License - such as meeting rooms, office space, records storage, etc.

2.5 References

2-1 Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, Rev. 2, July 2019

2-2 NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, Rev. 2, April 2018

2-3 CR-003, *Historical Site Assessment*, Revision 1, August 2023

2-4 CR-104, *Radiological Characterization (Outside of Reactor Compartment and Containment Vessel)*, Revision 1, March 2019

2-5 CR-109, *Radiological Characterization - Reactor Compartment and Containment Vessel*, Revision 1, February 2020

2-6 *Updated Final Safety Analysis Report*, STS-004-002, Revision 13, July 20, 2023

2-7 NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, Revision 1, dated August 2000

2-8 STS-100, *Post Shutdown Decommissioning Activities Report*, Revision 1, December 11, 2008

2-9 CR-142, *Reactor Vessel, Internals and Neutron Shield Tank Characterization and Classification Assessment*, April 3, 2004

2-10 CR-038, *Radiological and Non-Radiological Spaces Characterization Survey Report*, Revision 1, February 2, 2006

2-11 CR-056, *Reactor Pressure Vessel Drilling, Sampling and Radiochemical Analysis Project Report*, Revision 1, January 31, 2006

2-12 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated October 3, 2008, - *Submittal of Finding of No Significant Impact and Environmental Assessment* (ML082810182)

2-13 CR-143, *MARSSIM Survey of the Exterior Hull*, December 2020

2-14 CR-144, *Primary Shield Water Tank Lead Sample Results,* April 2022

2-15 NUREG/CR-5512, Vol. 3, *Residual Radioactive Contamination from Decommissioning – Parameter Analysis*, October 1999

2-16 *Visual Sample Plan*, Pacific Northwest National Lab, USDOE

3 IDENTIFICATION of REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(B), the guidance of *Regulatory Guide 1.179, Standard Format and Content for License Termination Plans for Nuclear Power Reactors* [Reference 3-1] and the guidance in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, [Reference 3-2], the purpose of this chapter is to provide a summary of remaining site dismantlement, decontamination activities, and related decommissioning activities at the time the LTP is submitted.

The information includes those areas and equipment that need further remediation and an assessment of the potential radiological conditions that may be encountered. Estimates of the occupational radiation dose and the quantity of radioactive material to be released to unrestricted areas during the completion of the scheduled tasks are provided. The projected volumes of radioactive waste that will be generated are also included. These activities will be undertaken pursuant to the current 10 CFR Part 50 license, are consistent with the *Post Shutdown Decommissioning Activities Report* (PSDAR) [Reference 3-3] and do not depend upon LTP approval to proceed.

Reference 3-2 further requires identification of any decommissioning tasks that require coordination with other Federal or State regulatory agencies and an explanation of how that coordination will occur. It also requires a list of the remaining activities that do not involve unreviewed safety questions or changes in a facility's technical specifications.

3.1.1 Dismantlement Scope and Planned Final Ship Configuration

The scope of dismantlement described in the PSDAR is based on several fundamental assumptions, which are supported by the initial characterization efforts described in Chapter 2. Among the assumptions are:

a) The ship itself is not dismantled as part of DECON;

b) Existing accesses are utilized to support dismantlement of systems and components;[11] and,

c) Major structures will not be dismantled.

These assumptions are based, in part, on National Historic Preservation Act (NHPA) requirements and satisfactory Final Status Surveys (FSSs). Among the initial structures planned to be retained are the Containment Vessel (CV) and its foundation, and the Secondary Shield described in Chapter 5 of the *Updated Final Safety Analysis Report* (UFSAR) [Reference 3-4]. Primary ship structures will be decontaminated and remediated to the extent necessary to meet the license termination criteria. These include the decks and bulkheads which form the boundaries of radiologically controlled areas,

[11] In practice, two additional accesses were created and support dismantlement activities. These are the CV horizontal portal installed in 2018, and the roughly 6' x 8' opening between Cargo Hold 4 and the RCLL installed in 2021. The CV portal was intended primarily for personnel access and egress but has also been used for material handling. The reverse is true for the CH 4 – RCLL access. Both accesses will remain in place after LT.

and the contaminated liquid storage tanks[12] that are integral to the ship's double bottom hull structure.

Major dismantlement was focused on the removal and disposal of systems and components as described in Section 3.3 of the PSDAR. Characterization efforts, after submittal of the PSDAR, refined the plan outlined in the PSDAR and supported detailed dismantlement planning. The initial scope of dismantlement involved systems and components described in the UFSAR [Reference 3-4]. The initial scope included the following: a) Reactor Plant Auxiliary Systems described in Chapter 9, b) the Primary Coolant Systems and Components described in Chapter 7, and c) the Reactor Compartment cooling and exhaust systems described in Chapter 3. Components of the reactor core described in Chapter 6 that are within the Reactor Pressure Vessel were removed with the RPV itself.[13] The Power Conversion Systems described in Chapter 8 are excluded from the scope of dismantlement. Temporary systems, such as the contaminated liquid transfer and collection system (see 3.1.2), will be dismantled when no longer required. They will be disposed as Low Level Radioactive Waste (LLRW).

After the 2019-2020 characterization was complete, MARAD modified the dismantlement scope and specified the secondary side elements of the port and starboard steam generators for retention. The primary coolant tubes and tube sheets in the heat exchangers were removed, packaged, and disposed in 2022. The heat exchanger shells were decontaminated and remediated as required to meet the license termination criteria.

Towards the end of CY 2021, a major change in the industrial approach towards removal of the RPV was accepted by MARAD. In this approach, the common outer circumferential wall of the Neutron Shield Tank (NST) / Fuel Transfer Tank (FTT), including the exterior lead shielding surrounding the NST, was designated for retention. Two diametrically opposed access openings were created in the NST a) to permit cutting operations to free the RPV from its supports and b) to disconnect the inner and outer walls of the NST/FTT. The RPV, with the inner NST/FTT circumferential wall attached, was lifted clear of the remaining tank structure in November 2022. The change in approach was accepted after activation analyses and surveys of the exterior wall and lead shielding was performed, as described in Section 2.3.7. The primary purpose for the change was to significantly improve occupational safety by eliminating the hand-removal of the lead shielding. Subject to confirmation during FSS, MARAD expects the structure to meet the license termination criteria, and to be retained in-situ.

Finally, another change was made during CY 2022 related to the Pressurizer. Initially, the Pressurizer was to be removed intact as a heavy lift. Retaining the NST/FTT outer wall created clearance problems within the CV, such that intact removal of the Pressurizer was no longer considered practical. Instead, a large access was cut into one side of it, so that its internals could be removed. Initial decontamination using laser ablation significantly reduced interior contamination. Further remediation was performed by grinding the inner stainless-steel cladding. The lower hemispherical section of the Pressurizer containing heater penetrations could not be decontaminated and was disposed as LLRW. If decontamination and remediation efforts are confirmed during FSS, the remaining portions of the Pressurizer shell will be retained in-situ.

[12] This includes the Fresh Water Shield Tank (FWST). The FWST - a component of the Clean Ballast System and not the Equipment Drain and Waste System - was used to store primary coolant when the plant was mothballed in 1975-76. Consequently, the FWST contained contaminated liquid, and is subject to remediation and survey.

[13] In practice, because the RPV Head was removed and shipped as a component, the internal Belleville Spring and Upper Flow Baffle were removed and shipped separately because they protruded above the RPV flange. The flange surface was mated to a new combined lifting and cover plate; therefore, the two internals were removed as interferences.

At the end of the project, the final configuration of the ship's former RCAs is expected to be as follows:

- All RCAs (including LLRW storerooms, the material handling area in Cargo Hold 4, and controlled areas in Cargo Hold 3) will be de-posted and removed from radiological controls as part of the License Termination process. The end-state configuration of these spaces will be unrestricted release from radiological controls.
- The CV and Secondary Shielding will be intact and retained in-situ. The CV Cupola Head and Shield Ring will be reinstalled. Access (gratings, platforms and ladders) and lighting within the CV will be restored.
- The Reactor Compartment Hatch will be closed, and its rigging system will be removed and stowed in Cargo Hold 1. Once complete, the hatch will be inoperative without the use of an exterior crane.
- The DHVAC system described in Chapter 3 of Reference 3-4 will be decontaminated to the extent necessary and retained in operating condition.

3.1.2 Completed Dismantlement Activities

Major dismantlement activities were completed in the first half of CY 2023. Previously completed dismantlement activities have been summarized in each calendar year annual report since dismantlement was authorized by License Amendment 15 in 2018. During the project's first phase, reactor auxiliary systems and components in RCAs outside the reactor compartment were dismantled. The resultant LLRW was stored onboard the ship in areas designated for the purpose. That waste was designated as legacy waste for disposal during Phase II. It was disposed of in 2022.

Dismantlement activities during Phase II of the DECON-LT project focused on the three major spaces within the Reactor Compartment.[14] Minor component, small bore piping, electrical cabling, ventilation ductwork, and interference removals began in September 2021. Also, during 2021, a contaminated liquid transfer system, including a storage bladder, was installed and tested. During 2022, major component removals were completed, including the CRDM tower, port and starboard Steam Generator tubes and tube sheets, the RPV Head, two RPV internal components, and the RPV itself. During 2023, the lower section of the Pressurizer was removed and shipped as LLRW. During 2022 and 2023, reactor auxiliary system components, contaminated liquids and mixed waste were disposed.

During dismantlement and decontamination operations, MARAD concluded that there was potential preservation value to retaining the aft Reactor Coolant System (RCS) piping, and deferred shipping of this waste material until sampling and analysis was completed with a view towards decontamination of the pipe segments to levels below the DCGLs. The aft RCS piping segments were moved to a temporary RMA created in Cargo Hold 3 pending evaluation. Metallurgical samples were shipped in the third quarter of 2023. At the time of LTP submittal, MARAD has not yet decided the fate of the pipe segments. If disposed, these segments will be among the last LLRW shipments.

All dismantlement was essentially complete at the time of LTP submittal.

[14] The three spaces are the Containment Vessel, and the Reactor Compartment Upper and Lower Levels that surround the CV.

3.1.3 Coordination of Activities and Unreviewed Safety Questions

There are no remaining decommissioning activities which require specific coordination with other Federal or State agencies. MARAD has coordinated prior completed activities with such agencies, including the transit of the RPV through the Howard Street Tunnel beneath Baltimore City. MARAD maintains contacts with Federal, State, and local Baltimore City agencies as a matter of routine. Periodic updates will be provided as the project progresses to license termination.

Decommissioning activities at NSS will continue to be conducted in accordance with the requirements of 10 CFR 50.82(a)(6) and (a)(7). At the time of LTP submittal, the remaining activities do not involve any unreviewed safety questions or changes in the Technical Specifications. If an activity requires prior NRC approval under 10 CFR 50.59, or a change to the technical specifications or license, a License Amendment Request will be made to the NRC for review and approval before implementing the activity in question.

3.1.4 Proposed Methods to Prevent Recontamination

The methods to prevent recontamination (i.e., isolation and control) are described in Section 5.4.4.

Figure 3-1 presents the current schedule for completing the remaining License Termination activities.

ID	Task Name	Duration	Start	Finish
1	NS SAVANNAH DECOMMISSIONING PROJECT REMAINING ACTIVITIES	1155 days	Apr. 2021 - Nov. 2025	
2	**DECON-LT Support Contract.**	**1260 days**	**Apr. 2021 - Mar. 2026**	
3	**Phase II - Industrial Dismantlement Remaining Activities**	**561 days**	**01/01/2022**	**03/31/2024**
4	Package, ship and dispose of radioactive waste materials.	561 days	01/01/2022	03/31/2024
5	**Phase III - License Termination**	**1155 days**	**04/01/2021**	**11/10/2025**
6	Prepare and submit LTP.	642 days	04/01/2021	10/23/2023
7	NRC review and approval of the LTP.	256 days	10/30/2023	11/5/2024
8	Prepare, perform and submit Final Status Survey (FSS) packages.	162 days	03/27/2024	11/15/2024
9	NRC confirmatory surveys and review of FSS packages.	384 days	04/30/2024	11/10/2025
10	MARAD submission of application for license termination to NRC.	1 day	11/10/2025	11/10/2025

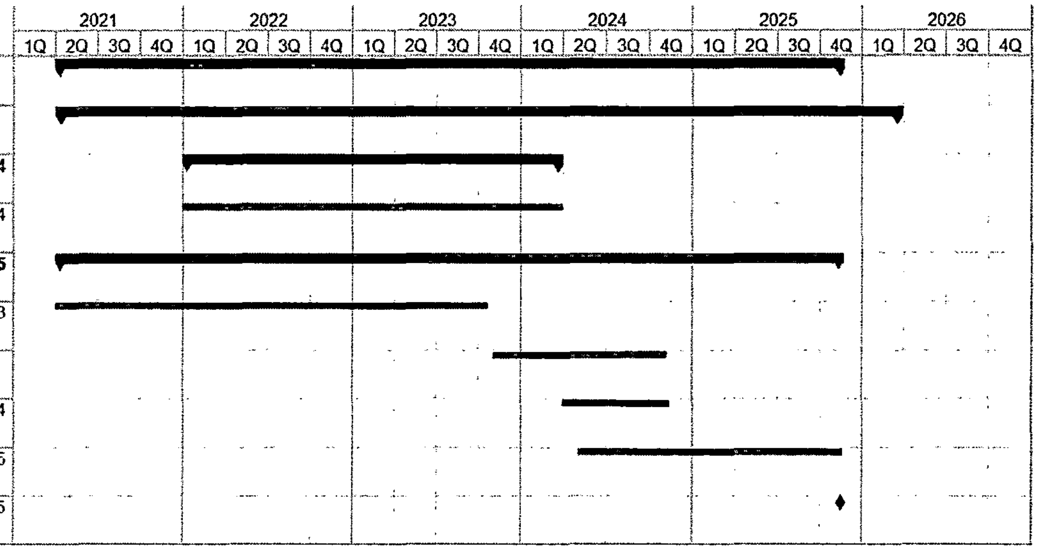

Figure 3-1 Remaining License Termination Activities

3.2 Remaining Activities

At the time of LTP submittal, all dismantlement was essentially complete. Shipment of LLRW will continue until complete.

Decontamination and remediation of structures will be ongoing and will include those structures listed in Section 3.1.1. These activities will continue, as required, during the FSS period. As described in Section 3.1.1, certain structures (NST/FTT outer wall and lead shielding, Pressurizer shell; CV shell; and CV upper hemisphere secondary shielding) are planned for retention subject to confirmation that they meet the license termination criteria. If these structures fail to meet the criteria of the approved LTP, they will be dismantled and shipped as LLRW. This would require a period of remobilization by MARAD's decommissioning contractor; consequently, no changes to the current RCA boundaries for personnel entry and exit, monitoring stations, dosimetry issue and material handling arrangements will be made until the status of the retained structures and components is confirmed by NRC. If required, a revision to the LTP will be submitted by MARAD.

3.3 Waste Projections

Table 3-1 provides a summary of remaining waste quantities. Table 3-2 provides a summary of the waste shipments through August 31, 2023. MARAD decided not to process waste onsite or to attempt to segregate waste streams. The total volume of low-level radioactive waste for disposal was originally estimated at 22,844 cubic feet. The solid waste was and future solid waste will be shipped to the licensed EnergySolutions radioactive waste disposal facility (ES) in Clive, Utah. Liquid waste and future liquid waste will be shipped to ES, Erwin, TN.

Table 3-1 Projected Remaining Waste Quantities as of September 30, 2023

Waste Type	Waste Class	Waste Volume (ft^3)
LLRW – metal 2 IMs standard size components	A	2025
Chromate/Lead Trash (bags)	Mixed	260
Dry Active Waste / Metal (bags)	A	25
5 Drums 250 gallons of water	Mixed	34
Radium 226 Sources (steering gear)	A	1.00
1 Drum 4 gallons of oil	Mixed	0.53
Totals		**2345.53**

Table 3-2 Summary of Waste Disposed from Ship through September 30, 2023

Container	Unit Quantity	Liquid Content	Solid Content	Total Volume	Generated	Hazard Classification
Intermodal	24		Dry Active Waste / Metal	16200 ft^3	Phase II	Radioactive
Intermodal	5		Dry Active Waste / Metal	3375 ft^3	Phase I	Radioactive
Intermodal	3		Lead (Pb)	21200 lbs.	Phase I/II	Recycle
Intermodal	2		Water/Chromate /Lead (Pb)	184 ft^3	Phase II	Mixed
B-12	2		Lead (Pb)	90 ft^3	Phase II	Mixed
Sealand	1		Reactor Internals	312 ft^3	Phase II	Radioactive
Reactor Head	1		Reactor Head	168 ft^3	Phase II	Radioactive
Reactor Pressure Vessel	1		RPV	2424 ft^3	Phase II	Radioactive
Control Rod Drive Tower	1		Metal	1600 ft^3	Phase II	Radioactive
Tanker	4	Water		18124 gal.	Phase II	Radioactive
Control Rod Drive Oil Drum	2	Oil	Personal Protective Equipment / Lead (Pb) fines	70 gal. and 280 ft^3	Phase II	EPA Code D007 and D008

3.4 Occupational Exposure

Table 3-3 provides the actual project dose as of September 30, 2023 and the dose estimate to complete the remaining activities. In April 2022, the total radiation exposure to complete decommissioning activities was estimated to be approximately 2.258 person-rem.

Table 3-3 Radiation Exposure - Project Total and Estimate to Complete Remaining Activities

Activity	Exposure (person-rem)
Project Dose as of September 30, 2023 (approximate)	2.168
Decontamination of rooms/areas	0.010
Total	**2.178**

No quantity of radioactive material will be released to unrestricted areas during the completion of the scheduled remaining tasks.

3.5 Major Project Milestones

Table 3-3 presents the general project milestones. MARAD recognizes that circumstances can change during decommissioning. If necessary, MARAD will provide an updated schedule to the NRC.

Table 3-4 General Project Milestones

Date	Milestone
4th Quarter 2023	Submit LTP to NRC for review
4th Quarter 2023 Through 1st Quarter 2024	Package, ship, and dispose of radioactive waste materials
1st through 4th Quarter 2024	Submit Final Status Surveys
2nd Quarter 2024 through 4th Quarter 2025	NRC Confirmatory Surveys
4th Quarter 2024	NRC approval of LTP
4th Quarter 2025	License terminated

3.6 References

3-1 Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, Rev. 2, July 2019

3-2 NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, Rev. 2, April 2018

3-3 STS-100, *Post Shutdown Decommissioning Activities Report*, Revision 1, December 11, 2008

3-4 *Updated Final Safety Analysis Report*, STS-004-002, Revision 13, July 20, 2023

4 REMEDIATION PLANS

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(C), the guidance of Regulatory Guide 1.179, Standard Format and Content for License Termination Plans for Nuclear Power Reactors [Reference 4-1] and the guidance in NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans, [Reference 4-2], this chapter describes plans for site remediation and discusses how residual radioactivity on the site will meet the As Low As Reasonably Achievable (ALARA) criterion for unrestricted release.

References 4-1 and 4-2 describe the content required for this chapter. Summarizing from these references, the License Termination Plan (LTP) should include the following items, which are not meant to be all-inclusive:

- The LTP discusses in detail how facility and site areas will be remediated to meet the NRC criteria for license termination in Subpart E of 10 CFR Part 20, including the proposed residual radioactivity levels [Derived Concentration Guideline Levels] (DCGLs). Discussions should focus on any unique techniques or procedures used to evaluate whether the DCGLs have been met including the following:
    - Summarize the techniques that will be used to remediate building structures and components (e.g., scabbling, hydrolazing, grit blasting, etc.).
    - Summarize the equipment that will be decontaminated and how the decontamination will be accomplished.
    - Summarize the radiation protection methods and control procedures that will be employed including a summary of the procedures already authorized under the existing license and any changes in the radiological controls to be implemented to control radiological contamination associated with the remaining decommissioning and remediation activities.
    - Commit to conduct decommissioning activities in accordance with approved written procedures.
    - Include a detailed description of the techniques that will be employed to remove or remediate surface and subsurface soils, groundwater, and surface water and sediments.
    - Describe plans, if any, for onsite disposal of decommissioning waste.
- The LTP includes a schedule that demonstrates how and in what time frames the licensee will complete the interrelated decommissioning activities. 10 CFR 50.82(a)(3) requires completion of decommissioning within 60 years. If the completion of decommissioning is delayed for more than 60 years, the LTP, must include a justification for the delay in accordance with 10 CFR 50.82(a)(3).

As described in sections 2.1.4 and 5.2 of this LTP, the NSS site contains no surface or subsurface soils, groundwater, or surface water features, consequently, there is no detailed description in the LTP of any techniques that would normally be employed to remove or remediate them.

There are no plans for onsite disposal of decommissioning waste.

Generically, an environmental remediation plan is a summary of planned remedies that will be used to achieve cleanup goals found on a property during a site characterization. To achieve license termination, 10 CFR 50.82(a)(9)(ii)(C) requires the LTP provide the "plans for site remediation." NRC focuses these plans to include the provisions to meet the criteria from Subpart E of 10 CFR 20 before the site may be released for unrestricted use. The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 [Reference 4-3] are:

1) The Total Effective Dose Equivalent (TEDE) from residual radioactivity that is distinguishable from background radiation must not be greater than 25 mrem/year to the AMCG; and,

2) Residual radioactivity levels must be ALARA.

Please note that as footnoted in Chapter 1 and described in Section 6.2 of this LTP, the TEDE selected on NSS is 15 mrem/year. Wherever a paragraph cites the 25 mrem/year standard, reviewers should interpret this to mean 15 mrem/year for NSS. For the purposes of the LTP, the Remediation Plan is a list of techniques, methods and technologies that will be used to meet 1) and 2) above. The effectiveness of these techniques will be confirmed by implementing the Final Status Survey (FSS) Plan described in Chapter 5.

Decontamination and dismantlement activities are conducted in accordance with NSS administrative programs and procedures. These programs and procedures are frequently assessed for technical content and compliance. Revisions have been, and will continue to be made, to these programs and procedures to accommodate the changing work environment inherent to reactor decommissioning. The revisions will continue to be documented, processed, and approved in accordance with the existing NSS License, Technical Specifications, and Administrative Procedures. Consistent with Regulatory Guide 1.179 [Reference 4-2], details regarding changes to the Radiation Protection Program (RPP) to address remediation and decommissioning activities are not provided in this LTP. Changes to the RPP will be provided in either 1) Annual Reports as required by the Technical Specifications or 2) in periodic updates to the UFSAR or LTP.

This chapter describes two key items:

- The methods that may be used to remediate contaminated systems, components, and structure surfaces; and,
- The methods to demonstrate compliance with the ALARA criterion in 10 CFR 20.1402 per the guidance in NUREG-1757 [Reference 4-4].

As described in Chapter 6, MARAD has adopted a dose scenario, with associated parameters and assumptions, based on a 15 mrem/year release limit. Also note that Chapter 3 describes in detail the remaining site remediation and dismantlement activities and the order in which they will occur for each structure, system and/or component. It also notes that MARAD recognizes circumstances can change during decommissioning. The current schedule for completing the remaining License Termination Activities is shown in Chapter 3, Figure 3-1. MARAD expects to complete decommissioning and request termination of its license well before the 60-year period prescribed by 10 CFR 50.82(a)(3). For reference, the NSS deadline is December 3, 2031.

Section 4.3 provides a summary of the radiation protection methods and control procedures that will be employed to address the impact of dismantlement and remediation activities.

4.1 Remediation Actions and ALARA Evaluations

When dismantlement and decontamination actions are completed, residual radioactivity may remain on the ship surfaces at concentrations that correspond to the NSS dose criterion of 15 mrem/year. The remaining residual radioactivity must also satisfy the ALARA criterion, which requires an evaluation as to whether it is feasible to further reduce residual radioactivity to levels below those necessary to meet the dose criterion (i.e., to levels that are ALARA). See Section 4.4 of this chapter for a discussion of ALARA.

4.2 Remediation Actions

Remediation actions are performed throughout the decommissioning process and are based on the results of radiological surveys. For example, a survey may be performed in an area where prior to a large component removal, no survey was possible and the collected survey data indicates that remediation is required. When decommissioning activities in a survey unit are complete, characterization surveys are conducted when additional information is needed for the proper planning of the survey unit's FSS. A turnover survey will be performed in preparation for FSS activities when a review of previous survey data for the survey unit cannot confirm a) there is sufficient data to perform FSS design and b) an FSS in the survey unit is likely to meet release criteria. If the characterization survey or turnover survey determines that additional remediation is necessary, then the completed remediation activities will be followed by a Remedial Action Support Survey to verify that a survey unit is likely to meet release criteria following the performance of an FSS.

These remediation techniques, methods, and technologies are standard to the commercial nuclear industry. They represent the current best practice methods and use a consistent approach intended to facilitate the most cost-effective balance between hazardous waste removal and decontamination cleaning. All remediation actions described may not necessarily be required but are listed as possible actions that may be taken during the decommissioning of the NSS. The appropriate remediation technique(s), method(s) and/or technologies that will be employed are dependent on the physical composition and configuration of the contaminated media requiring remediation. At the NSS, the principal media that will be subjected to remediation are structural surfaces. Characterization survey results and historical survey data indicate that generally there is minimal contamination identified to date.

4.2.1 Pressure Washing

Pressure washing uses a nozzle of intermediate water pressure to direct a jet of pressurized water that removes superficial materials from the suspect surface. A header may be used to minimize overspray. A wet vacuum system is used to suction the potentially contaminated water into containers for filtration or processing.

4.2.2 Needle Guns

A method of scabbling is accomplished using needle guns. The needle gun is a pneumatic air-operated tool containing a series of tungsten-carbide or hardened steel rods enclosed in housing. The rods are connected to an air-driven piston to abrade and fracture the media surface. The media removal depth is a function of the residence time of the rods over the surface. Typically, one to two millimeters are removed per pass. Generated debris collection, transport, and dust control are accomplished in the same manner as other scabbling methods. Use of needle guns for removal and chipping of media is usually reserved for areas not accessible to normal scabbling operations. These include, but are not limited to, inside corners, cracks, joints, and crevices. Needle gunning techniques can also be applied to painted and oxidized surfaces.

4.2.3 High Pressure Water Blasting

Most contaminated piping will be removed and disposed of as radioactive waste. If radiological conditions inside the pipe are in excess of the release criteria and the system is to remain in place, then *in situ* remediation will be performed. One method that may be used to remediate the pipe interior surfaces is high pressure water blasting. A typical High-Pressure Liquid-Jetting System has a high-pressure water pump capable of producing a water pressure of 10,000 psi to 20,000 psi at an actual flow rate that ranges from 44 gallons per minute at 10,000 psi to 23 gallons per minute at 20,000 psi. A rotating jet-mole tip is used for 360-degree coverage of pipe interiors. The jet mole is attached to a lance and high-pressure hose. The lance is manually advanced though the interior of the pipe. As the lance is advanced, the high-pressure water abrades the interior surface of the pipe to remove the corrosive layer, internal debris, and radiological contamination. The wastewater containing the removed contamination is then collected and stored for processing as liquid radiological waste.

4.2.4 Laser Ablation

Laser Ablation is a process utilizing a very controlled Laser System, where pulses of light energy are used to dislodge surface contamination such as oxides and coatings from the surface of a substrate. Up to 90% of the dislodged material is instantly turned into vapor by the plasma of the laser. Vaporized residues are collected immediately after ablation by a strong point-source vacuum system that captures process residues. They pass through multi-stage filtering to be scrubbed free of particles and vapors, preventing hazardous airborne contaminants and minimizing clean-up. The particulates are deposited in a canister for disposal. The integral vacuum system can make a two-step process of cleaning and vacuuming in a single step. Waste reduction can be up to 98% of a typical sand blasting operation. The surface of the substrate after laser ablation leaves a clean uniform profile. This system has been used to decontaminate some of the inside surfaces of the Pressurizer.

4.2.5 Chemical Strippers

Chemical stripping is a physical separation process where one or more components are removed from a surface. Chemical strippers may be used, as appropriate, for the removal of certain contaminants in small areas, after evaluation and approval from NSS management.

4.2.6 Grinding

Grinding is a type of abrasive machining process which uses a grinding wheel as a cutting tool. Portable tools such as angle grinders are employed and use a rotating abrasive wheel to remove material from surfaces, cracks, or corners. The outcome is similar to needle gunning.

4.2.7 Sponge and Abrasive Blasting

Sponge and abrasive blasting are similar techniques that use media or materials coated with abrasive compounds such as silica sands, garnet, aluminum oxide, and walnut hulls. Sponge blasting is less aggressive, incorporating a foam media that, upon impact and compression, absorbs contaminants. The medium is collected by vacuum and the contaminants are washed from the medium so the medium may be reused. Abrasive blasting is more aggressive than sponge blasting but less aggressive than scabbling. Both operations use intermediate air pressures. Sponge and abrasive blasting are intended for the removal of surface films and paints.

In addition to the above techniques, MARAD is considering a combination of chemical decontamination and ultrasonic cleaning for select sections of reactor coolant system piping, with an objective of preserving sections of the aft Reactor Coolant System piping.

4.3 Remediation Activities Impact on the Radiation Protection Program

The RPP approved for decommissioning at the NSS is similar to the one that was implemented during its commercial operation. That program was scaled back after NSS was mothballed and the Possession-only License was issued. During the museum period, the state of South Carolina employed its own radiological protection processes and procedures on NSS. After NSS was returned to sole MARAD custody, MARAD employed the Army Corps of Engineers' STURGIS radiological monitoring and surveillance program on NSS. This lasted until 2005, when MARAD began developing renewed administrative programs and procedures in anticipation of decommissioning. Throughout this period of the Possession-only License and until License Amendment 15 was issued in 2018 to authorize dismantlement and disposal of waste, radiological activity was limited to performing only routine radiological activities such as surveys and confirmation of correct postings. No remediation was performed prior to Amendment 15 because the Technical Specifications allowed no activity that was likely to generate radiological waste.

The current approved RPP at NSS, developed after 2005 through to the present, complies with all federal and state regulatory requirements for the protection of occupational personnel from radiological hazards encountered or expected to be encountered during the decommissioning. In addition, the program ensures the protection of the public from radiological hazards and ensures occupational, effluent, and environmental dose from exposure to radioactive materials is and remains ALARA. To ensure that adequate and proper engineering controls and hazard mitigation techniques are employed, work control programs and procedural requirements allow radiation protection personnel to integrate radiation protection and radiological hazard mitigation measures directly into the work planning and scheduling process. Consequently, the necessary radiological controls are correctly implemented to accommodate each remediation technology as appropriate. Remedial action support survey design is described in Sections 5.3 and 5.4.

The spread of loose surface contamination is mitigated by the routine remediation of work areas by washing and wiping. Water washing with a detergent is effective in reducing low levels of loose surface contamination over large surface areas. Wiping with detergent-soaked or oil-impregnated media is an effective technique to reduce loose surface contamination on small items, overhead spaces and small hand tools. These activities are referred to as "good housekeeping." These same techniques are also effective in reducing low levels of surface contamination on structural surfaces.

More aggressive methods for intermediate levels of surface contamination may be appropriate, such as:

- Pressure washing;
- High-pressure water blasting;
- Grinding; and,
- Sponge and Abrasive Blasting.

Pipes, surfaces and drain lines can be cleaned and hot spots removed using these techniques and technologies. Small tools, hoses, and cables can also be pressure washed in a containment to reduce contamination levels.

The RPP implements radiological controls 1) to reduce personnel exposure to radiation and contamination and 2) to prevent the spread of contamination from established contaminated areas. Decommissioning does not present any new challenges to the RPP. There are no known unique safety issues associated with remediating any contaminated structures. Decommissioning planning allows radiation protection personnel to focus on each area of the site and plan each activity well before execution of the remediation technique. The decommissioning organization is staffed by contractors experienced in and capable of applying these remediation techniques on contaminated systems, structures or components during decommissioning.

A comprehensive review of all radiation procedures was performed in 2020, after which numerous procedures were revised to make them consistent with current industry practice and with other NSS administrative procedures. These procedures were approved in accordance with the Technical Specifications, Decommissioning Quality Assurance Plan (DQAP), and NSS administrative processes.

4.4 ALARA Evaluation

Guidance for conducting ALARA analyses is provided in NUREG-1757 Vol. 2 [Reference 4-4], which describes acceptable methods for determining when further reduction of residual radioactivity is required to concentrations below the levels necessary to satisfy the 25 mrem/yr. dose criterion.

In Chapter 6 of NUREG-1757, there is a section identified as "Predetermined Compliance Measure." The following paragraph is from this section.

> *Under the predetermined compliance measure, the licensee would agree to meet the dose calculated for the preferred option or the radiological concentrations associated with this dose. This could be met by either establishing deterministic concentration limits for the site or agreeing to use a specified dose scenario with associated parameters and assumptions. If the licensee's final survey results meet the self-imposed concentration limits (or dose limit), the licensee has met the ALARA requirement.*

As described in Section 3.1.1 of Chapter 3, several components will remain in-situ at the end of dismantlement and will be retained - subject to satisfactory FSSs. These components are located within the Containment Vessel (CV) and include the secondary sides of both steam generator assemblies, the upper section of the pressurizer shell (internals removed and interior surfaces decontaminated), and the common exterior annular wall of the Neutron Shield Tank / Fuel Transfer Tank. The structure of the CV itself will also remain. MARAD is confident that these components and structures will meet the DCGLs proposed in Chapter 6. The steam generator shells had no detectable contamination during the 2019 characterization campaign. Initial internal characterization data for many of the systems remaining after decommissioning is completed was obtained in 2019. The results of those surveys showed that the internal surfaces of those systems have low loose surface contamination levels and total surface contamination levels below the DCGLs proposed in Chapter 6 of this LTP. All reasonable efforts will be made to locate and remove any loose contamination on external surfaces of the ship. Based upon the short operating history, long decay time, and unique design of the NSS, routine surveys of the ship indicate very low levels of contamination in the CV and Reactor Compartment.

The dose limit for FSSs and License Termination is specified in Chapter 6 of the LTP and is equal to 15 mrem/year. Chapter 6 also presents the radiological concentrations associated with this dose. It is anticipated that the FSS of components and structures will show contamination levels well below the 15 mrem/year DCGLs. The goal of the FSS is to meet the self-imposed concentration limits. As described in Chapter 6, MARAD has adopted a dose scenario, with associated parameters and assumptions, based on a 15 mrem/year release limit.

When FSS is complete and the dose scenario is confirmed met, MARAD will have met the ALARA requirement, and no further analyses are required.

4.5 Summary of Remediation Techniques, Procedure and Issues

NUREG-1700, *Standard Review Plan for License Termination Plan* [Reference 4-2] and its Appendix A list a number of acceptance criteria associated with remediation that "... focus on any unique techniques or procedures used to evaluate whether the DCGLs have been met ..."

MARAD has conducted remediation activities as described in Section 3.1. Thus far, MARAD has identified no unique techniques or procedures that will be used to evaluate whether the DCGLs have been met. MARAD has identified no unique safety issues and no remediation issues associated with remediating contaminated SSCs.

4.6 References

4-1 Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, Rev. 2, July 2019

4-2 NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, Rev. 2, April 2018

4-3 Code of Federal Regulations, Title 10, Part 20.1402, *Radiological Criteria for Unrestricted Use.*

4-4 NUREG-1757, *Consolidated Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria*, September 2006

5 FINAL STATUS SURVEY PLAN

5.1 Introduction

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(D), the guidance of *Regulatory Guide 1.179, Standard Format and Content for License Termination Plans for Nuclear Power Reactors* [Reference 5-1] and the guidance in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, [Reference 5-2], this chapter describes the methods to be used in planning, designing, conducting, and evaluating Final Status Surveys (FSSs) at the NSS. These surveys will demonstrate that the dose from residual radioactivity is less than the maximum dose criterion for license termination for unrestricted use specified in 10 CFR 20.1402 [Reference 5-3]. As described in Chapter 6 and elsewhere in this LTP, MARAD has committed to a lower dose criterion of 15 mrem/year. The additional requirement of 10 CFR 20.1402 that all residual radioactivity at the site be reduced to levels that are As Low As Reasonably Achievable (ALARA) is addressed in Chapter 4.

References 5-1 and 5-2 describe the content required for this chapter. Summarizing from these references, the LTP should include the following items, which are not meant to be all-inclusive:

- Describe the methods proposed for surveying all equipment, systems and structures, as well as a method for ensuring that sufficient data are included for a meaningful statistical survey. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.
- Describe the methods the licensee will use to establish background radiation levels. Discuss variances in background radiation that can be expected (e.g., between structures constructed of different materials), as outlined in draft NUREG-1501, "Background as a Residual Radioactivity Criterion for Decommissioning," issued August 1994 (Ref. 14).
- Describe the QA program to support both field survey work and laboratory analysis. Address the QA organization; training and qualification requirements; survey instructions and procedures, including water, air, and soil sampling procedures; document control; control of purchased items; inspections; control of survey equipment; handling, storage, and response [(i.e., operational)] checks; shipping of survey equipment and laboratory samples; disposition of nonconformance items; corrective action; QA records; and survey audits, including methods to be used for reviewing, analyzing, and auditing data.
- Describe the verification surveys and evaluations used to support the delineation of radiologically affected (contaminated) areas and unaffected (uncontaminated) areas.
- Identify the major radiological contaminants.
- Discuss methods used for addressing hard-to-detect radionuclides.
- Describe access control procedures to avoid recontamination of clean areas.
- Identify survey units having the same area classification.
- Describe scanning performed to locate small areas of elevated concentrations of residual radioactivity.
- Discuss levels established for investigating significantly elevated concentrations of residual radioactivity. Include survey instrument calibration and efficiency calculations.
- Describe the reference coordinate system established for the site areas.

The FSS Plan was developed using the guidance of NUREG-1575, *The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* [Reference 5-4]; and NUREG-1757; *Consolidated*

Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria [Reference 5-5]. The process described in this plan adheres to the guidance of MARSSIM for the design of FSS. MARSSIM allows for advanced survey technologies that can effectively scan 100% of the surface and record the results. MARAD does not expect to use any advanced survey technologies.

5.2 Scope

The FSS Plan encompasses the radiological assessment of all affected structures and systems for the purpose of quantifying the concentration of any residual activity that remains following all decontamination activities. Concentration limits have been established to represent the maximum dose rate criterion for unrestricted release specified in 10 CFR 20.1402 and are presented in Chapter 6 of the LTP.

The NSS site possesses unique features that both simplify, and perhaps complicate, the FSS process. The features and their characteristics, follow:

- The licensed site of the NSS is the boundary defined by the ship's hull (see Figures 1-1, 1-2 and 1-3). The hull is waterborne, and its former sea connections are fitted with welded steel blanks which prevent any auxiliary or secondary system from discharging overboard. So long as the primary hull structure, including the blanks, is intact, the site (i.e., the ship) is isolated from the marine and terrestrial environments.
- Being waterborne and isolated from the terrestrial environment, the site contains no soil and no ground water. Surveys of such features are not applicable to NSS. There is also no surface water feature such as may be present on a land-based site.
- Being isolated from the marine and terrestrial environments, the background radiation within the NSS site is lower than typical sites.
- The site is mobile, and thus is not permanently fixed to any particular location. The ship is periodically removed from any long-term berthing site for routine maintenance of the hull (e.g., drydocking or shipyard repairs such as occurred following museum service in 1994, after removal from layup in the MARAD James River Reserve Fleet during the period 2006-2008, and the temporary removal from the ship's current Baltimore layberth site for drydocking in 2019-2020). The future site of dispositioning activities (preservation, shipbreaking or artificial reefing) is unknown at the time of LTP submittal.
- The nature of the ship's construction means that the site contains no embedded pipe, as the term is commonly understood. Piping penetrations through structural steel bulkheads are typically less than one (1) inch in length, and can readily be surveyed, decontaminated, or removed. The longest lengths of piping penetrations are those which pass through the concrete secondary shielding. Here, the maximum length of pipe is about four (4) feet, and again, such piping is accessible from at least one side. Similarly, there will be no rubble backfill, structures remaining below grade, and no buried or surface paved parking lots, such as might be found at typical decommissioning sites.

The FSS Plan developed for NSS accounts for each of these characteristics and features.

5.2.1 Final Status Survey Organization

As described in the Section 2, Organization and Responsibilities of the Decommissioning Quality Assurance Plan (DQAP) [Reference 5-6], the Savannah Technical Staff (STS) organizational unit is MARAD's licensee organization and is primarily made up of a combination of direct employees and contractors. The organization is structured on the basis that the objectives of the DQAP will be met by those who manage, perform and support the activities within the scope of this plan. Those who perform work on quality related functions are directly responsible for meeting quality standards and reporting nonconformances or conditions adverse to quality. The requirements stipulated in the DQAP shall be imposed on all personnel and organizations, including contractors, who perform quality related decommissioning activities.

With submittal of the LTP, the function of FSS is an activity added to the scope of the DQAP. The major changes are adding an LTP Manager and an FSS Manager to the existing organization. The LTP Manager is responsible for overall FSS project coordination and direction as it relates to Phase 3 Decommissioning and License Termination. The FSS Manager is responsible for overall management and onsite implementation of the FSS Project. The existing STS organization supports FSS activities. As licensee, MARAD is responsible, with support from STS, for performing assessments of the implementation of the FSS program, including procedure adherence and conformance reviews of selected FSS reports. Figure 5-1 is the organizational chart of the FSS organization.

5.2.2 Final Status Survey Administrative Procedures

Procedure STS-005-029, *Final Status Survey Program* establishes the requirements for activities performed in support of FSSs. It includes a description of key FSS positions and their responsibilities. Other FSS administrative procedures include but are not limited to the following:

- STS-005-030, Preparation of FSS Packages
- STS-005-031, Calculation of the Number of Measurements
- STS-005-032, Survey Unit Turnover and Control
- STS-005-033, Final Status Survey Data Assessment
- STS-005-034, Survey Unit Classification
- STS-005-035, Preparation of Survey Unit Release Records

5.3 Summary of the Final Status Survey Process

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

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

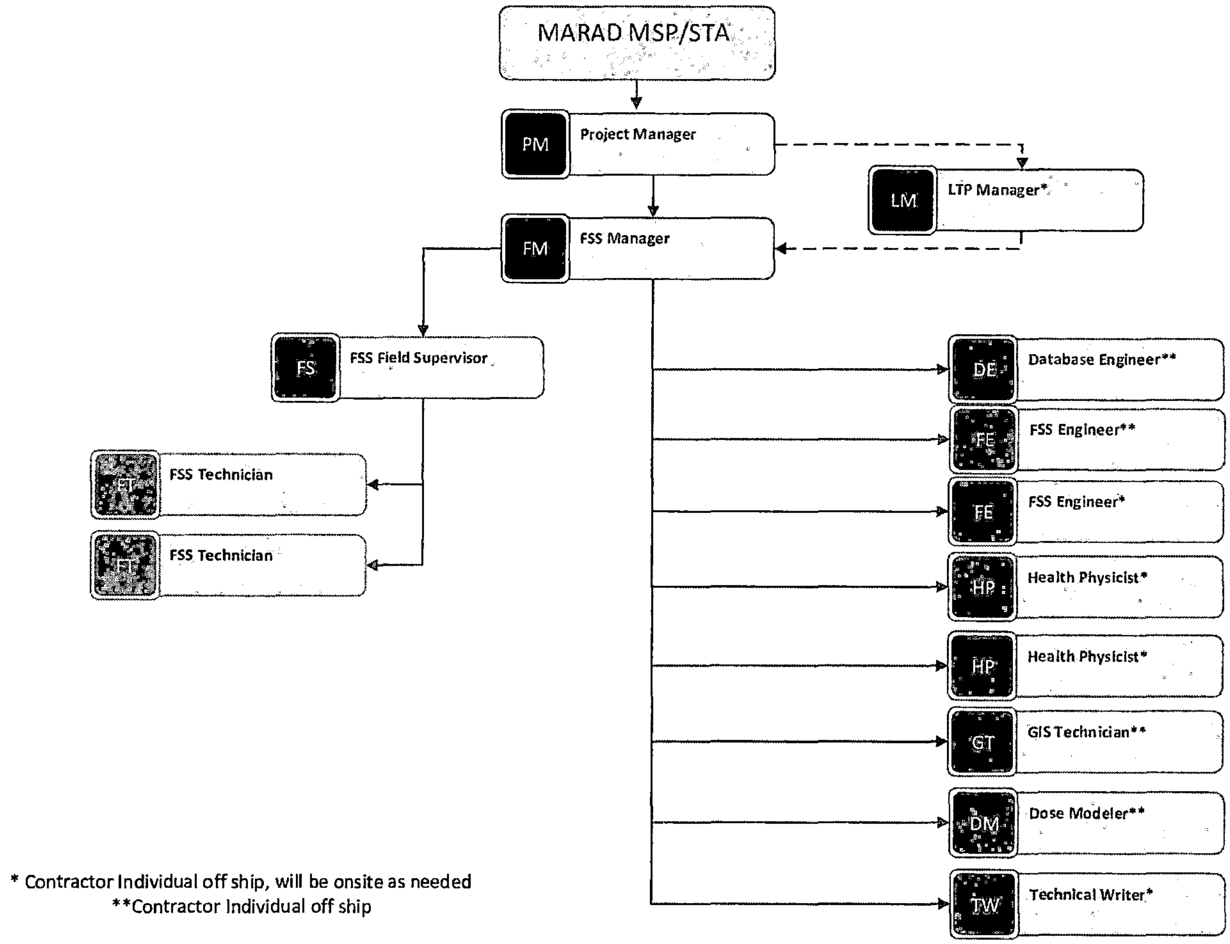

Figure 5-1 FSS Organization

The FSS process consists of four principal elements:

- Planning;
- Design;
- Implementation; and,
- Assessment.

The DQOs and Data Quality Assessment (DQA) processes are applied to these four principal elements. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions, as is the case in FSS. The DQA process is an evaluation method used during the assessment phase of FSS to ensure the validity of survey results and demonstrate achievement of the sampling plan objectives (e.g., to demonstrate compliance with the release criteria in a survey unit).

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

Before the survey process can proceed to the design phase, concentration levels representing the maximum dose criterion of 10 CFR 20.1402 must be established. As described in Chapter 6, MARAD has committed to a lower dose criterion of 15 mrem/year. These concentrations are established for surface contamination. These surface contamination concentrations are used in the survey design process to establish the minimum sensitivities required for the survey instruments and techniques, and in some cases, the spacing of fixed measurements to be made within a survey unit. Surface concentrations that correspond to the maximum dose criterion are referred to as Derived Concentration Guideline Levels (DCGLs). A DCGL established for the average member of the critical group exposed to residual radioactivity in a survey unit is called a $DCGL_W$.[15]

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

Survey implementation is the process of performing the survey plan for a given survey unit. This consists of scan measurements and fixed measurements. Data will be stored and controlled.

The DQA approach is applied to FSS results to ensure their validity and to demonstrate that the objectives of the FSS are met. Data assessment includes data Verification and Validation (V&V), review of survey design bases, and data analysis. For a given survey unit, the survey data are evaluated to determine if the residual activity levels in the unit meet the applicable release criterion and if any areas of elevated activity

[15] The "W" in $DCGL_W$ stands for Wilcoxon Rank Sum test, which is the statistical test recommended in MARSSIM for demonstrating compliance when the contaminant is present in background. The Sign test recommended for demonstrating compliance when the contaminant is not present in background also uses the $DCGL_W$.

exist. In some cases, the data evaluation will show that all measurements made in a given survey unit were below the applicable $DCGL_W$. If so, demonstrating compliance with the release criterion is a simple matter and requires little in the way of analysis. In other cases, residual radioactivity may exist where measurement results both above and below the $DCGL_W$ are observed. In these cases, statistical tests must be performed to determine if the survey unit meets the release criterion. The statistical tests that may be required to make decisions regarding the residual activity levels in a survey unit relative to the applicable $DCGL_W$ must be considered in the survey design to ensure that a sufficient number of measurements are collected.

MARSSIM specifies two (2) non-parametric statistical tests to be applied to FSS data to evaluate whether a set of measurements demonstrates compliance with the release criterion for a given survey unit. Those are the 1) Wilcoxon Rank Sum Test and 2) Sign Test. On NSS, FSS data will only be evaluated with the Sign Test because the radionuclide(s) of interest either does not exist in background or is not present in a concentration that is a relevant fraction of the $DCGL_W$.

Quality assurance and quality control measures are employed throughout the FSS process to ensure that all decisions are made based on data of acceptable quality. Quality assurance and control measures are applied to ensure:

- The plan is correctly implemented as prescribed;
- DQOs are properly defined and derived;
- All data are collected by individuals with the proper training following approved procedures;
- All instruments are properly shipped, handled, stored, operationally checked and calibrated in accordance with approved procedures;
- All collected data are validated, recorded, and stored in accordance with approved procedures;
- All required documents are properly maintained;
- If necessary, corrective actions are prescribed and implemented; and,
- Effectiveness of corrective actives is reviewed, when appropriate.

These measures apply to any services provided in support of remedial action support surveys and FSSs.

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

Documentation of the FSS survey will transpire in two types of reports. An FSS Survey Unit Release Record will be prepared to provide a complete record of the "as left" radiological status of an individual survey unit, relative to the specified release criteria. An FSS Final Report, which is a written report submitted to the NRC, will be prepared to provide a summary of the survey results and the overall conclusions from multiple survey units. It will include the associated Release Records. These reports will demonstrate that the NSS meets the radiological criteria for unrestricted use. They are discussed in detail in Section 5.10. Reports will be compiled after FSS activities for all survey units for a given final report are completed. This approach should minimize the submittal of redundant historical assessment information and provide for a logical approach to perform reviews and independent verification.

5.4 Survey Planning

5.4.1 Data Quality Objectives

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

The DQO process consists of performing the following seven steps:

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

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

<u>State the Problem</u>

The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated and the estimated resources. The problem associated with FSS is to determine whether an area meets the radiological release criterion of 10 CFR 20.1402 and the 15 mrem/year standard employed by EPA.

<u>Identify the Decision</u>

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

Identify Inputs to the Decision

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

Each of the following:

- Sampling methods;
- Sample quantity;
- Sample matrix;
- Type(s) of analyses; and,
- Analytic methods and measurement process performance criteria, including detection limits

are established to ensure adequate sensitivity relative to the action level and to minimize bias. Action levels provide the criterion for choosing among alternative actions (e.g., whether to take no action, perform confirmatory measurements or sampling). These action levels may be radioactivity concentration (dpm/100cm^2) or measurement device response (count rate corrected for background, etc.).

Define the Boundaries of the Study

This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame for collecting the data, practical constraints and the scale of decision making. In FSS, the target population is the set of direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest is specified during the planning process. The spatial boundaries include the entire area of interest including contamination depth, area dimensions, and natural boundaries, as needed. Temporal boundaries include those activities impacted by time-related events including weather conditions, seasons (i.e., more daylight available in the summer), operation of equipment under different environmental conditions, resource loading, and work schedule.

Develop a Decision Rule

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

Specify Tolerable Limits on Decision Errors

This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true and that sufficient evidence must be provided for rejection. The primary consideration during FSS will be demonstrating compliance with the release criteria. The following statement may be used as the null hypothesis at NSS: "The survey unit exceeds the release criteria."

Decision errors occur when the data set leads the decision-maker to make false rejections or false acceptances during hypothesis testing. The α error (Type I error) is set at 0.05 (5%). The β error may be variable depending upon the objectives of the surveys. A nominal value of 0.05 (5%) has been established for the β error (Type II error). Another output of this step is assigning probability limits to points above and below the gray region where the consequences of decision errors are considered acceptable. The upper bound corresponds to the release criteria. The Lower Bound of the Gray Region (LBGR) is determined in this step of the DQO process. LBGR is influenced by a parameter known as the relative shift. The relative shift is set between (and including) 1 and 3. If the relative shift is not between (or including) 1 and 3, then the LBGR is adjusted. Graphing the probability that a survey unit does not meet the release criteria may be used during FSS. This graph, known as a power curve, may be performed retrospectively (i.e., after FSS) using actual measurement data. This retrospective power curve may be important when the null hypothesis is not rejected (i.e., the survey unit does not meet the release criteria) to demonstrate that the DQOs have been met.

Optimize the Design for Obtaining Data

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

5.4.2 Survey Units

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

Survey units are limited in size based on classification, exposure pathway modeling assumptions, and site-specific conditions. The surface area limits, used in establishing the initial set of survey units for the FSS Plan, are provided in Table 5-1 for rooms and structures. The area limits for structures refer to the deck area and not the total surface area, which would include the bulkheads and overheads. This is consistent with the guidance of MARSSIM. The deck area limits given in Table 5-1 were also used to establish survey unit sizes for structures such as exterior surfaces. The limits given in Table 5-1 will also be used should the need arise to establish any new survey units beyond the initial set given in this plan.

As indicated in Table 5-1, associated areas of NSS that are classified as impacted have been divided into survey units to facilitate survey design. Each survey unit has been assigned an initial classification based on the site characterization process and the historical site assessment as described in Chapter 2 of the LTP.

Table 5-1 Survey Unit Surface Area Limits

Survey Unit Classification	Suggested Surface Area Limit
Class 1: Structures/Rooms (deck area)	$\leq 100\ m^2$
Class 2: Structures/Rooms (deck area)	$100\ m^2 \leq$ area $\leq 1{,}000\ m^2$
Class 3: Structures/Rooms (deck area)	No Limit
Systems and Components	No Limit

A survey unit can have only one classification. Thus, situations may arise where it is necessary to create new survey units by subdividing areas within an existing unit. For example, residual radioactivity may be found within a Class 3 survey unit, or residual radioactivity in excess of the $DCGL_W$ may be found in a Class 2 unit. In such cases, it may be appropriate to define a new survey unit within the original unit that has a lower (more restrictive) classification. Alternately, the classification of the entire unit can be made more restrictive.

5.4.3 Reference Coordinate Systems

On a ship, the location of any given point is defined by identifying its distance from one of three axes or reference points. Elevation is typically defined by the deck (floor) on which the point exists, or by its height above Baseline (BL). The BL is a horizontal line that follows the molded line of the keel and represents the "bottom" of the ship. Because some exterior and interior decks have an upward rise known as sheer, the height of a particular deck above BL may vary along the ship's length.

The point will be defined by its distance from the Forward Perpendicular (FP). The FP is an arbitrary fixed vertical line that runs through the intersection of the ship's bow with its design waterline. The ship's transverse frames are numbered running aft, with the FP having no frame designation, but essentially being equal to Frame 0. Transverse bulkheads (walls) are typically aligned on frames.

Finally, the point is defined by its distance from the ship's Centerline (CL). The CL is a vertical plane running the length of the ship. Distances are measured to port (left) or starboard (right). Ship compartments (rooms or spaces) may be numbered using an alpha-numeric designation to represent its location relative to the three axes (Deck, Frame, and Center). Compartments that are to port of center use an even number designation. Compartments that are to starboard of center use an odd number designation. Compartments which extend to both sides of the CL are said to be "on Center" and use the numeral "0" in their designation. However, on NSS, many compartments employ a simple noun-name designation.

The reference coordinate system for NSS survey units will include a benchmark, or origin point. This benchmark will be defined using the ship's reference system and will typically be located at the lowest, forwardmost point of the survey unit that is closest to the CL. The coordinate system used for surveys will typically take the form of a triangular grid of intersecting, perpendicular lines; but other patterns (e.g., rectangular and polar) may be used as convenient, depending on the circumstances or shape of the survey unit. The system will serve as a convenience for documenting survey efforts and other information pertaining to a given survey unit. The coordinate system also provides a means to specify general locations for measurements performed for quality control or verification purposes. The benchmark and survey pattern will be provided for each survey unit in the FSS packages.

Physical gridding of a survey unit will only be done in cases where it is beneficial and cost effective to do so. When physical gridding is used, benchmark locations will be designated by marking a spot by a suitable technique.

5.4.4 Area Preparation: Isolation and Control

Before FSS activities can begin in an area, a transition must occur where planned decommissioning activities are completed, and the area is subsequently assessed to scope the required isolation and control measures. A walkdown will occur to establish if the area is ready for final survey activities and identify any work practice issues that must be addressed in survey planning and design. Determination of readiness for FSS will be based on a characterization survey, turnover survey and/or a Remedial Action Support Survey (RASS) indicating that the residual radioactive material is likely to comply with the FSS criteria.

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

- Known contaminated decommissioning activities in the area are complete and any additional decommissioning activities identified shall pose a very low risk to add contamination to an area, including removal, as necessary, of items (e.g., equipment mounts, bulkhead hangers, and exposed studs) that could interfere with final survey activities;
- All planned decommissioning activities in areas either adjacent to the survey unit to be isolated or that could otherwise affect it are controlled using isolation and control techniques, are complete or are deemed not to have any reasonable potential to spread radioactive material to the survey unit;
- All tools and equipment not needed for final survey activities are removed;
- Any equipment to be used for final survey activities is evaluated to ensure it does not pose the potential for introducing radioactive material into the survey unit; and,
- Where practical, transit paths to or through the survey unit, except those required to support final survey activities, are eliminated or re-routed.

Once the area meets the isolation and control criteria, isolation and control will be achieved through a combination of personnel training, physical barriers, postings, and site notices as appropriate, to prevent unauthorized access to an isolated survey unit.

Isolation and control measures will be implemented through approved plant procedures. An administrative process will be used to evaluate, approve (or deny), and document all activities conducted in these areas during and following FSSs.

5.4.5 Selection of DCGLs

Chapter 6 of this plan describes in detail the modeling performed to develop the radionuclide specific DCGLs for room, system, and equipment surfaces. For situations where gross activity measurement methods are used to demonstrate compliance with the license termination criteria, the radionuclide specific DCGLs will be used to establish gross activity DCGLs. These gross activity DCGLs will be established based on a representative radionuclide mix established for the entire ship. In cases where measurable activity still exists, scaling factors will be used to establish the activity contribution for any hard-to-detect radionuclides that may be present. Scaling factors will be selected from available composite waste stream analyses or similar assays. Such analyses may be performed periodically and documented in support of waste characterization needs.

For cases of survey units for which there is no measurable activity distinguishable from background, a representative radionuclide mix may be selected based upon historical characterization information for the survey unit of interest or for units with similar history and physical characteristics (e.g., information from adjacent areas).

5.4.5.1 Gross Activity DCGLs

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

$$Gross\ Activity\ DCGL = \frac{1}{\frac{f_1}{DCGL_1}+\frac{f_2}{DCGL_2}+\cdots\frac{f_n}{DCGL_n}} \qquad \text{(Equation 5-1)}$$

Where:

f_n = *fraction of the total activity contributed by radionuclide n*

n = *the number of radionuclides*

$DCGL_n$ = *DCGL for the Radionuclide of Concern (ROC) - presented in Chapter 6*

Gross activity DCGLs will be developed for gross beta measurements. No gross alpha activity measurements will be made.

In NUREG-1757, Vol 2, Section 3.3, the NRC staff considers radionuclides and exposure pathways that contribute no greater than 10 percent of the dose criteria to be insignificant contributors. Equation 5-1 will be applied to all radionuclides in which no single radionuclide can be screened out if greater than or equal to 5% of the mix and the sum of all screened radionuclides cannot exceed 10%. If Hard-To-Detect radionuclides are not screened out, they will be included by surrogate ratio DCGL as shown below.

5.4.5.2 Surrogate Ratio DCGLs

It is acceptable industry practice to assay a Hard-To-Detect (HTD) radionuclide by using a surrogate relationship to an Easy-To-Detect (ETD) radionuclide. A common example would be to use a beta measurement to assay an alpha emitting radionuclide. Another example would be to relate a specific radionuclide, such as Cesium-137 or Co-60, to one or more radionuclides of similar characteristics. In such cases, to demonstrate compliance with the release criteria for the survey unit, the DCGL for the surrogate radionuclide or mix of radionuclides must be scaled to account for the fact that it is being used as an indicator for an additional radionuclide or mix of radionuclides. The result is referred to as the surrogate DCGL.

The following process will be applied to assess the need to use surrogate ratios for FSS:

- Determine whether HTD radionuclides (e.g., Ni-63, Sr-90, H-3) are likely to be present in the survey unit based on process knowledge, historical data or characterization;
- When HTD radionuclides are likely to be present, establish a relationship using a representative number of samples (typically six or more). The samples may come from another survey unit if the source of the contamination and expected concentrations are reasonably the same. These samples will be analyzed for ETD and HTD radionuclides using gross alpha, alpha spectroscopy, gross beta analysis or gamma spectroscopy techniques;
- Screen HTD radionuclides in which no single radionuclide can be screened out if greater than or equal to 5% of the mix and the sum of all screened radionuclides cannot exceed 10%; and,

- Radionuclides not screened out will require a surrogate DCGL. Surrogate relationships will be determined from the samples results by developing a surrogate relationship for each HTD radionuclide.

The surrogate DCGL is computed as:

$$DCGL_{surrogate} = DCGL_{ETD}\ x\ \frac{DCGL_{HTD}}{(f_{HTD:ETD} x DCGL_{ETD}) + DCGL_{HTD}} \qquad \text{(Equation 5-2)}$$

Where:

$DCGL_{ETD}$ = *the DCGL for the easy-to-detect radionuclide*

$DCGL_{HTD}$ = *the DCGL for the hard-to-detect radionuclide*

$F_{HTD:ETD}$ = *the activity ratio of the hard-to-detect radionuclide to the easy-to-detect radionuclide*

5.4.5.3 Elevated Measurement Comparison and Area Factors

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

Area factors are only used on Class 1 land and building surveys in which the scanning Minimum Detectable Concentrations (MDC) is greater than the DCGL. Review of the scanning MDCs for the instruments to be used for the FSS are a small fraction of the DCGLs. Therefore, no area factors have been calculated and the $DCGL_{EMC}$ will not be used.

5.4.5.4 Release Limits for Non-Structural Components and Systems

Non-structural components and systems will be surveyed to the DCGLs specified in Chapter 6 of this plan.

5.5 Final Status Survey Design Elements

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

FSSs for the NSS rooms and areas will be designed, following MARSSIM guidance, using combinations of fixed measurements, traditional scanning surveys, and other advanced survey methods, as appropriate, to evaluate survey units relative to their applicable release criteria.

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

For combinations of fixed measurements and traditional scanning, MARSSIM methodology is to select a requisite number of measurement locations to satisfy the decision error rates for the non-parametric statistical test to be used for data evaluation and to account for sample losses or data anomalies. The purpose of scans is to confirm that the area was properly classified and to identify elevated areas in the event that contamination is not uniformly distributed and represented by the results of fixed measurements. Depending on the sensitivity of the scanning method used, the number of fixed measurement locations may need to be increased so the spacing between measurements is reduced. Details on selecting the number and location of fixed measurements are presented in subsequent subsections of this plan. The coverage requirements that will be applied for scans performed in support of FSSs for the NSS are:

- For Class 1 survey units, 100% of the accessible surface will be scanned;
- For Class 2 survey units, between 10% and 100% of the surface will be scanned in a combination of systematic and judgmental measurements for external surface area units and for deck and lower bulkheads of structures; and 10% to 50% of the surface will be covered for upper bulkheads and overheads;
- Scanning will be done on a judgmental basis for Class 3 survey units.

Though the emphasis of MARSSIM is on conducting FSSs through a combination of fixed measurements and scans, MARSSIM also allows for use of advanced survey technologies when these techniques meet the applicable requirements for data quality and quantity. Advanced technologies in this context refers to survey techniques where the instrument is capable of recording data as an area is surveyed and the measurement sensitivity is an acceptable fraction of the applicable $DCGL_W$. Such methods are desirable for FSSs since they allow survey units to be assessed with a single measurement rather than separate fixed measurements and scans. Advanced survey techniques may be used alone or in combination with fixed measurements and scans to assess a survey unit.

MARSSIM states that MDCs should be as far below the $DCGL_W$, as possible, with values less than 10% of the $DCGL_W$ being preferred and up to 50% of the $DCGL_W$ being acceptable. These same criteria will be used when deciding if advanced survey techniques can be used in place of fixed measurements and traditional scans for a given survey unit.

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

- For Class 1 or Class 2 survey units, advanced survey technologies may be used exclusively only in survey units for which the above coverage requirements can be achieved and MDCs are no greater than 50% of the applicable $DCGL_W$.
- For Class 1 or Class 2 survey units for which advanced technologies would have an acceptable MDC, but the above coverage requirements cannot be achieved, advanced technologies may be

used over 100% of the accessible area with a combination of fixed measurements and traditional scans used over the remainder of the area.

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

5.5.1 Selecting the Number of Fixed Measurements and Locations

The MARSSIM methodology for evaluating whether a survey unit meets its applicable release criterion using fixed measurements plus scans is based on using non-parametric statistical tests for data assessment. Specifically, the methods of MARSSIM are based on two non-parametric tests: the Wilcoxon Rank Sum (WRS) test and the Sign test. Selection of the required minimum number of data points depends on which statistical test is going to be used to evaluate the data, and thus depends on what type of measurements are to be made (gross measurement, net measurement or radionuclide specific) and if the radionuclide(s) of interest appear(s) in background. On NSS, FSS data will only be evaluated with the Sign Test because the radionuclide(s) of interest either does not exist in background or is not present in a concentration that is a relevant fraction of the $DCGL_W$.

5.5.1.1 Establishing Acceptable Decision Error Rates

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

The decision error rates for FSSs designed for the NSS site will be set as follows:

- The α value will always be set to 0.05 unless prior NRC approval is granted for using a less restrictive value; and,
- The β value is nominally set to 0.05 but may be changed without NRC approval if it is found that more fixed measurements than necessary are being made to demonstrate compliance with the release criterion.

5.5.1.2 Determining the Relative Shift

Another input to the process of selecting the required number of measurements that is somewhat independent of the statistical test to be employed is the determination of what is called the relative shift. The relative shift is a parameter that quantifies the concentrations to be measured in a survey unit relative to the variability in these measurements. The relative shift is a function of the $DCGL_W$, a parameter called the "lower bound of the gray region" (LBGR) and the expected standard deviation of the measurements to be made in the survey unit (σ_s). The σ_s values will be selected by:

- Using existing characterization or remediation support survey data; or,

- Making preliminary measurements.

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

The LBGR represents the concentration to which the survey unit must be cleaned (decontaminated) to have an acceptable probability of passing the statistical test. The difference between the $DCGL_W$ and the LBGR, known as the shift, can be thought of as a measure of the resolution of the measurements that will be made in a survey unit. If the LBGR is near the $DCGL_W$, the shift will be small, and thus a strong potential for Type I errors will exist. Likewise, if the shift is large, the probability of Type II errors increases. The shift is denoted as Δ.

The relative shift (Δ/σ_s) is computed as the quotient of the shift and the appropriate standard deviation values. If no reference area data are needed to evaluate the survey results, the expected standard deviation of the measurements (σ_s) is used. When preliminary data are not obtained, it may be reasonable to assume a coefficient of variation on the order of 30%, based on experience.

To compute the relative shift, the appropriate sigma value and an initial LBGR are selected. Per MARSSIM, the initial value for the LBGR will be set to one-half of the $DCGL_W$. If the resulting relative shift is not between 1.0 and 3.0, the LBGR is adjusted until it is. If the relative shift is too low, the LBGR is decreased; if the relative shift is too high, the LBGR is increased.

5.5.1.3 Selecting the Required Number of Measurements for the Sign Test

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

$$N = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(Sign\ \ p - 0.5)^2} \qquad \text{(Equation 5-3)}$$

Where:

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

In lieu of calculating the value of N by Equation 5-3, the value of N will be obtained from Table 5.5, "Values of N for Use with the Sign Test" in MARSSIM. On NSS, FSS data will only be evaluated with the Sign Test because the radionuclide(s) of interest either does not exist in background or is not present in a concentration that is a relevant fraction of the $DCGL_W$.

5.5.1.4 Determining Measurement Locations

For Class 1 and Class 2 survey units, fixed measurements will be performed over a systematic measurement pattern consisting of a grid having either a triangular or a square pitch. The pitch (grid spacing) will be determined based on the number of measurements required and whether the desired grid is triangular or square. Given that a triangular grid in general is more efficient than a square grid for detecting small areas of elevated activity, triangular grids should be employed for FSSs involving fixed measurements in Class 1 and Class 2 survey units when practical.

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

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

$$L = \sqrt{\frac{A}{0.866N}} \quad \text{for a triangular grid, or} \qquad \text{(Equation 5-4)}$$

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

Where:

L = *grid spacing (dimension is square root of the area)*

A = *the total area of the survey unit*

N = *the desired number of measurements*

The value of N should include additional measurements required to ensure against losses or unusable data.

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

Measurement locations may also be determined with *Visual Sample Plan (VSP)*. Pacific Northwest National Laboratory created VSP. VSP is a software tool that supports the development of a defensible sampling plan based on statistical sampling theory and the statistical analysis of sample results to support confident decision making. VSP couples the site, building, room and sample location visualization capabilities with optimal sampling design and statistical analysis strategies.

5.5.2 Judgmental Assessments

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

5.5.3 Data Investigations

5.5.3.1 Investigation Levels

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

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

Table 5-2 Investigation Levels

Survey Unit Classification	For fixed measurements, perform investigation if:	For scan measurements, perform investigation if:
Class 1	> Operational $DCGL_W$ and an outlier.	> Operational $DCGL_W$
Class 2	> Operational $DCGL_W$	> Operational $DCGL_W$
Class 3	> 0.25 × Operational $DCGL_W$	>0.25 x Operational $DCGL_W$
Systems and Components	> Operational $DCGL_W$	> Operational $DCGL_W$

For Class 1 survey units, measurements above the $DCGL_W$ are not necessarily unexpected. However, such a result may still indicate a need for further investigation if it is significantly different than the other measurements made within the same survey unit. Thus, some additional evaluation criterion is needed to assess if results from fixed measurements in a Class 1 survey unit that exceed the $DCGL_W$ warrant further attention. Measurements in Class 1 survey units that exceed the $DCGL_W$ and differ from the mean of the remaining measurements by more than three (> 3) standard deviations will therefore be investigated. Measurements in Class 1 units that exceed the $DCGL_W$, but do not differ from the mean by less than or equal to three (≤ 3) standard deviations may still be investigated based on professional judgment, as may any measurements that differ significantly from the rest of the measurements made within a given survey unit.

In Class 2 or Class 3 areas, neither measurements above the $DCGL_W$ nor areas of elevated activity are expected. Thus, any fixed measurements or sampling results that exceed the $DCGL_W$ in these areas will be investigated. In the case of Class 3 areas, where any residual radioactivity would be unexpected, fixed measurement or sample results that are greater than 0.25 x $DCGL_W$ will be investigated.

In cases where an advanced survey method is used instead of fixed measurements, the investigation levels given in Table 5-1 for fixed measurements will be applied.

5.5.3.2 Investigations

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

5.5.3.3 Remediation

If during the performance of an FSS, any areas of residual activity are found to be in excess of the $DCGL_W$ and an outlier, those areas will be remediated with the goal to reduce the activity to less than or equal to the $DCGL_W$. Remediation actions are discussed in Section 4 and documented as described in Section 5.10.

5.5.3.4 Re-classification

If survey results from a Class 2 unit indicate residual activity in excess of the $DCGL_W$, all or part of the survey unit will be re-classified as Class 1. The decision to either re-classify the entire survey unit or sub-divide it, will depend on the areal extent of the activity found to exceed the $DCGL_W$. If a survey unit is sub-divided, the portion of the unit containing the elevated activity will be assigned the more restrictive classification, while the remainder of the unit will retain its original classification.

If survey results for a Class 3 survey unit indicate unexpected residual activity, all or part of the unit containing the activity will be re-classified. Since Class 3 survey units tend to be physically large, it is not anticipated that an entire unit would require re-classification in the event unexpected residual activity is found. However, if the unit is small enough, it is possible an entire Class 3 unit could be re-classified.

If residual activity is found in a Class 3 survey unit at levels equal to 50% of the $DCGL_W$, the affected portion of the unit will be re-classified as Class 2. In the unlikely event that residual activity is found in excess of the $DCGL_W$, the affected area will be re-classified as Class 1.

Re-classification of areas from a less to a more restrictive classification may be done without prior NRC approval; however, re-classification to a less restrictive classification would require prior NRC approval. MARAD does not expect to make less restrictive reclassifications after submittal of the LTP.

5.5.3.5 Re-survey

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

If a survey unit is sub-divided, the survey design for the remaining area of the original survey unit may or may not be affected depending on the remaining surface area of the unit and its classification. If the original survey unit was Class 3, then the only impact on the survey design (in the case of fixed measurements or sampling) is to perform additional measurements at randomly selected locations until the required total number of measurements is met. If the original survey unit was Class 2, the spacing of the measurement locations may need to be adjusted depending on the remaining surface area of the survey unit relative to its original area. If there is a large change in the surface area, then a new survey design will be necessary to accommodate the required number of measurements in the smaller area. If there is not a large change in area, then the impact on the grid spacing is minimal (with respect to areal coverage), and additional measurement locations need only be selected at random to obtain the required number of measurements. Thus, for the purpose of FSSs at the NSS, a change in the surface area of a Class 2 survey unit that changes the grid spacing by 25% or less will not require fixed measurements or sampling to be repeated using a tighter spacing. If the change in surface area is such that the grid spacing is changed by more than 25%, a new survey design will be required. Assessments against the 25% criterion, if applicable, will be an element of the investigation process and documented in the FSS report for the affected survey unit.

If remediation is required in only a small area of a Class 1 survey unit, any replacement measurements required will be made within the remediated area at systematically selected locations following verification that the remediation activities did not affect the remainder of the unit. Re-survey will be required in any area of a survey unit affected by subsequent remediation activities.

5.6 Survey Protocol for Non-structural Systems and Components

The guidance provided in MARSSIM for conducting FSSs does not include guidance for conducting FSSs for non-structural system or components. Non-structural systems and components refer to anything not attached to or not an integral part of a building or structure.

Surface activity assessments for non-structural systems and components can be made by making measurements at traps, tanks, open piping and other appropriate access points where activity levels should be representative of those on the interior surfaces. Assessments may also be made via in-situ gamma-spectroscopy or pipe crawlers, provided adequate instrument efficiencies and detection limits can be achieved. Detection limits for surface activity assessments should be at least 50% of the release limits. If necessary, scaling factors may be applied to establish gross activity levels via radionuclide-specific measurements or other assessments, as appropriate. MARAD expects limited use of in-situ gamma-spectroscopy or pipe crawlers.

Evaluations as to whether 1) material should be considered as a structure or a component will be made and 2) comparisons with the dose modeling scenarios used to develop the DCGLs that govern release of structures. Examples of parts of structures that are considered in the development of DCGLs include decks, bulkheads, overheads, doors/hatches, windows/portholes, sinks, hoods, lighting fixtures, built-in laboratory benches, and built-in furniture. Examples of non-structural systems and components include pumps, motors, heat exchangers, and piping between components.

5.7 Survey Implementation and Data Collection

The requirements and objectives outlined in the LTP and the DQAP [Reference 5-6] will be incorporated into approved procedures. Procedures will govern the survey design process, survey performance and data assessment (decision making). The FSS design will be carried out in accordance with the approved procedures and the DQAP, resulting in the generation of raw data. The product of the survey design process is a survey package, which addresses various elements of the survey, including, but not limited to:

- Maps of the survey area showing the survey unit(s) and measurement/sample locations, as appropriate;
- Applicable DCGLs;
- Instrumentation to be used;
- Types and quantities of measurements to be made or collected;
- Investigation criteria;
- QA/QC requirements (e.g., replicate measurements);
- Applicable health and safety procedures; and,
- Applicable operating procedures.

5.7.1 Survey Methods

The survey methods to be employed in the FSSs will consist of combinations of scanning, fixed measurements, advanced technologies, sampling, and other methods as needed to meet the survey objectives. Additional methods may be used if such become available between the time this plan is adopted and final survey activities are completed. However, any new technologies must still meet the applicable requirements of this plan. Note that in some cases, the same instrument may be used for more than one type of survey.

5.7.1.1 Scanning

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., bulkhead, deck, equipment). The term scanning survey is used to describe the process of moving portable radiation detectors across a surface with the intent of locating residual radioactivity. Investigation levels for scanning surveys are determined during survey planning to identify areas of elevated activity. Scanning surveys are performed to locate contamination anomalies indicating residual gross activity that may require further investigation or action. These investigation levels may be based on the $DCGL_W$.

No matter what survey approach is selected (combination of instrumentation and techniques), one of the most important elements of a survey is *a priori* scanning to confirm that the unit is properly classified and to identify any areas where residual activity levels are elevated relative to the $DCGL_W$. The purpose of scanning is to detect areas of residual activity that may not be detected by other measurement methods. Thus, scanning should always be performed prior to any fixed measurements or sample collections in a survey unit. If the scanning indicates that the unit or some area within the unit has been improperly classified, then the survey design process must be evaluated to either assess the effect of re-classification on the survey unit as a whole (if the whole unit requires re-classification) or a new design must be established for the new unit(s) (in the case of sub-division). A new survey design will require a re-evaluation of the survey strategy to decide if it can meet the requirements of the revised survey design. If not, the survey strategy must be revised based on the available instrumentation and methods.

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

Table 5-3 Traditional Scanning Coverage Requirements

Survey Unit Classification	Required Scanning Coverage Fraction
Class 1	100%
Class 2	Decks, or lower bulkheads of rooms: 10% to 100% Upper bulkheads or overheads: 10% to 50%
Class 3	Judgmental
Systems and Components	Judgmental

5.7.1.2 Fixed Measurements

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

5.7.1.3 Advanced Technologies

In the context of this Plan, advanced technologies refer to survey instruments or methods that create a spatially correlated log of the measurements made as the detector is passed over an area, such as a pipe crawler or a Surface Contamination Monitor (SCM) which is a position sensitive gas proportional counter. This logging of all measurements allows quantitative assessments of activity levels to be made, thus serving the same role as fixed measurements. Having all measurements logged allows statistical analyses to be made using a large number of measurements, which provides for enhanced detection sensitivity relative to traditional scanning. The sensitivity achieved using advanced survey methods may, in some cases, be small enough relative to the $DCGL_W$ that the advanced method alone will allow a decision to be made as to whether a survey unit meets the release criterion without the need for additional fixed measurements. The fact that the instrument records every measurement made over the entire area it covers inherently addresses the issue of small areas of elevated activity. Average and maximum residual activity concentrations can be quantified over any area desired, allowing one to assess compliance with the $DCGL_W$ by inspection. MARAD does not expect to use advanced technologies.

5.7.1.4 Samples

Sampling is the process of collecting a portion of a medium as a representation of the locally remaining medium. The collected portion of the medium is then analyzed to determine the radionuclide concentration. Examples of materials that may be sampled include trap sediments and lagging. Samples will be collected during characterization surveys. No samples are anticipated during FSSs.

If water or sludge is encountered in a system during FSS, sample results will be compared the Effluent Concentrations (ECs) listed in Table 2, Column 2 of Appendix B to 10CFR20. If the sample results are greater than the ECs, the medium will be remediated or removed.

All collected samples will be controlled under the chain of custody protocols. Samples will be sent to off-site laboratories that 1) participate in inter- and intra-laboratory comparisons, 2) use NIST traceable sources, 3) have a corrective action program and 4) have an internal audit program.

5.7.2 Survey Instrumentation

5.7.2.1 Instrument Selection

The selection and proper use of appropriate instruments for both fixed measurements and scanning is one of the most important factors in assuring that a survey accurately determines the radiological status of a survey unit and meets the survey objectives. The particular capabilities of a radiation detector establish its potential for being used in conducting a specific type of survey. For certain radionuclides or radionuclide mixtures, both alpha and beta radiation may have to be measured. In addition to assessing average radiological conditions, the survey objectives must address identifying small areas of elevated activity.

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

- Gas-filled detectors;
- Scintillation detectors; and,
- Solid-state detectors.

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

Instruments must be stable and reliable under the environmental and physical conditions where they will be used, and their physical characteristics (size and weight) should be compatible with the intended application. The instrument must be able to detect the type of radiation of interest, and, depending on the application, the measurement system should be capable of measuring levels that are less than the $DCGL_W$. However, in some cases instruments used for scanning may have detection limits that are greater than the $DCGL_W$. This is allowed by MARSSIM and is acceptable as long as the grid spacing (for Class 1 survey units) and investigation levels used are in accordance with this plan.

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

In general, instruments used for measurements to demonstrate that the average concentration in a survey unit is less than the $DCGL_W$ should have an MDC that is no greater than 50% of the $DCGL_W$. However, though it is not anticipated, there may be special circumstances where the best available technology cannot meet this goal. In such a case, measurements will be made at the best MDC that can be achieved.

5.7.2.2 Calibration and Maintenance

All instrumentation used for measurements to demonstrate compliance with the radiological criterion for license termination at the NSS will be calibrated and maintained under approved procedures and the DQAP or vendor QA plan that satisfies the requirement of the DQAP. Instruments will be calibrated for normal use under typical field conditions. Calibration standards will be traceable to the National Institute of Standards and Technology. If external vendors are used for instrument calibration or maintenance, these services must be approved and conducted under the DQAP. Calibration records will be maintained as required by approved procedures and the DQAP.

Instruments used to measure gross beta surface activity will be calibrated to Tc-99 or Co-60 to bound the beta energies for the beta-emitting radionuclides that will be encountered during final survey activities. Based upon the characterization data in CR-109 [Reference 5-7], measurements for alpha activity will not be needed. If additional characterization activities are performed and alpha measurements are needed, Pu-239 or Th-230 will likely be used to calibrate instruments used to assess alpha surface activity or the alpha surface activity will be scaled to an Easy-to-Detect radionuclide.

Instrument efficiencies may require modifications to account for surface conditions or paint coverings. Such modifications, if necessary, will be established using the information in Section 5 of NUREG-1507 [Reference 5-8] and pertinent site characterization data.

5.7.2.3 Operational Checks

Instrumentation will be checked for proper operation in accordance with approved procedures. If the instrument operational check does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable operation is again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure.

5.7.2.4 MDC calculations

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

5.7.2.4.1 MDCs for Static Measurements

Per NUREG-1507, MDCs for static measurements in which the background and sample count times are equal are computed as:

$$MDC_{static} = \frac{3+4.65\sqrt{B}}{Kt} \qquad \text{(Equation 5-6)}$$

Where:

3 and 4.65 = constants as described in NUREG-1507

B = background counts during the measurement time interval (t)

t = counting time

K = a proportionality constant that relates the detector response to the activity level in the sample being measured

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

Calculation of the MDC in which the background and measurement count times are not equal are computed as:

$$\mathrm{MDC_{static}} = \frac{3+3.29\sqrt{R_b t_s (1+\frac{t_s}{t_b})}}{eff * t_s * (\frac{A}{100})} \quad \text{(Equation 5-7)}$$

Where:

R_b = *background count rate (c/m)*

t_s = *sample count time (min)*

t_b = *background count time (min)*

eff = *counting efficiency (c/d) = source efficiency (ϵ_s) x instrument efficiency (ϵ_i)*

A = *physical area of detector (cm²)*

5.7.2.4.2 *MDCs for Beta-Gamma Scan Surveys for Structure Surfaces*

As recommended in NUREG-1507, MDCs for surface scans for structure surfaces for beta and gamma emitters will be computed as:

$$\mathrm{MDC_{scan}} = \frac{1.38\sqrt{B_t} * (\frac{60}{t})}{\sqrt{p} * eff * \left(\frac{A}{100}\right)} \quad \text{(Equation 5-8)}$$

Where:

1.38 = *sensitivity index,*

B = *number of background counts in time interval t*

p = *surveyor efficiency = 0.5*

eff = *counting efficiency (c/d) = source efficiency (ϵ_s) x instrument efficiency (ϵ_i)*

A = *physical area of the detector (cm²)*

t = *time interval of the observation while the probe passes over the source*

The value of 1.38 used for the sensitivity index corresponds to a 95% confidence level for detection of a concentration at the scanning MDC with a false positive rate of 60%. The factor of 100 corrects for probe areas that are not $100 cm^2$. In the case of a scan measurement, the counting interval is the time the probe is over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. The efficiency term in Equations 5-7 and 5-8 may be adjusted to account for effects such as self-absorption, as appropriate.

5.7.2.5 Typical Instrumentation and MDCs

Table 5-4 provides nominal data for the types of field instrumentation anticipated for use in the final survey efforts for the NSS.

Table 5-4 Available Instruments and Associated MDCs

Instrument	Application	Nominal Efficiency	Nominal Background	Nominal MDC (fixed measurement)	Nominal Scan MDC[1]
Pancake GM Model 44-9 ($15cm^2$)	beta-gamma scans or fixed measurements	10% (Co-60)	50 cpm	2393 dpm/$100cm^2$ (1 minute count)	7126 dpm/$100cm^2$
ZnS scintillator Model 43-89 ($125cm^2$)	beta-gamma scans or fixed measurements	12% (Co-60)	150 cpm	400 dpm/$100cm^2$ (1 minute count)	1234 dpm/$100cm^2$
ZnS scintillator Model 43-93 ($100cm^2$)	beta-gamma scans or fixed measurements	15% (Co-60)	150 cpm	400 dpm/$100cm^2$ (1 minute count)	1234 dpm/$100cm^2$
Gas Flow Proportional Counter Floor monitor Model 43-37 (584 cm^2)	beta-gamma scans	15% (Co-60)	350 cpm	NA	1908 dpm/100 cm^2
HPGe	in-situ gamma spectroscopy	Varies with energy and geometry	Varies with energy and geometry	0.05 pCi/g Co-60 0.05 pCi/g Cs-137 (10-minute counts)	N/A
Position-sensitive gas proportional counter	scan-and-record surveys	18% (Co-60)			Typical value 1,925 dpm/$100cm^2$. The MDC varies with detector speed and distance.

Note 1: Assumes a 1 second observation period

5.7.3 Survey Considerations for Structures and Equipment

The condition of surfaces following decontamination activities can affect the choice of survey instruments and techniques. Surface irregularities may also cause difficulty in rolling or maneuvering detector systems on wheels. In such cases, depending on the survey instrumentation used and the radiations of interest, evaluations of instrument efficiencies may be required to assess the need for corrections for variables such as source-to-detector distance or surface condition. Surface efficiency corrections, if necessary, may be established using the information given in Section 5 of NUREG-1507 or, if practical, via fixed measurements. If any corrections to measurement efficiencies are required, the impact of these corrections on instrument MDCs will be assessed to ensure that measurements can still be performed with the required sensitivity relative to the applicable DCGLs.

Expansion joints, stress cracks, deck/bulkhead interfaces and penetrations into decks and bulkheads for piping, conduit, anchor bolts, etc., are potential sites for accumulation of contamination and pathways for migration into bilges and hollow bulkhead spaces. External surfaces and drainage points are also important survey locations.

5.8 Survey Data Assessment

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

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

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

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

The initial data evaluation will be as described in Table 5-5.

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

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

Note that given the site contains no soil, laboratory analysis is not anticipated. Similarly, split sampling is not anticipated given that it is typically associated with evaluating soil.

5.8.1 Sign Test

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

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

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

If S+ is greater than the critical value in MARSSIM Table I.3, then the null hypothesis is rejected, and the survey unit meets the release criteria.

5.8.2 Unity Rule

When radionuclide specific measurements are made in survey units having multiple radionuclides, compliance with the radiological release criterion will be assessed through use of the unity rule, also known as the sum of fractions. The unity rule, represented in the expression below, is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one, i.e.:

$$\frac{C1}{DCGL1} + \frac{C2}{DCGL2} + \ldots + \frac{Cn}{DCGLn} \leq 1 \qquad \text{(Equation 5-9)}$$

Where:

C_n = *Concentration of radionuclide n*

$DCGL_n$ = *DCGL for radionuclide n*

5.8.3 Data Assessment Conclusions

The result of the data assessment is the decision to reject or not to reject the null hypothesis. A rejection of the null hypothesis leads to the decision that the survey unit meets the release criterion. If the data assessment concludes that the null hypothesis cannot be rejected, this may be due to one of two things: 1) the average residual concentration in the survey unit exceeds the $DCGL_W$; or 2) the analysis did not have adequate statistical power. "Power" in this context refers to the probability that the null hypothesis is rejected when it is indeed false. A retrospective power analysis can be used if a survey unit is found not to meet the release criterion to determine if this is indeed due to excess residual activity or if it is due to an inadequate sample size.

Retrospective power analyses, if necessary, will be performed following the methods of MARSSIM Section I.9 for the Sign test. If the analysis finds that an inadequate number of measurements were collected to support the data assessment for a given survey unit, additional measurements may be collected, and the analysis will be repeated. Increasing the number of measurements acquired within a given survey unit increases the probability of rejecting the null hypothesis when it is indeed false. Likewise, if the analysis with the additional measurements still concludes that the residual concentration in the survey unit exceeds the $DCGL_W$, then the unit must be remediated and resurveyed.

5.9 Notes on Structure and System Surveys

As described in Chapter 3, a number of structures and components have been left in-situ and are expected to be retained subject to confirmation through FSS. This section describes FSS considerations for these structures and components, which includes the ship's hull itself.

5.9.1 Exterior Hull Survey

As described in detail in Chapter 2, a MARSSIM based survey of the exterior hull while the ship was on drydock was performed in 2019 [Reference 5-9].

The MARSSIM survey of the exterior hull demonstrates that:

- No unexpected results or trends are evident in the data;
- The sampling and survey results demonstrate that residual radioactivity in the survey areas are indistinguishable from background levels;
- The data quality meets the necessary requirements and is deemed to be acceptable for its intended purpose;
- The amount of data collected from each survey unit is adequate to provide the required statistical confidence needed to decide that the DCGLs are met; and,
- All measurements were well below the Co-60 building surface screening level in NUREG-1757.

MARAD proposes that these surveys are acceptable for FSS and therefore, the null hypothesis that residual radioactivity in the survey units exists in concentrations above the applicable DCGLs should be rejected for the survey units of the hull.

An FSS report will be prepared and submitted for these hull surveys.

5.9.2 Neutron Shield Tank / Fuel Transfer Tank

The primary (neutron) shield tank surrounds the reactor vessel up to the hot legs. It is approximately 185 inches in outside diameter and 17 feet, 8 inches high. When filled, the shield tank formed a 33-inch-thick water annulus that provided the required neutron shielding to prevent excessive neutron activation of material inside the containment vessel and to reduce the neutron doses outside the secondary shield. Peripheral lead shielding, varying in thickness from 1 to 4 inches of lead, is placed on the outside of the primary shield tank.

The fuel transfer tank is similar in dimensions to the primary shield tank. It surrounds the reactor vessel head above the hot legs and is normally empty. It has no lead shielding. During refueling operations before the reactor vessel head is lifted, the fuel transfer tank (also called the upper shield tank in the 1968 refueling log) is filled with water to 36 inches above the lower reactor vessel flange to reduce dose during refueling when the head is removed from the vessel.

Portions of the outer shells of these tanks have been decontaminated and will be left in place. A MARSSIM based survey will be designed and implemented for both tanks. Based upon the current decommissioning schedule, performance of these surveys will be completed prior to approval of the LTP. MARAD proposes that these surveys are acceptable for FSS, and therefore, the null hypothesis that residual radioactivity in these tank shells exists in concentrations above the applicable DCGLs should be rejected. The NRC will be notified prior to performance of the surveys.

5.9.3 Steam Generators

The heat exchanger tube sheet and U-tubes were removed from both steam generators. The interior of each SG/Heat Exchanger shell has been decontaminated. They will be reassembled after license termination and left in place. Prior to being reassembled, the interior surface of both steam generator shells will be surveyed. A MARSSIM based survey will be designed and implemented. Based upon the current decommissioning schedule, performance of these surveys will be completed prior to approval of the LTP. MARAD proposes that these surveys are acceptable for FSS, and therefore, the null hypothesis that residual radioactivity in the SG/Heat Exchanger shells exists in concentrations above the applicable DCGLs should be rejected. The NRC will be notified prior to performance of the surveys.

5.9.4 Pressurizer

The interior of the upper section of the Pressurizer has been decontaminated to allow leaving it in place on completion of the decommissioning. The lower section containing the heater sleeves and other penetrations will be disposed as Low Level Radioactive Waste (LLRW). The remaining interior surface of the Pressurizer will be surveyed. A MARSSIM based survey will be designed and implemented. Based upon the current decommissioning schedule, performance of this survey will be completed prior to approval of the LTP. MARAD proposes that these surveys are acceptable for FSS, and therefore, the null hypothesis that residual radioactivity in the Pressurizer exists in concentrations above the applicable DCGLs should be rejected. The NRC will be notified prior to performance of the survey.

5.9.5 Double Bottom Tanks

Per current plans, the Fresh Water Shield Tank, PD-T5, and PD-T6 will be decontaminated to allow leaving each tank in place on completion of the decommissioning. The interior surfaces of these tanks will be surveyed. A MARSSIM based survey will be designed and implemented. Based upon the current decommissioning schedule, performance of these surveys will be completed prior to approval of the LTP. MARAD proposes that these surveys are acceptable for FSS, and therefore, the null hypothesis that residual radioactivity in the tanks exists in concentrations above the applicable DCGLs should be rejected. The NRC will be notified prior to performance of the surveys.

5.10 Final Status Survey Release Records and Reports

The documentation describing the FSS for a given survey unit will include:

- A physical description of the survey unit;
- A summary of any characterization data associated with the survey unit including any required investigations, re-classifications or subdivisions;
- The classification history of the unit;
- A description of remediation activities (if any) performed during FSS;
- Results and discussion of any ALARA evaluations, if performed;

- A discussion of the survey design (combination of scans and fixed measurements used; number of measurements; grid spacing; etc.);
- Tabular and graphical depictions of survey results including quality control results;
- Discussions of data assessments; and,
- A statement that the survey unit meets the applicable release criteria.

As noted in Section 5.3, FSS results will be documented and made available to the NRC for multiple survey areas rather than for individual survey units. Reports will be compiled after FSS activities for all the survey units within a given area are completed. These reports will be prepared and submitted per NSS administrative procedures.

When a survey unit fails, the FSS report will include the following:

- A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity;
- A summary of the investigation conducted to ascertain the reason for the failure;
- A summary of the effect that the failure has on the conclusion that the facility is ready for final radiological surveys; and,
- A summary of the effect of the failure has on other survey unit information.

5.11 Quality Assurance and Quality Control Measures

MARAD implements a comprehensive DQAP [Reference 5-6] to assure conformance with established NRC requirements and accepted industry standards. All activities, including FSS activities, performed at the NSS are required to meet the requirements of the DQAP. The participants in the DQAP ensure that the FSS activities are performed in a safe and effective manner.

The DQAP makes no distinction between performing FSS activities and performing any other activities in the scope of the DQAP. All activities in the scope of the DQAP are performed in accordance with the DQAP. The function of FSS is an activity in the scope of the DQAP.

Quality control (QC) and quality assurance (QA) measures are integrated into all decommissioning activities, including implementation of the FSS. All FSS activities essential to data quality will be implemented and performed under approved procedures. Effective implementation of administrative controls will be verified through self-assessments, monitoring and audit activities, with corrective actions being prescribed, implemented and verified in the event any deficiencies are identified. These measures apply to the related services provided by off-site vendors. Note that self-assessments are performed by individuals with direct responsibilities in the area they assess. Audits and monitoring are performed by individuals with no direct responsibilities in the area they are auditing or monitoring.

QA/QC activities for the FSS effort will serve to ensure that surveys are performed by trained individuals using approved written procedures and properly calibrated instruments that are sensitive to the suspected contaminant. In addition, QC measures will be taken to obtain quantitative information to demonstrate that measurement results have the required precision and are sufficiently free of errors to accurately represent the site being investigated. QC checks will be performed as prescribed by the implementing procedures for field measurements. For field measurements, replicate measurements in each survey unit will be made for randomly chosen location(s) of the original measurements by either a different technician or by a different instrument.

5.12 References

5-1 Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, Rev. 2, July 2019

5-2 NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, Rev. 2, April 2018

5-3 Code of Federal Regulations, Title 10, Part 20.1402, *Radiological Criteria for Unrestricted Use*

5-4 NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, Revision 1, dated August 2000

5-5 NUREG-1757 *Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria* Vol. 2, Revision. 1, dated September 2006

5-6 STS-003-001, *Decommissioning Quality Assurance Plan*, Revision 3

5-7 CR-109, *Radiological Characterization - Reactor Compartment and Containment Vessel*, Revision 1, February 2020

5-8 NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, Revision 1, August 2020

5-9 CR-143, *MARSSIM Survey of the Exterior Hull*, December 2020

6 COMPLIANCE with the RADIOLOGICAL CRITERIA for LICENSE TERMINATION

6.1 Introduction

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(E), the guidance of *Regulatory Guide 1.179, Standard Format and Content for License Termination Plans for Nuclear Power Reactors* [Reference 6-1] and the guidance in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, [Reference 6-2], this chapter describes the methods by which MARAD will demonstrate compliance with the radiological criteria for license termination with unrestricted release of the site.

References 6-1 and 6-2 describe the content required for this chapter. Summarizing from these references, the LTP must clearly present the radiological criteria proposed for license termination. If a licensee requests unrestricted release of the site in accordance with Subpart E of 10 CFR Part 20, then the LTP should demonstrate that the dose from residual radioactivity that is distinguishable from background radiation does not exceed 25 millirem (mrem) (0.25 millisievert (mSv)) per year to an average member of the critical group from all appropriate pathways over a 1,000-year period, including the methods and assumptions used to demonstrate compliance. The LTP should also demonstrate that residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).

With respect to the long-term dose period from residual radioactivity, Section 4.2.2 of NUREG-1496, Vol 1 (*Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities* [Reference 6-3] states that the 1,000-year period applies to soils and assumes a 70-year lifespan period for buildings. Section 5.2 of the LTP describes the features and characteristics of the NSS site in some detail and emphasizes how the NSS differs from typical land-based sites. NSS neither contains soil, nor is situated in soils. Post license termination and pre-disposition, NSS is more akin to a building than a land site, therefore, MARAD considers the 70-year period to be more appropriate for NSS. MARAD addresses the potential longevity of NSS after license termination in Section 6.14 of this chapter.

This chapter considers and describes three post license termination end-state scenarios for NSS; preservation, shipbreaking, and artificial reefing. The worst-case scenario from a dose perspective is immediate shipbreaking. The dose to workers and the public in the preservation end state is minimal. Furthermore, the preservation end-state benefits from continued radiological decay over time, with the result that the dose to workers from deferred shipbreaking (i.e., shipbreaking that occurs after preservation uses end) is less than that in the immediate shipbreaking end-state. Artificial reefing, as described later in the chapter, is a less-likely but plausible scenario that requires significant environmental remediation in its own right (see Section 6.3 description of artificial reefing) and would in all likelihood require the removal of any remaining reactor-generated residual radioactivity. Section 6.3.1 of this chapter provides details of MARAD's ship disposal programs to provide context to reviewers for these post license termination activities.

As described throughout the LTP, the NSS is not dismantled as part of the MARAD DECON-LT project. The ship, as the boundary element of the licensed site and nuclear facility contained therein, will remain intact after license termination, with certain structural elements and components containing residual radioactivity retained in-situ and released (see Chapter 3 for a description of structures and components expected to remain in place). Materials that exceed the license termination criteria will be removed. No waste will be disposed of onsite. For these reasons, only surface contamination Derived Concentration Guideline Levels (DCGLs) have been calculated.

Based on the potential end-state conditions described in Section 6.3, calculations have been performed to estimate potential doses to workers and members of the public from materials with potential residual surface contamination that remain on the NSS following license termination. This chapter of the LTP presents the evaluations of the impact of these remaining radioactive materials on workers and members of the public. There are eleven (11) radionuclides that have been evaluated against many scenarios including the ship in a Preservation status and activities during the Shipbreaking status. The Shipbreaking evaluation includes seven (7) handling and processing scrap steel scenarios and five (5) groundwater leachate scenarios as shown in NUREG-1640[16] [Reference 6-4]. The scenarios bound both domestic and possible international disposition of the scrap steel. The maximum dose rate coefficient from all evaluated scenarios for each radionuclide was chosen to calculate the surface contamination limit or DCGL for use in the Final Status Surveys (FSSs) of the ship.

6.2 Proposed Radiological Criteria for License Termination

MARAD intends to request unrestricted release of the NSS site in accordance with the radiological criteria for unrestricted release specified in Title 10, Section 20.1402, of the Code of Federal Regulations (10 CFR 20.1402). The criteria are:

- Dose Criterion: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/year, including that from groundwater sources of drinking water; and,
- ALARA Criterion: The residual radioactivity has been reduced to levels that are As Low As Reasonably Achievable (ALARA).

Chapter 4, Sections 4.1 and 4.4 of the LTP describe the methods and results for demonstrating compliance with the ALARA Criterion. This Chapter describes the methods for demonstrating compliance with the Dose Criterion. As noted in Chapters 1 and 4, MARAD has adopted a TEDE of 15 mrem/year rather than the prescribed 25 mrem/year. Two factors influenced this decision. The first, and of lesser importance, is to demonstrate ALARA compliance by use of the predetermined compliance measure, as described in Chapter 4. The second, and of greater importance, is the recognition by MARAD that after the license is terminated, the disposition methods for the remaining site, i.e., the ship, are all subject to U.S. Environmental Protection Agency (EPA) regulation and enforcement (see sections 6.3 and 6.3.1 of this chapter). By selecting the 15 mrem/year standard employed by EPA, MARAD hopes to avoid jurisdictional conflict after NRC terminates the NS-1 license.

Also, regarding the dose criterion, please note that as described throughout the LTP, the NSS is not a land-based site and is isolated from the terrestrial environment. Consequently, the site cannot be a groundwater source of drinking water, and thus that portion of the dose criterion is not applicable to NSS.

6.3 Potential End State Conditions

Once the NRC license is terminated, and NSS is released without restrictions, MARAD will be free to dispose of the ship. As described elsewhere in this LTP, NSS is a federally owned NHL. Its disposition is thus an integral part of the overarching undertaking being considered by MARAD and NRC under the NHPA. The disposition process that MARAD will employ is detailed in the 2023 Programmatic Agreement (PA), to which both MARAD and NRC are signatories. The disposition, regardless of its outcome, is itself a controlled process executed under MARAD control, with regulatory oversight by the

[16] Use of NUREG-1640 is consistent with guidance in NUREG 1757, Vol 2, Rev. 1 Appendix I pages I-27 and I-28 See 5.0 *Scenarios Evaluated* for full discussion.

EPA through enforcement of the Toxic Substances Control Act (TSCA) because of the presence of polychlorinated biphenyls (PCBs) in regulated quantities dispersed throughout the ship.

The PA disposition process is based on the existing statutory methods available to MARAD. These include 1) physical destruction and recycling of the ship through shipbreaking; 2) beneficial reuse by sinking the vessel in shallow water to form, or act as part of an existing, artificial reef; and 3) preservation for public use. Each of these methods is evaluated in this chapter. For the purposes of the evaluation, the methods are defined as follows:

Shipbreaking

The Occupational Safety and Health Administration (OSHA) defines shipbreaking[17] as: "... *the process of dismantling an obsolete vessel's structure for scrapping or disposal. Conducted at a pier, drydock or dismantling slip, it includes a wide range of activities, from removing all gear and equipment to cutting down and recycling the ship's infrastructure.*" Shipbreaking is a complex and hazardous activity, with workers exposed to hazardous materials (lead (Pb), asbestos, PCBs, and others), hazardous working conditions, and hazardous work activities. OSHA standards at 29 CFR 1915 are intended to protect workers from the industry-specific nature of these hazards. In addition, OSHA maintains a National Emphasis Program[18] to reduce and eliminate workplace hazards in shipbreaking operations. This program was renewed following a 2015 Memorandum of Understanding among OSHA, EPA, the US Navy and MARAD to coordinate and share information on domestic ship recycling activities. Both OSHA and EPA have prepared guidelines for use by shipbreaking facilities that address key aspects of 1) environmental compliance and 2) protection of worker safety and health. MARAD was a member of the interagency working group that helped develop the guideline documents. In short, shipbreaking is a domestic industry that has substantial, continuing federal oversight and enforcement.

Reefing

The National Oceanographic and Atmospheric Administration (NOAA) is responsible for developing and maintaining the long-term National Artificial Reef Plan, as provided in the National Fishing Enhancement Act (Act) of 1984 (33 U.S.C. § 2101 et seq.). NOAA first published the plan in 1985. The plan was comprehensively revised and republished in February 2007. The plan defines the term "artificial reef" as ". . . a structure which is constructed or placed in waters covered under this title for the purpose of enhancing fishery resources and commercial and recreational fishing opportunities." The term "waters" covered under this chapter is defined as ". . . the navigable waters of the United States and the waters superjacent to the outer continental shelf as defined in Section 2 of the Outer Continental Shelf Lands Act . . . to the extent such waters exist in or are adjacent to any State." Based on the best scientific information available, artificial reefs in waters covered under the Act "... shall be sited and constructed, and subsequently monitored and managed in a manner which will:

1) enhance fishery resources to the maximum extent practicable;
2) facilitate access and use by US recreational and commercial fishermen;
3) minimize conflicts among competing uses of waters covered under this title and the resources in such waters;

[17] See OSHA "Shipbreaking Safety Fact Sheet"
[18] See OSHA CPL 03-00-020 dated March 7, 2016.

4) minimize environmental risks and risks to personal health and property; and

5) be consistent with generally accepted principles of international law and shall not create any unreasonable obstruction to navigation."

Although not a focus of the 1985 reef plan, sport diving has developed substantially in the past few decades and now rivals the impact of recreational fishing. The 2007 amended reef plan added sport diver enhancement and access to the design considerations for artificial reefs. Similarly, the use of reefs as mitigative strategies against the loss or degradation of marine habitats and aquatic resources has been added to the 2007 plan.

Artificial reefing programs are generally administered by states. The Army Corps of Engineers has permit authority for the construction and placement of an artificial reef under Section 404 of the Clean Water Act (CWA) (33 U.S.C. §1344), and Section 103 of the Marine Protection, Research and Sanctuaries Act (MPRSA) (33 U.S.C. §1413). In each case, the EPA provides guidance on the application of the statutes.

Obsolete ships may be employed in the creation or expansion of an artificial reef. When this is the case, the ship is normally prepared in accordance with the joint MARAD-EPA *National Guidance: Best Management Practices for Preparing Vessels Intended to Create Artificial Reefs* [Reference 6-5] which includes radioactive materials in its list of "other materials of environmental concern" that should be removed when preparing a ship for reefing. As a matter of policy, MARAD does not make ships constructed prior to 1985 available for the artificial reef program, based on the presumed presence of PCBs in ships constructed before that date.

Preservation

Preservation is any prospective non-transportation use of the ship that involves unrestricted public access (e.g., museum, conference center, entertainment venue, educational facility, etc.). The ship may be operated by MARAD directly, but it is more likely that it would be conveyed to a third party for preservation use. Preservation scenarios are not indefinite and may at some future date result in shipbreaking. It is generally accepted that conveyance of a vessel for preservation is a distribution in commerce for the purposes of TSCA compliance. In the case of NSS, therefore, any preservation scenario either requires the federal government to retain title to the vessel, or to maintain a reversionary interest in it. This is consistent with MARAD's existing authorities for conveyance of NSS for preservation purposes and is outlined in the PA. The action ensures that TSCA compliance is guaranteed by a federal agency, and that MARAD, acting in its capacity as an agent for federal ship disposal (see 6.3.1 below), will undertake any future shipbreaking activity.

6.3.1 MARAD Ship Disposal Program

MARAD is the disposal agent for surplus federally owned merchant-type vessels of 1,500 gross tons[19] or more, as provided in the Federal Property and Administrative Services Act of 1949 (40 U.S.C. § 548). Since 2001, MARAD's Office of Ship Disposal Programs is responsible for the disposition of MARAD-owned obsolete vessels and for other qualifying federally owned vessels transferred to MARAD for disposal. Prior to 2001, ships were disposed via brokers to both domestic and international shipbreaking facilities. Except for four (4) vessels sold to an English firm in a 2003 pilot project, the export of MARAD obsolete ships for scrapping physically ended in 1994. The primary method employed by MARAD to dispose of obsolete ships is shipbreaking, either through competitive sales, or fee for service

[19] Gross tonnage is a measure of volume (one ton equals 100 cubic feet) used for a variety of regulatory purposes. NSS measures 15,590 gross tons

contracts. The choice of method is heavily dependent on external market factors, such as the price of scrap steel. The proceeds of ship sales are deposited in MARAD's Vessel Operations Revolving Fund (VORF),[20] consequently the program is expected to maximize the proceeds derived from its activities. To ensure consistency in its approach, the program makes little distinction between competitive sales and fee for service contracts from a technical standpoint and uses a Federal Acquisition Regulation (FAR) Best Value approach to making awards.

MARAD works closely with EPA, OSHA, and the U.S. Navy (USN) to monitor domestic ship recycling facilities. MARAD pre-qualifies recycling facilities to ensure that such facilities meet environmental and occupational safety and health standards. Only qualified facilities may bid for MARAD-sponsored ship recycling opportunities, whether by competitive sales or fee for service contracts. Ship Disposal Program staff act as Contracting Officer's Representatives to monitor the performance of work, and MARAD Office of Environmental Compliance staff periodically inspect and monitor contractor facilities and performance.

A similar technical approach is taken to disposal via artificial reefing, except that MARAD is not responsible for the costs for vessel preparation and sinking. Over the twenty-year period from FY 2001 to FY 2020, MARAD has disposed 231 ships.[21] Only four (4) of these ships have been reefed, and the most recent such project was in 2010. Similarly, only one (1) vessel has been donated (preservation) in that timeframe. The Navy's deep-ocean live-fire sinking exercises, known as SINKEX, are another low-frequency method of ship disposal. Any vessel proposed for SINKEX must be prepared to the same standards as those for artificial reefing. Candidate vessels are chosen by the USN and have exclusively been drawn from former naval auxiliaries and demilitarized former combatant vessels. Since 2001, no merchant vessels have been employed in a SINKEX and only two (2) vessels have been disposed in this manner by MARAD, both in FY 2005.

As described above, future shipbreaking of NSS, whether immediately following LT, or at some deferred date after a period of preservation use, will be performed by MARAD[22] in accordance with its established processes and procedures for vessel disposal. This activity will be a controlled industrial process with independent regulatory oversight by EPA and OSHA. Similar processes would be involved in the less likely but plausible event that NSS is reefed. Disposition for preservation will be considered in its own right under the PA and may deviate from the current MARAD donation process. Regardless of the methods employed, preservation will also be a controlled process that involves EPA from the standpoint of TSCA compliance and will require a continued federal interest in the ship.

6.3.1.1 Foreign Shipbreaking of MARAD Vessels

MARAD's current ship disposal program was established after passage of the Floyd D. Spence National Defense Authorization Act for Fiscal Year 2001, Pub. L. No. 106-398, Appendix, § 3502, 114 Stat. 1654 (2000). Among its provisions was authorization of pilot projects for foreign disposal. Given the lack of domestic shipbreaking capacity at the time, foreign disposal was viewed as a likely necessary action to remove the growing backlog of ships awaiting disposal (at the time, the backlog was over 150 ships). MARAD's efforts to export obsolete ships for disposal were unsuccessful. In its June 2004 progress

[20] The VORF account has three subaccounts which support MARAD operations (50%), maritime education and training (25%), and maritime heritage projects (25%, of which approximately ¾ are transferred to the National Park Service to award and administer National Maritime Heritage Grants). The distribution is specified by the National Maritime Heritage Act of 1994, as amended.

[21] Source – MARAD Office of Ship Disposal Programs Annual Report for FY 2020, January 2021.

[22] MARAD acknowledges that over a seventy-year or greater timeframe, there may be reorganizations that affect MARAD. MARAD presumes that any such reorganization will not eliminate the need for federal ship disposal, and that the extent federal obligations for ship disposal will be considered in such a reorganization.

report to Congress, MARAD concluded that foreign disposal was not practicable, and, because the underlying conditions have not changed, it has not attempted foreign disposal since that time. The conclusion from that report is repeated below.

> *Based upon years of futile attempts to export ships for recycling and MARAD's experience with the pilot project to send ships to AbleUK, MARAD has concluded that foreign vessel disposal is commercially impractical under current U.S. law and regulation. The central barrier to the export of MARAD vessels is the Toxic Substances Control Act (TSCA) and the prohibition on export of polychlorinated biphenyls (PCBs), 15 U.S.C. §2605(e). As demonstrated by the current litigation surrounding the AbleUK contract, only an exemption established by a full rulemaking process would enable vessels to be disposed overseas. The length of time required to pursue such a rulemaking (likely a year or more), the uncertainty of receiving an exemption, and the restrictive nature of the exemption -- valid for only one year and limited to a specific export contract (such as ships going to the AbleUK facility), make overseas recycling untenable given the existing statutory constraints.*

6.3.2 Ship Conditions at License Termination

The preservation end state is defined above as "any prospective use of the ship that involves unrestricted public access (museum, conference center, entertainment venue or educational facility)." Although the future configuration of the ship is dependent on satisfactory final status and confirmatory surveys, it is useful to contemplate the practical considerations of what structures containing residual radioactivity are likely to be present at license termination and how those structures might actually contribute to the use of the ship in the preservation scenario. The preservation scenario is considered herein because the structures and components are removed in the shipbreaking scenario, and they will most likely be removed if the ship is reefed.

Decades of environmental monitoring and reporting clearly show that exposure inside NSS and outside of former RCAs is indistinguishable from background. MARAD expects that this condition will be confirmed as part of license termination. Therefore, it is reasonable to project that the only likely source for individual exposure in the preservation end state will be from residual structures inside the former RCAs. License Amendment 8 was issued in 1976 and established the Possession-only License. In its Safety Evaluation, the Office of Nuclear Reactor Regulation discussed the condition of the ship in a manner that supports both the historical record and MARAD's expectations. The Safety Evaluation Report for License Amendment 8 stated in part *"... the survey results demonstrate that all areas of the ship, external to the containment vessel, have surface contamination levels that are significantly less than the areas acceptable for release to unrestricted access (RG 1.86). Also, only six (6) areas of the ship external to the containment vessel showed any direct radiation above background (0.02mr/hr)."* The six (6) areas listed were the Hot Chemistry Lab, Port and Starboard Charge Pump Rooms, Port Stabilizer Room, Lower Secondary Room, and Forward Control Room.[23] These six spaces along with the Health Physics Lab, the Reactor Compartment Upper Level, Containment Vessel and Starboard Stabilizer Room, were designated as RCAs prior to post dismantlement surveys. Note that the Starboard Stabilizer Room was designated because a cross flooding duct connected it to the Port Stabilizer Room, in which contaminated systems and equipment were located. The Health Physics Lab became an RCA when the 1981 museum-era Technical Specifications came into effect. Those Technical Specifications contained a very restrictive definition of RCA which was intended to prevent inadvertent public access to radioactive

[23] Since 2006, the Lower Secondary Room has been referred to as the Reactor Compartment Lower Level (RCLL). The Forward Control Room which includes on C-Deck the Cold Water Chemistry Lab (port) and Radiation Monitoring Room (starboard) plus on D-deck the Gas Adsorption Equipment Room (port) and Radiation Sampling Room (starboard) is colloquially referred to as the Cold Water Chemistry Lab (CCL).

material[24] by posting and securing spaces that might not otherwise have been controlled. The Health Physics Lab is an example of such a space; it had been used as a MARAD office space prior to the ship's transfer to the PPDA museum. Radioactive material inside the lab was confined to a sink drain. The lab is located inside the hospital complex, which was on the PPDA public self-guided tour route. License Amendment 14 removed this restrictive definition in favor of the definitions contained in 10 CFR 20. Surveys after the drain piping was removed during decommissioning confirmed the space was no longer an RCA.

Outside of the Reactor Compartment itself, the several former RCAs are positioned in such a way that they are unlikely to contribute to the preservation condition in anything other than an incidental manner. The components, equipment, and systems contained in these spaces have been dismantled as of the date of LTP submittal. Any remaining contamination is located on the steel structure of the compartments and is subject to further remediation before FSS. This would be the source for any residual radioactivity below the DCGLs at license termination. Among these spaces, only the Stabilizer Rooms offer any interest from an interpretive standpoint.[25] These spaces are inconvenient to access and would be unlikely to be part of any self-guided tour of the ship. "Behind the scenes" guided tours might be conducted but would be unlikely to involve more than a few minutes of exposure on any such event. Other spaces, such as the Port and Starboard Charge Pump Rooms, would be accessed periodically for inspection and survey, including entry into the double bottom tanks below the pump rooms.[26]

This leaves the Reactor Compartment, including the Containment Vessel, as the most likely source of exposure. Not coincidentally, these are the spaces which hold the greatest interest and which will be occupied for the longest amount of time in any given year. They will also contain the bulk of residual reactor-generated detectable radioactivity, in the form of retained structures. At the time of LTP submittal, MARAD expects that the retained structures and components will include the upper section of the pressurizer shell, the common exterior annular wall of the Fuel Transfer Tank and Neutron Shield Tank (including its exterior lead shielding), and the secondary side of both Steam Generator assemblies. Gratings and ladders inside the CV are required by MARAD's integrated services contract to be free-released or replaced. The final decisions on retained structures and components were not made at the time of LTP submittal. Using the 2018 CV Portal as a means of access, MARAD envisions that guided and self-guided tours of the CV will feature in the preservation end-state scenario. This is the basis for the representative "Tour Guide on Ship" dose model described in Section 6.9.1.

6.4 Radionuclides for Evaluation

As described in Chapter 2 of the LTP, a significant effort was undertaken in 2019 to perform a radiological and environmental hazard characterization for the RC and CV. It is documented in CR-109 [Reference 6-6]. The surveys and sampling consisted of area baseline surveys of the RC and CV areas and survey and sampling of component interior surfaces.

Primary systems and waste treatment systems were sampled extensively. In accordance with the Decommissioning Quality Assurance Plan and the Radiation Protection Program, composite smears from twelve (12) locations were sent to GEL Laboratories, LLC in South Carolina for gamma spectrometry.

[24] License Amendment 9 introduced TS 3.4, Radiological Criteria for Radiation Control Areas. Amendment 9 is sometimes referred to as the "museum ship" amendment and was designed to allow for unrestricted visitation by the public. TS 3.3 defined a radiation control area as "an area of the ship with radiation levels from reactor generated radioactive materials in excess of 0.25mR/hr above natural background as measured at one meter from any surface, and/or surface contamination in excess of the limits prescribed in Table I of NRC Reg. Guide 1.86."
[25] These spaces contain the operating machinery for the ship's gyrofin stabilizers.
[26] Note that double bottom tank entry is infrequent and is normally performed during a drydocking availability.

Five (5) of those composite smear samples were also analyzed for HTD radionuclides. In addition, a sludge sample was also sent for HTD analysis. Table 6-1 presents the samples sent for HTD radionuclides.

Table 6-1 Samples for Hard to Detect Analyses

Sample ID	Location of Sample
RCCV-RS-001	Smears from Ion Exchange piping
RCCV-RS-005	Smears from Containment Drain Tank PD-T4
RCCV-RS-007	Smears from Pressurizer
RCCV-RS-010	Smears from Steam generator
RCCV-RS-012	Smears from Ventilation System
RCLL-RM-001A	Sludge sample from Makeup Storage Tank PD-T2

The initial suite of radionuclides is presented in Table 6-2.

Table 6-2 Initial Suite of Radionuclides

Nuclide	T 1/2 (years)
C-14	5.73E+03
Co-60	5.27E+00
Ni-63	1.00E+02
Sr-90	2.86E+01
Tc-99	2.13E+05
Cs-137	3.02E+01
H-3	1.23E+01
Fe-55	2.70E+00

The results of the laboratory analyses of the samples presented in Table 6-1 are presented in Table 6-3.

Table 6-3 Offsite Laboratory Results of Sample Analyses

Nuclide	SG smear composite (pCi)	PZR smear composite (pCi)	PD-T4 smear composite (pCi)	RC Exhaust Vent smear composite (pCi)	RC IX Piping smear composite (pCi)	Makeup Storage Tank Sludge (pCi/g)
C-14	3.50E+02	2.23E+03	5.10E+01	6.43E+02	1.48E+04	8.60E+03
Co-60	3.58E+04	4.26E+04	3.96E+03	2.15E+02	8.21E+04	2.35E+05
Ni-63	3.38E+07	1.79E+06	1.95E+05	1.20E+04	5.24E+06	7.70E+06
Sr-90	<MDA	1.39E+00	<MDA	<MDA	1.59E+00	6.61E+01
Tc-99	7.90E+01	2.74E+01	1.42E+01	<MDA	1.67E+02	6.23E+01
Ag-108m	<MDA	<MDA	3.94E+01	1.14E+01	2.52E+02	6.91E+02
Cs-137	1.39E+02	3.39E+02	3.05E+03	9.81E+03	9.09E+02	6.97E+04
H-3	NA	NA	NA	NA	NA	1.56E+04
Fe-55	NA	NA	NA	NA	NA	6.56E+04
Am-241	NA	NA	NA	NA	NA	4.69E+00
Pu-239/240	NA	NA	NA	NA	NA	3.89E+00
Totals	**3.38E+07**	**1.84E+06**	**2.02E+05**	**2.27E+04**	**5.34E+06**	**8.10E+06**

Note: NA = not analyzed

The fraction of the total activity in each sample for each radionuclide was calculated.

For radionuclide results that were less than the MDA, the fraction of the total activity for that radionuclide was assigned a value of zero (0). The fraction of total activity for each sample is presented in Table 6-4.

Table 6-4 Radionuclide Fractions of the Sample Analyses

Nuclide	SG composite fractions	PZR composite fractions	PD-T4 composite fractions	RC exhaust vent composite fractions	RC IX piping composite fractions	Makeup Storage Tank Sludge fractions
C-14	1.03E-05	1.22E-03	2.52E-04	2.84E-02	2.77E-03	1.06E-03
Co-60	1.06E-03	2.32E-02	1.96E-02	9.48E-03	1.54E-02	2.90E-02
Ni-63	9.99E-01	9.75E-01	9.65E-01	5.29E-01	9.82E-01	9.51E-01
Sr-90	0.00E+00	7.57E-07	0.00E+00	0.00E+00	2.98E-07	8.17E-06
Tc-99	2.33E-06	1.49E-05	7.03E-05	0.00E+00	3.13E-05	7.70E-06
Ag-108m	0.00E+00	0.00E+00	1.95E-04	5.03E-04	4.72E-05	8.54E-05
Cs-137	4.11E-06	1.85E-04	1.51E-02	4.33E-01	1.70E-04	8.61E-03
H-3	-	-	-	-	-	1.93E-03
Fe-55	-	-	-	-	-	8.10E-03
Am-241	-	-	-	-	-	5.79E-07
Pu-239/240	-	-	-	-	-	4.81E-07
Totals	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**

The radionuclide fractions in Table 6-4 were summed and then normalized to obtain the radionuclide distribution and for calculation of the Radionuclides of Concern.

Table 6-5 presents the results of the calculations.

Table 6-5 Radionuclide Sum of Fractions and Normalized Sum of Fractions

Nuclide	Sum of Fractions	Normalized Sum of Fractions
C-14	3.37E-02	5.61E-03
Co-60	9.78E-02	1.63E-02
Ni-63	5.40E+00	9.00E-01
Sr-90	9.22E-06	1.54E-06
Tc-99	1.27E-04	2.11E-05
Ag-108m	8.30E-04	1.38E-04
Cs-137	4.57E-01	7.61E-02
H-3	1.93E-03	3.21E-04
Fe-55	8.10E-03	1.35E-03
Am-241	5.79E-07	9.66E-08
Pu-239/240	4.81E-07	8.01E-08
Totals	6.00E+00	1.00E+00

6.5 Documents for Guiding Calculations

Two primary documents were reviewed to provide perspective on calculating doses and generation of DCGLs. They are IAEA Safety Report Series No. 44 [Reference 6-7] which provides calculations and concentrations of radionuclides for exemption and clearance that are valid for all types of solid material containing radionuclides of artificial or natural origin except foodstuffs and drinking water. The second document is NUREG-1640 [Reference 6-4] which provides calculations of both mass-based and surficial dose rate conversion factors for scrap steel, aluminum, copper and concrete.

Both documents use a multiple scenario approach which include a foundry worker, landfill worker, residents living near a landfill using contaminated groundwater and being exposed to re-purposed products. The IAEA report presents a set of realistic parameters and low probability parameters. The realistic parameters were used to calculate a 10 μSv/y (1 mrem/y) activity concentrations and the low probability parameters were used to calculate a 1 mSv/y (100 mrem/y) activity concentrations. The lowest concentration from either calculation was then rounded to the nearest power of 10 and listed as the release value. The report describes only one (1) scenario for the foundry worker. There is no scenario for the scrap pile worker. In addition, the analyses were deterministic, meaning only single values were used for the parameters.

In contrast, the NUREG report has six (6) foundry/steel mill scenarios and a scrap yard worker scenario. Almost all parameters have distributions which allows for a probabilistic analysis. The scenarios were established based upon site visits to scrap yards and foundries, making the calculations realistic as compared to the IAEA report.

Based upon this review, NUREG-1640 was a much more robust evaluation and directly applicable to the shipbreaking scenario of the NSS.

6.6 Exposure Scenarios Evaluated

As described in Section 6.1 of this chapter, the worst-case scenario from a dose perspective is the immediate scrapping of NSS following license termination. NUREG-1640 evaluated 37 steel scrap scenarios that included handling, processing, transportation, product use, landfill disposal, groundwater leachate, and atmospheric release. Those evaluations are based on some of the steps that would most likely be involved in processing scrap steel. A review of the radionuclides present on the NSS against the list of critical groups for steel in Table 3.22 in NUREG-1640, shows that the critical groups are from the seven (7) handling and processing scenarios or from the five (5) groundwater leachate scenarios. These 12 scenarios are directly applicable to the Shipbreaking of the NSS. Table 6-6 presents the potentially exposed individuals in each phase (i.e., the critical group). Those individuals listed as significant, are those who are full time employees at the task, or their tasks require handling or manipulating potentially contaminated components. Calculations of potential doses were performed for significant individuals in the timeframe immediately following license termination. In the less likely but plausible event that NSS is reefed after license termination, the immediate shipbreaking scenario is considered a bounding scenario given the likely need, as described in Section 6.3, to remove at least some of the structures and components containing residual radioactivity as part of the ship preparations for reefing. Section 6.11 evaluates the potential exposure to recreational divers in the reefing scenario if the structures and components are not removed. Finally, Table 6-6 also includes the potentially exposed individuals in the preservation scenario, with the understanding that their potential exposures are not evaluated against NUREG-1640 (see Section 6.8 regarding the significant individual in the preservation scenario).

The use of NUREG-1640 is consistent with the plain language of Appendix I of NUREG 1757, Vol. 2, Rev. 1. Appendix I has the following purpose:

> *This appendix consists of the technical guidance for the use of the site-specific dose modeling, applicable to Decommissioning Groups 4–9.*

The NSS is in Decommissioning Group 4. *Section I.3.3.3.6 Offsite Scenarios* provides the following guidance on developing scenarios:

> *Licensees can use generic analyses to screen the importance of offsite uses with such sources as NUREG-1640, "Radiological Assessments for Clearance of Materials from Nuclear Facilities." (NRC 2003) Pg I-27*

On page I-28, NUREG-1757 [Reference 6-8] provides an even broader array of guidance for developing scenarios:

> *Licensees may be able to use information from NUREG/CR-5512, NUREG-1640, and NUREG-1717, as well as other licensees' analyses to screen their potential scenarios with quantitative methods.*

Developing scenarios using NUREG-1640 is consistent with NRC guidance contained in NUREG-1757. It is understood that NRC reviewers will evaluate the appropriateness of the scenarios during their review of the LTP.

Table 6-6 Scenarios Evaluated for Calculation of Surface Contamination DCGLs

Ship Status	Description	Exposed Individual	Significance
Preservation	Office Worker/Tour Guide	Adult worker	Significant – full time employee
Preservation	Housekeeping	Adult worker	Insignificant – part time employee
Preservation	Minor repairs/ maintenance	Adult worker	Insignificant – part time employee
Preservation	Tours and meetings on ship	Members of the Public	Insignificant – few hours per year
Pre-Shipbreaking	Housekeeping	Adult worker	Insignificant – part time employee
Pre-Shipbreaking	Minor repairs/ maintenance	Adult worker	Insignificant – part time employee
Shipbreaking	Remediation of hazardous materials on ship	Adult worker	Significant – full time employee
Shipbreaking	Component removal/metal cutting on ship	Adult worker	Significant – full time employee
Shipbreaking	7 steel handling and processing scenarios defined in NUREG-1640	Adult worker	Significant – full time employee
Dismantlement complete	Groundwater infiltrated by leachate from landfills or storage piles defined in NUREG-1640	Members of the Public	Significant – potential daily exposure

6.7 Dose Rate Equations for Remediation and Component Removal Workers

The Remediation worker prepares the ship for the Component Removal worker. The Remediation worker removes fuels, oils, PCBs, asbestos, and combustible materials. Following the removal of combustible materials, asbestos and PCBs, the paint or preservative coatings must be stripped from surfaces to be cut. The Component Removal workers make the cuts on pipes and components of suitable size for removal from the ship.

The following equations are based upon those presented in NUREG-1640. They have been modified to obtain dose rates, convert gm to cm^2 and for varying contamination levels for the Remediation and Component Removal worker scenarios.

The surface area factor, f_{sa}, is used in the ingestion and inhalation calculations. It is the inverse of the mass to surface area (g/cm^2). The values used in these calculations were derived from several sources such as NUREG-1640, ANSI N13.12-2013 [Reference 6-4] and steel pipe schedule charts. Values for Schedule 40 steel pipes in pounds per foot (lb/ft) were converted to g/cm^2. Schedule 40 pipe has a thinner wall thickness than a Schedule 80 pipe and is, therefore, conservative. The mass to surface area factors for 2 inch to 12 inch diameter pipes were calculated with a range of 2.9 to 7.8 g/cm^2. This range spans the 4.53 to 5.34 g/cm^2 values in Appendix A of NUREG-1640 for converting mass-based values to surficial values for steel. Due to the large range of pipe diameters on the ship, mass to surface area factors for the 2 inch to 12 inch diameter pipes was appropriate for use in the calculations. Therefore, the surface area factor in the calculations was allowed to vary between 0.1 and 0.35 cm^2/g.

The contamination factor, f_{cf}, is used in the external exposure calculations. It accounts for varying contamination levels in pipes and components encountered during their work activities. For these calculations, the contamination levels on the pipes and components were assumed to be uniform; therefore, the contamination factor has been conservatively set to 1.

The dilution factor, f_d, is used in the inhalation calculations. It accounts for respirator use. The Remediation worker is expected to handle and remove hazardous materials and will wear a full face, negative pressure respirator with a protection factor of 50. The dilution factor for the Remediation worker is 0.02. The Component Removal worker is not expected to wear a respirator; therefore, the dilution factor for the Component Removal worker is 1.

Equation 6-1, derived from Equation 3.9 in NUREG-1640, was used to calculate the dose rate from ingestion:

$$D_{ig} = F_{ig}\, I_g\, f_{sa}\, t_e\, e^{-\lambda_i t_s} \qquad \text{Equation 6-1}$$

Where:

D_{ig}	=	*the dose rate from ingestion per unit surface activity concentration on the material for radionuclide i [(μSv/y)/(Bq/cm²)]*
F_{ig}	=	*the committed dose equivalent coefficient from FGR-11 for ingestion of radionuclide i [μSv/Bq]*
I_g	=	*the ingestion rate [g/h]*
f_{sa}	=	*the surface area factor [cm²/g]*
t_e	=	*the duration of internal exposure [h/y]*
λ_i	=	*the radioactive decay constant [1/y]*
t_s	=	*the interval from time scrap is cleared until scenario begins [y]*

Equation 6-2, derived from Equation 3.8 in NUREG-1640, was used to calculate the dose rate from inhalation:

$$D_{ih} = F_{ih}\, R_h\, f_{sa}\, f_d\, t_e\, \chi_d\, e^{-\lambda_i t_s} \qquad \text{Equation 6-2}$$

Where:

D_{ih}	=	*the dose rate from inhalation per unit surface activity concentration on the material for radionuclide i [(μSv/y)/(Bq/cm²)]*
F_{ih}	=	*the committed dose equivalent coefficient from FGR-11 for inhalation of radionuclide i [μSv/Bq]*
R_h	=	*the inhalation rate [m³/h]*
f_{sa}	=	*the surface area factor [cm²/g]*

f_d	=	*the dilution factor used for wearing respirators [-]*
t_e	=	*the duration of internal exposure [h/y]*
χ_d	=	*the effective dust concentration in the air [g/m³]*
λ_i	=	*the radioactive decay constant [1/y]*
t_s	=	*the interval from time scrap is cleared until scenario begins [y]*

Equation 6-3, derived from Equation 3.6 in NUREG-1640, was used to calculate the dose rate from external exposure:

$$E_{ext}^{i} = e_{ext}^{i}\, t_e\, f_{cf}\, U_x e^{-\lambda_i t_s} \qquad \text{Equation 6-3}$$

Where:

E_{ext}^{i}	=	*the dose rate from external exposure per unit surface activity concentration on the material for radionuclide i [(μSv/y)/(Bq/cm²)]*
e_{ext}^{i}	=	*the effective dose equivalent rate per unit surface activity concentration on the material, for radionuclide i generated by MicroShield [(mSv/h)/(Bq/cm²)]*
t_e	=	*the exposure time [h/y]*
f_{cf}	=	*the contamination factor [-]*
U_x	=	*the uncertainty factor, which accounts for the variation of dose rate with position [-]*
λ_i	=	*the radioactive decay constant [1/y]*
t_s	=	*the interval from time scrap is cleared until scenario begins [y]*

6.8 Tools Used for Calculations and Analyses

The calculated surface contamination values in the Remediation worker and Component Removal worker scenarios were developed using a computerized risk analysis modeling tool, "ModelRisk 4.0", developed by Vose software. This analysis tool is a 3rd party add-on to Microsoft's Excel spreadsheet program. This tool allows for a variety of analysis and distribution propagations including:

- Monte Carlo Simulations of the sampling and propagation of a variety of distributions,
- Correlations of parameter values from data sets,
- Creation of empirical or pre-defined distributions from data sets,

An unlimited number of parameters contained within a standard spreadsheet calculation can be identified as distributions (either pre-defined or empirical) with this tool. Once the parameters have been appropriately defined and selected, and once the calculation framework (i.e., formulas) has been completed, the simulation can be performed. The number of iterations selected for executing the simulations in this analysis was set to 10,000 iterations. The 95th percentile was selected for the assessments of the Remediation and Component Removal workers. This provides the precision and confidence levels for good approximations to the theoretical distribution of potential doses.

RESRAD-BUILD code, Version 3.5 was used to calculate the dose rate coefficients for the Tour Guide on Ship scenario. Uncertainty analyses were run with the built-in distribution parameters for breathing rate, ingestion rate and resuspension rates allowed to vary. The 95th percentile value for each radionuclide was chosen for the calculations.

For the steel scrap scenarios in NUREG-1640, the effective dose equivalent rate coefficients have been calculated by Monte Carlo simulations. For parameters that are uncertain or variable, Monte Carlo sampling methods were used to pick the particular set of values in a given calculation, called a realization. In the NUREG-1640 analyses, the estimation of each dose, radionuclide concentration, or other intermediate parameter involved between 5,800 and 10,000 realizations. The 95th percentile was used for assessments of the Scrap Yard and Foundry workers and members of the public from groundwater leachate.

6.9 Inputs to the Scenario Calculations

As previously stated, the components being evaluated here are those that do not exceed the FSS release limits. These components will remain installed even though they may have measurable levels of radioactivity that are less than the DCGLs. The impact of these radioactive components on workers and members of the public is evaluated in the following sections.

6.9.1 Tour Guide on Ship

In this scenario, the tour guide is assumed to spend 8-hours per workday for 250 days per year in the CV and RC. The RESRAD-BUILD Version 3.5 model assumes the room size to be approximately equal to the size of the CV: 12 meters wide by 12 meters long. Five large area circular sources with a radius of 5 meters each were generated. One source is on each bulkhead at the midpoint, and one is on the deck in the middle. The Tour Guide is assumed to be standing on the source on the deck. The dose point for the calculations is 1 meter above the deck. Figure 6-1 is the RESRAD-BUILD depiction of the relationship between the sources and receptor (i.e., Tour Guide).

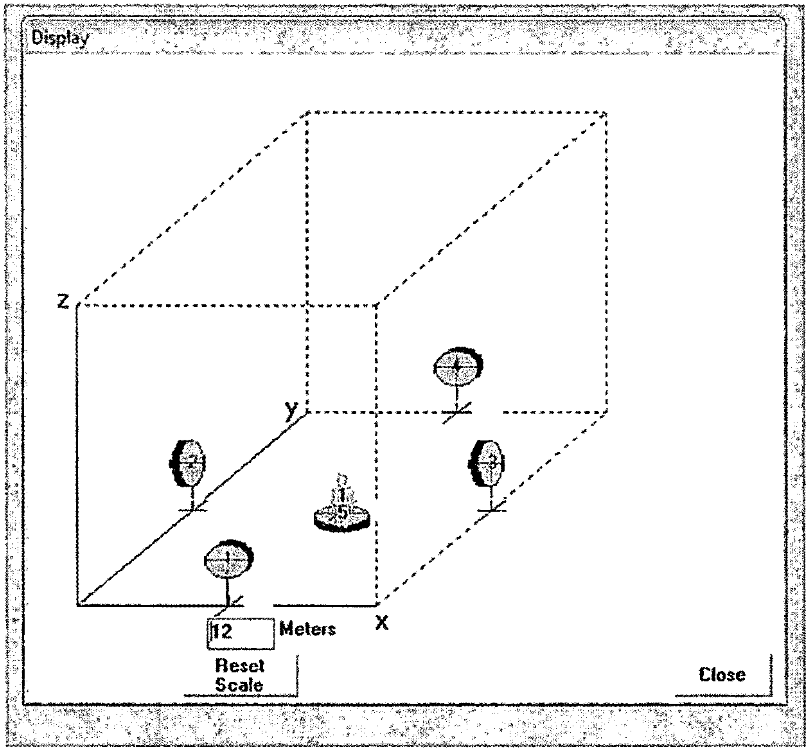

Figure 6-1 Receptor and Sources for RESRAD-BUILD Model

The initial runs were deterministic with default values for all parameters except the indoor fraction, which was set to 0.23. These initial runs were performed to get an estimate as to the general magnitude of the results. Uncertainty analyses were then run with the built-in distribution parameters for breathing rate, ingestion rate and resuspension rates allowed to vary. The 95th percentile value for each radionuclide was chosen for the calculations. The results with the uncertainty analyses were compared against the results of the other scenarios and showed that the Tour Guide dose rate coefficients were orders of magnitude lower than the other scenarios. Further modifications to the input parameters could not be justified. Parameters for Tour Guide on Ship Calculations are presented in Table 6-7.

Table 6-7 Parameters for Tour Guide on Ship Calculations

Parameter	Units	Value	Notes
Indoor Fraction	unitless	0.23	Equal to 2000 h/y / 8760 h/y
Sources and strength	dpm/m^2	1	Each source is a 5m radius disk on 4 bulkheads and the deck. Receptor is on the deck source.
Breathing Rate	m^3/d	Varies	Triangular. min = 12, mode = 33.6 and max = 44
Ingestion Rate	m^2/h	Varies	Loguniform. min = 0.000028, max =0.00029
Resuspension Rate	1/s	Varies	Loguniform. min = 2.8E-10, max =0.000014
Remaining parameters			Default values

The RESRAD-BUILD reports are attachments to CR-139, TSD No. 21-089, Rev. 0, *Calculations to Support NS Savannah Surface Contamination DCGLs* [Reference 6-9].

6.9.2 Remediation Worker on Ship

External Exposure

After license termination, the first step in the shipbreaking process is to remove any hazardous materials from the ship. The individual performing this task is identified as a "Remediation worker." The Remediation worker on Ship scenario assumes that the greatest external exposure occurs during removal of asbestos and other hazardous materials from components and pipes.

A cylinder surface – external dose point model in MicroShield® Version 8.03 was developed for the source of exposure. The length of the pipe was 10 feet with the dose rate at 1 foot from the pipe was calculated to simulate the location of the worker. Three pipe diameters were modeled: 2," 4" and 12" diameter pipes. The source activity concentration was 1 Bq/cm^2 for each of the radionuclides. The thickness of a Schedule 40 pipe was used as the shield. Schedule 40 pipe has a thinner wall thickness than a Schedule 80 pipe and is, therefore, conservative. Cobalt-60, Ag-108m and Cs-137 are the only gamma emitters in the list of radionuclides on the ship. The largest effective dose coefficient is from the 12" diameter pipe and was used in the calculations. The Effective Dose Equivalent Rate, Anterior/Posterior Geometry with build-up was used to calculate the worker dose. The worker could be cutting a pipe with another pipe nearby; therefore, the Effective Dose Equivalent rate was multiplied by two (2) for conservatism.

Equation 6-3 was used to calculate the dose rate from external exposure per unit surface activity concentration in $(\mu Sv/y)/(Bq/cm^2)$. Table 6-8 presents the values used in the calculations.

Table 6-8 Parameters for Remediation Worker on Ship External Dose Calculations

Parameter	Symbol	Units	Value	Notes
Exposure time	t_e	h/y	1500	Assumed 6 h/d for 250 days. 6 h/d equals the maximum daily exposure duration for the Scrap Yard worker in Table B.8 of NUREG-1640.
Contamination factor	f_{cf}	-	1	Assumes uniform contamination levels
Uncertainty factor	U_x	-	1	Assumed no uncertainty
Interval from time scrap is cleared until scenario begins	t_s	y	0	Assumed no decay
Lambda	λ_i	1/y		From NRC Toolbox version 3.0.0
Effective Dose Equivalent Rate Coefficient	e_{ext}	(mSv/h)/(Bq/cm^2)		Effective dose equivalent rate, Anterior/Posterior Geometry with build-up from MicroShield

The MicroShield® computer code reports are attachments to CR-139, TSD No. 21-089, Rev. 0, Calculations to Support NS Savannah Surface Contamination DCGLs [Reference 6-9].

Ingestion

The rate of secondary ingestion is a highly uncertain parameter. A minimum value of the ingestion rate is zero. This corresponds to the ingestion by workers practicing good industrial hygiene. Company policies that prohibit eating or smoking in the workplace, along with wearing gloves, would minimize hand-to-mouth transfer.

For this parameter, the values presented in NUREG-1640 Appendix B, Table B.8 were used. The ingestion rate is a uniform distribution with a min at 0.0 and a max at 0.02 g/h.

The values for the committed dose equivalent coefficients for ingestion were taken from Federal Guidance Report No. 11 (FGR-11) [Reference 6-10].

Equation 6-1 was used to calculate the dose rate from ingestion per unit surface activity concentration in (μSv/y)/(Bq/cm^2). Table 6-9 presents the values used in the calculations.

Table 6-9 Parameters for Remediation Worker on Ship Ingestion Dose Calculations

Parameter	Symbol	Units	Value	Notes
Inadvertent (Secondary) ingestion rate	I_g	g/h	variable	NUREG-1640 App B, Table B.8. Uniform distribution with a min at 0.0 and a max at 0.02 g/h.
Exposure duration	t_e	h/y	1500	Assumed 6 h/d for 250 days. 6 h/d equals the maximum daily exposure duration for the Scrap Yard worker in Table B.8 of NUREG-1640.
Surface area factor	f_{SA}	cm^2/g	variable	ModelRisk Pert distribution with max at 0.35, min at 0.1 and mode at 0.2 based on mass/length data for Schedule 40 pipe.
Interval from time scrap is cleared until scenario begins	t_s	y	0	Assumed no decay.
Lambda	λ_i	1/y		From NRC Toolbox version 3.0.0.
Committed Dose Equivalent Coefficient	F_{ig}	μSv/Bq		Most conservative Dose Conversion Factors (DCFs) from FGR-11.

Inhalation

Inhalation rates vary with physical activity. For this parameter, the values presented in NUREG-1640 Appendix B, Table B.8 were used. The inhalation rate is a triangular distribution with mode at 1.2, min at 0.6 and max at 1.8 m^3/h.

Airborne dust concentrations vary with ventilation rates and work activities. For this parameter, the values presented in NUREG-1640 Appendix B, Table B.8 were used. The airborne mass loading of material is a lognormal distribution with the mean at 2.433, the standard deviation at 1.27 and a max at 5 mg/m^3.

The values for the committed dose equivalent coefficients for inhalation were taken from FGR-11.

Equation 6-2 was used to calculate the dose rate from inhalation per unit surface activity concentration in $(\mu Sv/y)/(Bq/cm^2)$. Table 6-10 presents the values used in the calculations.

Table 6-10 Parameters for Remediation Worker on Ship Inhalation Dose Calculations

Parameter	Symbol	Units	Value	Notes
Inhalation rate	R_h	m^3/h	variable	NUREG-1640 App B, Table B.8. Triangular distribution with mode at 1.2, min at 0.6 and a max at 1.8 m^3/h.
Exposure duration	t_c	h/y	1500	Assumed 6 h/d for 250 days. 6 h/d equals the maximum daily exposure duration for the Scrap Yard worker in Table B.8 of NUREG-1640.
Dilution factor	f_d	-	0.02	Assumed workers are wearing respirators with a PF = 50.
Surface area factor	f_{SA}	cm^2/g	variable	ModelRisk Pert distribution with max at 0.35, min at 0.1 and mode at 0.2 based on mass/length data for Schedule 40 pipe.
Air Concentration	X_d	g/m^3	variable	NUREG-1640 App B, Table B.8. Lognormal distribution with mode at 2.433, standard deviation of 1.27, min at 0.0 and a max at 5 mg/m^3.
Interval from time scrap is cleared until scenario begins	t_s	y	0	Assumed no decay.
Lambda	λ_i	1/y		From NRC Toolbox version 3.0.0.
Committed Dose Equivalent Coefficient	F_{ih}	µSv/Bq		Most conservative DCF from FGR-11.

6.9.3 Component Removal Worker on Ship

External Exposure

After the ship has been remediated, the next step in shipbreaking is to remove piping and components and start breaking the ship up for scrap. The individual performing this task is identified as the "Component Removal worker." The component removal worker on ship scenario assumes that the greatest external exposure occurs during removal of components and pipes.

The same source, used for the Remediation worker on Ship, was used to calculate the external exposures to the Component Removal worker on Ship.

Equation 6-3 was used to calculate the dose rate from external exposure per unit surface activity concentration in (μSv/y)/(Bq/cm^2). Table 6-11 presents the values used in the calculations.

Table 6-11 Parameters for Component Removal Worker on Ship External Dose Calculations

Parameter	Symbol	Units	Value	Notes
Exposure time	t_e	h/y	1500	Assumed 6 h/d for 250 days. 6 h/d equals the maximum daily exposure duration for the Scrap Yard worker in Table B.8 of NUREG-1640.
Contamination factor	f_{cf}	-	1	Assumes uniform contamination levels.
Uncertainty factor	U_x	-	1	Assumed no uncertainty.
Interval from time scrap is cleared until scenario begins	t_s	y	0	Assumed no decay.
Lambda	λ_i	1/y		From NRC Toolbox version 3.0.0.
Effective Dose Equivalent Rate Coefficient	e_{ext}	(mSv/h)/(Bq/cm^2)		Effective dose equivalent rate, Anterior/Posterior Geometry with build-up from MicroShield.

Ingestion

The same values for secondary ingestion rates used for the Remediation worker on Ship are used for the Component Removal worker on Ship.

The same values for the effective dose coefficients for ingestion, used for the Remediation worker on Ship, are used for the Component Removal worker on Ship.

Equation 6-1 was used to calculate the dose rate from ingestion per unit surface activity concentration in (μSv/y)/(Bq/cm^2). Table 6-12 presents the values used in the calculations.

Table 6-12 Parameters for Component Removal Worker on Ship Ingestion Dose Calculations

Parameter	Symbol	Units	Value	Notes
Inadvertent (Secondary) ingestion rate	I_g	g/h	variable	NUREG-1640 App B, Table B.8. Uniform distribution with a min at 0.0 and a max at 0.02 g/h.
Exposure duration	t_e	h/y	1500	Assumed 6 h/d for 250 days. 6 h/d equals the maximum daily exposure duration for the Scrap Yard worker in Table B.8 of NUREG-1640.
Surface area factor	f_{SA}	cm^2/g	variable	ModelRisk Pert distribution with max at 0.35, min at 0.1 and mode at 0.2 based mass/length data for Schedule 40 pipe.
Interval from time scrap is cleared until scenario begins	t_s	y	0	Assumed no decay.
Lambda	λ_i	1/y		From NRC Toolbox version 3.0.0.
Committed Dose Equivalent Coefficient	F_{ig}	μSv/Bq		Most conservative DCF from FGR-11.

Inhalation

The same values for the inhalation rate used for the Remediation worker on Ship are used for the Component Removal worker on Ship.

The same airborne dust concentrations used for the Remediation worker on Ship are used for the Component Removal worker on Ship.

The same values for the effective dose coefficients for inhalation used for the Remediation worker on Ship are used for the Component Removal worker on Ship.

Equation 6-2 was used to calculate the dose rate from inhalation per unit surface activity concentration in (μSv/y)/(Bq/cm^2). Table 6-13 presents the values used in the calculations.

Table 6-13 Parameters for Component Removal Worker on Ship Inhalation Dose Calculations

Parameter	Symbol	Units	Value	Notes
Inhalation rate	R_h	m^3/h	variable	NUREG-1640 App B, Table B.8. Triangular distribution with mode at 1.2, min at 0.6 and max at 1.8 m^3/h.
Exposure duration	t_c	h/y	1500	Assumed 6 h/d for 250 days. 6 h/d equals the maximum daily exposure duration for the Scrap Yard worker in Table B.8 of NUREG-1640.
Dilution factor	f_d	-	1	Assumed workers are not wearing respirators.
Surface area factor	f_{SA}	cm^2/g	variable	ModelRisk Pert distribution with max at 0.35, min at 0.1 and mode at 0.2 based on mass/length data for Schedule 40 pipe.
Air Concentration	X_d	g/m^3	variable	NUREG-1640 App B, Table B.8. Lognormal distribution with mode at 2.433, standard deviation of 1.27, min at 0.0 and max at 5 mg/m^3.
Interval from time scrap is cleared until scenario begins	t_s	y	0	Assumed no decay.
Lambda	λ_i	1/y		From NRC Toolbox version 3.0.0.
Committed Dose Equivalent Coefficient	F_{ih}	μSv/Bq		Most conservative DCF from FGR-11.

6.9.4 Scrap Steel Scenarios in NUREG-1640

The following paragraph has been taken directly from NUREG-1640.

> *Assessments have been performed of the potential radiation doses to individuals from the recycling or disposal of iron and steel scrap that could be cleared from nuclear facilities. The assessment addresses 37 scenarios that depict exposures resulting from the handling and processing of cleared scrap and the products of melting and refining this scrap at steel mills and foundries, emission of airborne effluents from these facilities, transportation of scrap and furnace products, the use of these products, the landfill disposal of cleared scrap and furnace by-products, and the infiltration of well water by leachate from landfills and storage piles containing cleared scrap or furnace by-products. The analysis utilizes data on ferrous metal recycling, as currently practiced in the United States, and on contemporary U.S. work practices and living habits.*

Figure 3-1 from NUREG-1640 presents a schematic diagram of the flow of steel scrap, as characterized in the analysis. The analysis starts with the scrap steel released from the nuclear facility.

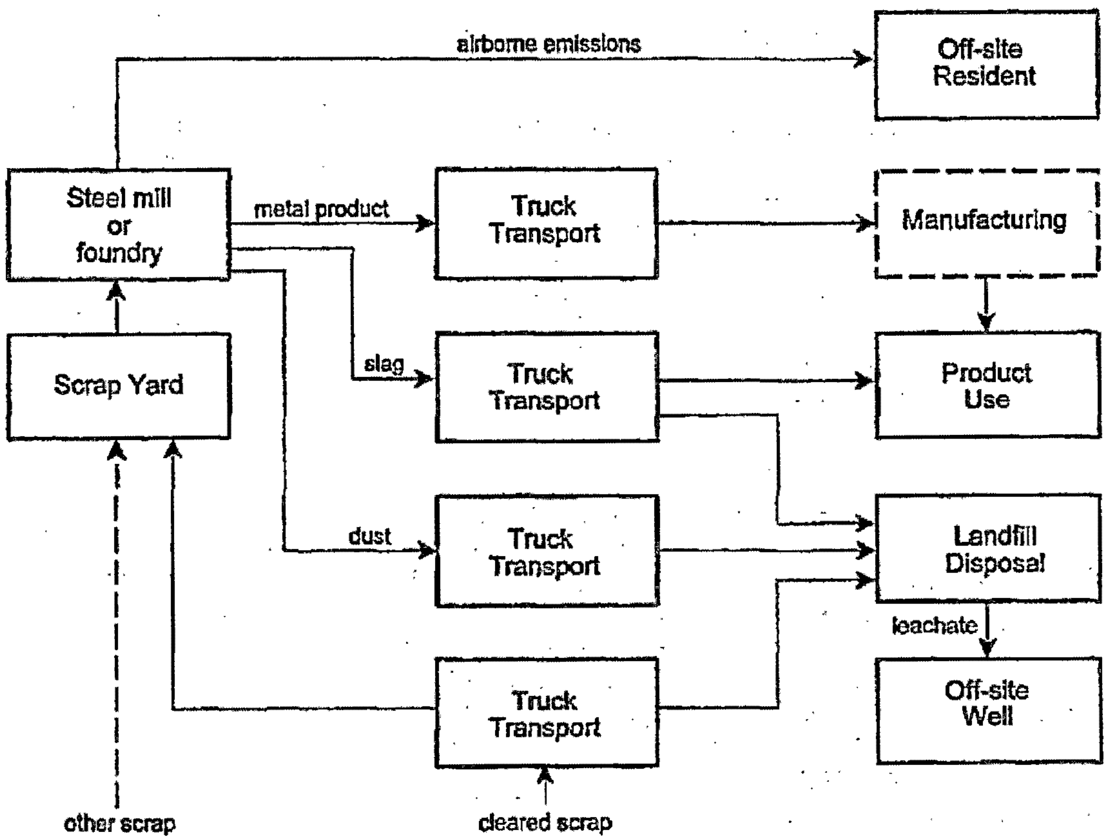

Figure 3.1 Flow of steel scrap

As previously noted, NUREG-1757 Volume 2 Section I.3.3.3.6 states:

> *Licensees can use generic analyses to screen the importance of offsite uses with such sources as NUREG-1640, "Radiological Assessments for Clearance of Materials from Nuclear Facilities." (NRC 2003).*

In the Forward to NUREG-1640 Volume 1 states:

> *The large variety of scenarios analyzed in this report may be realistically applied to certain situations in licensing requests. In other cases, the models and scenarios may be adapted to specifically fit the situation at hand. Such applications are facilitated by the explicit presentation of the exposure scenarios, calculational models, modeling parameters, and mathematical formulations in this report.*

After the Component Removal worker has completed breaking up the ship, the steel from the NSS will likely be processed in the scrap yard. Following processing, the scrap steel would be sent to a foundry/steel mill where it is mixed with non-radiologically impacted steel and processed. Activities in the steel mill include handling and processing the steel, slag and electric arc furnace (EAF) dust. Airborne emissions from the foundry also occur. Waste products from the foundry will eventually be placed into landfills where leachate from landfills could infiltrate nearby downgradient wells used as sources of drinking water. These exposure scenarios have been evaluated in NUREG-1640.

A review of the scrap yard and steel mill scenarios in Volume 1 of NUREG-1640 and the supporting model parameters in Appendix B of Volume 2 of NUREG-1640 was performed. Specific key parameters reviewed included:

Hours exposed

The daily duration of external and internal exposure for the scrap yard worker was assigned a range of 4 to 6 hours, assuming a uniform distribution, based on information obtained during visits to two scrap yards. This individual is assumed to work 250 days per year.

The daily duration of external and internal exposure for the slag worker was assigned a range of 2 to 6 hours, assuming a uniform distribution. This individual is assumed to perform other duties while being away from the slag pile. This individual is also assumed to work 250 days per year.

Inhalation Rates

NUREG-1640 states that the inhalation rates in the *1975 ICRP 23* [Reference 6-11] and *1997 EPA Exposure Factors Handbook* [Reference 6-12] were reviewed to determine a reasonable range for the analysis. The *1997 EPA Handbook* cites a number of studies performed under a variety of conditions. The study, which agrees best with the ICRP model, lists a mean adult inhalation rate of 1.2 m^3/hr for light activities. Based on these data, this parameter was assigned a triangular distribution, with a mode of 1.2 m^3/hr and a range of 0.6 to 1.8 m^3/hr.

Ingestion Rates

The *1997 EPA Handbook* cites an adult soil ingestion rate from gardening of 20 mg/hr. NUREG-1640 states this appears to be a reasonable estimate of the maximum hourly rate, averaged over one year of exposure in a work-related scenario, and is the value adopted for the exposure of the "reasonably maximally exposed individual" in the scrap metal assessments. A minimum value of the ingestion rate is zero. This corresponds to the ingestion of workers practicing good industrial hygiene. This parameter was assigned a uniform distribution of 0 to 0.02 g/hr.

External Exposure Modeling

NUREG-1640 states: Scrap piles observed by members of the project team during visits to scrap metal dealers vary widely in size and shape. A hemispherical pile was adopted for the analysis. Such a pile has rotational symmetry; thus, a scrap pile worker would receive the same dose regardless of his angular orientation with respect to the pile. Furthermore, a hemispherical pile is completely specified by only one dimension-e.g., the volume. Consequently, a hemisphere is the simplest and, therefore, the most generic shape.

NUREG-1640, Appendix C states: Bulk densities of ferrous scrap range from 16 to 22 lb/cu ft prior to compaction. Twenty pounds per cubic foot (0.32 g/cc) was adopted as the bulk density for the analysis.

The source of external exposure for a scrap yard worker is a 3,500 ton (3,175 t) pile of steel scrap, at an average distance of 2 meters from the worker. A 3,500 ton pile at a density of 20 lb/cu ft equates to a volume of 350,000 cu feet. This is equivalent to a hemisphere with a diameter at the base of approximately 100 feet. This model is reasonable and appropriate.

The exposed individual in the slag handling scenario is a worker at a steel mill who transfers slag using a front-end loader. The slag is spread over a large, flat area - the vehicle is either on top or at the edge of the pile. A vehicle shielding factor is applied which accounts for the shielding afforded by the loader. Based on interviews with equipment manufacturers and landfill operators, a geometric model was developed to characterize the effective shielding for operators of this type of large equipment. Based on calculated transmission factors, this parameter is assigned a triangular distribution with a range of 0.3 to 0.7 and a mode of 0.5. Informal calculations were performed using MicroShield and found the vehicle shielding factor range and mode to be appropriate.

The uncertainty factor is applied and is used to account for different locations of the worker with respect to the slag. If the worker is on top of the slag, they are exposed to an effectively infinite slab of slag, which conforms to the exposure conditions modeled in FGR 12 [Reference 6-13], so uncertainty factor equals 1. When the worker is loading slag from the edge of the large, flat pile, they are exposed to one-half of an infinite slab, so the uncertainty factor equals 0.5. The uncertainty factor is assigned a uniform distribution of 0.5 to 1.

Airborne Concentration

NUREG-1640 states: the airborne concentration of dust that is the source of inhalation exposure for the scrap pile worker was modeled based on of data collected by the U.S. Occupational Safety & Health Administration (OSHA) at the site of the World Trade Center in New York. After eliminating non-detect results and results greater than the OSHA 10 mg/m^3 limit, a random sample of 250 concentrations was used to calculate the average concentration. This parameter was assigned a custom distribution with a minimum of 0.962 mg/m^3, maximum of 1.378 mg/m^3 and a mode of 1.170 to fit the data.

The parameters used for the airborne concentration for handling and processing the slag and EAF dust shown in Appendix B was a lognormal distribution with a range of 1.27 to 5.0 mg/m^3 with the mode at 2.433 mg/m^3.

A 2016 study of dust exposure in a steel plant in Malaysia [Reference 6-14] was reviewed. Respirable particulate matter ($PM_{2.5}$, PM_{10}) and total particulate matter concentrations were evaluated. The mean concentration of the $PM_{2.5}$ dust ranged from 0.02 to 0.5 mg/m^3. The mean concentration of the PM_{10} dust ranged from 0.08 to 1.58 mg/m^3 which is bounded by the assumed distribution applied in this analysis.

Effective Dose Equivalent Conversion Factors

There are 7 exposure scenarios evaluated for the scrap yard and steel mill/foundry in NUREG-1640. The Effective Dose Equivalent (EDE) rate from all pathways for the 7 scenarios was evaluated to determine the most conservative dose rate coefficients. The 95th percentile dose rate coefficient was taken from Appendix F, Tables F1.1, F1.5, F1.9, F1.13, F1.14, F1.18 and F1.22. Table 6-14 presents the results of the evaluation.

There are 5 exposure scenarios evaluated for the leachate into groundwater in NUREG-1640. The EDE rate from all pathways for the 5 scenarios was evaluated to determine the most conservative dose rate coefficients. The 95th percentile dose rate coefficient was taken from Appendix F, Tables F1.62, F1.63, F1.64, F1.65 and F1.66. Table 6-15 presents the results of the evaluation.

6.10 Analysis and Results

Simulations were run for the Remediation and Component Removal worker scenarios. Each simulation set was run with 10,000 iterations. The total dose rate per unit surface activity concentration on the material for each radionuclide in (μSv/y)/(Bq/cm^2) was calculated. The value for the 95th percentile of the simulation results was chosen to calculate the surface contamination limits. The dose rate coefficients in (μSv/y)/(Bq/cm^2) were converted to (mrem/y)/(dpm/cm^2) by dividing by 600 (10 μSv/mrem x 60 dpm/Bq).

The RESRAD-BUILD model for the Tour Guide on Ship scenario generates results in (mrem/y)/(dpm/m^2). The results include all three exposure pathways. The dose rate coefficients were converted to (mrem/y)/(dpm/cm^2) by dividing by 10,000 cm^2/m^2.

The evaluation of the NUREG-1640 scrap yard/foundry worker scenarios is the most conservative scenario for 3 radionuclides. The NUREG-1640 evaluation of the leachate scenarios is the most conservative for 3 radionuclides. The maximum 95th percentile EDE for each radionuclide was converted to (mrem/y)/(dpm/cm^2).

The maximum value of the effective dose equivalent rate coefficients for all the scenarios for each radionuclide was used for calculating the surface contamination DCGL. Table 6-16 presents the results of the scenario calculations, showing the dose conversion factors (DCF), the maximum DCF and the scenario with the maximum DCF for each radionuclide.

Table 6-14 95th Percentile EDE from All Pathways - Scrap Yard and Foundry Worker

(μSv/y)/(Bq/cm²)

Nuclide	Scrap Yard	Handling Slag	Transferring EAF dust	Baghouse Maintenance	Handling Metal Product	Processing EAF Dust	Processing Steel Slag	Maximum	Worker
C-14	9.0E-04	0.0E+00	0.0E+00	0.0E+00	4.9E-04	0.0E+00	0.0E+00	9.0E-04	Scrap Yard
Co-60	3.8E+01	0.0E+00	1.8E-01	5.6E-03	7.9E-01	5.4E-02	0.0E+00	3.8E+01	Scrap Yard
Ni-63	3.5E-04	0.0E+00	5.8E-08	6.2E-10	2.9E-04	9.8E-07	0.0E+00	3.5E-04	Scrap Yard
Sr-90	2.0E-01	6.9E-02	5.1E-03	4.3E-05	0.0E+00	1.3E-03	8.7E-02	2.0E-01	Scrap Yard
Tc-99	1.2E-03	0.0E+00	1.9E-05	9.5E-08	7.8E-04	3.3E-08	0.0E+00	1.2E-03	Scrap Yard
Ag-108m	2.1E+01	0.0E+00	1.4E+00	4.5E-02	5.3E-01	3.9E-01	0.0E+00	2.1E+01	Scrap Yard
Cs-137	7.5E+00	7.9E-02	2.3E+00	9.2E-02	0.0E+00	9.6E-01	8.7E-02	7.5E+00	Scrap Yard
H-3	2.7E-05	0.0E+00	0.0E+00	0.0E+00	1.1E-06	0.0E+00	0.0E+00	2.7E-05	Scrap Yard
Fe-55	3.4E-04	5.9E-06	4.7E-06	2.1E-13	1.9E-04	7.9E-07	5.5E-06	3.4E-04	Scrap Yard
Am-241	2.2E+01	2.8E+01	2.2E+00	1.1E-04	0.0E+00	4.7E-01	2.9E+01	2.9E+01	Processing Steel slag
Pu-239/240	2.1E+01	2.0E+01	1.6E+00	1.1E-06	0.0E+00	3.3E-01	2.0E+01	2.1E+01	Scrap Yard

Table 6-15 95th Percentile EDE from All Pathways - Leachate (μSv/y)/(Bq/cm²)

Nuclide	Leachate - Ind dust	Leachate - Mun dust	Leachate - Ind scrap	Leachate - Mun scrap	Leachate - steel slag	Maximum	Scenario
C-14	4.3E-07	7.3E-09	9.5E-03	3.8E-03	0.0E+00	9.5E-03	Leachate - Ind scrap
Co-60	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
Ni-63	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
Sr-90	8.2E-09	1.1E-09	4.5E-08	7.1E-08	2.9E-02	2.9E-02	Leachate - steel slag
Tc-99	5.8E-03	2.6E-03	1.4E+00	5.3E-01	0.0E+00	1.4E+00	Leachate - Ind scrap
Ag-108m	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
Cs-137	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
H-3	0.0E+00	0.0E+00	1.4E-02	5.4E-03	0.0E+00	1.4E-02	Leachate - Ind scrap
Fe-55	0.0E+00	0.0E+00	0.0E+00	0.0E+00	1.5E-25	1.5E-25	Leachate - steel slag
Am-241	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
Pu-239/240	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	

Table 6-16 Scenario Results (mrem/y)/(dpm/cm^2)

Radionuclide	Tour Guide	Comp Removal Worker	Remediation Worker	NUREG-1640 Scrap Yard/Foundry Worker	NUREG-1640 Leachate	Maximum	Scenario
Ag-108m	9.56E-09	2.23E-02	2.21E-02	3.50E-02	0.00E+00	3.50E-02	NUREG-1640 Scrap Yard/Foundry Worker
C-14	1.96E-09	7.08E-06	6.29E-06	1.50E-06	1.58E-05	1.58E-05	NUREG-1640 Leachate
Co-60	1.88E-09	3.07E-02	3.06E-02	6.33E-02	0.00E+00	6.33E-02	NUREG-1640 Scrap Yard/Foundry Worker
Cs-137	4.04E-09	7.81E-03	7.80E-03	1.25E-02	0.00E+00	1.25E-02	NUREG-1640 Scrap Yard/Foundry Worker
Ni-63	9.99E-11	5.93E-06	1.79E-06	5.83E-07	0.00E+00	5.93E-06	Comp Removal Worker
Sr-90	7.10E-09	1.26E-03	4.39E-04	3.33E-04	4.83E-05	1.26E-03	Comp Removal Worker
Tc-99	2.37E-09	9.13E-06	4.46E-06	2.00E-06	2.33E-03	2.33E-03	NUREG-1640 Leachate
H-3	4.53E-12	2.17E-07	1.93E-07	4.50E-08	2.33E-05	2.33E-05	NUREG-1640 Leachate
Fe-55	8.21E-12	3.26E-06	1.85E-06	5.67E-07	2.50E-28	3.26E-06	Comp Removal Worker
Am-241	4.80E-07	3.58E-01	1.52E-02	4.83E-02	0.00E+00	3.58E-01	Comp Removal Worker
Pu-239/240	9.31E-07	3.46E-01	1.47E-02	3.50E-02	0.00E+00	3.46E-01	Comp Removal Worker

The maximum effective dose equivalent rate coefficient was then converted to a 15 mrem/year DCGL limit by dividing 15 by the maximum dose rate coefficient and then multiplying by 100. The resulting values are presented in Table 6-17.

Table 6-17 Surface Contamination Limits (DCGLs)

Radionuclide	Maximum (mrem/y)/(dpm/cm^2)	15 mrem/year limit (dpm/100cm^2)
Ag-108m	3.50E-02	4.29E+04
C-14	1.58E-05	9.47E+07
Co-60	6.33E-02	2.37E+04
Cs-137	1.25E-02	1.20E+05
Ni-63	5.93E-06	2.53E+08
Sr-90	1.26E-03	1.19E+06
Tc-99	2.33E-03	6.43E+05
H-3	2.33E-05	6.43E+07
Fe-55	3.26E-06	4.60E+08
Am-241	3.58E-01	4.19E+03
Pu-239/240	3.46E-01	4.34E+03

Levels of removable (non-fixed) contamination shall be reduced to the lowest levels that are reasonably possible. The recommended maximum removable contamination levels should be set at 10% of the total levels.

When multiple radionuclides are present in the waste stream, the Sum of the Fractions (SOF) must be solved. Equation 6-4 presents the formula used to calculate the Sum of Fractions.

$$SOF = \sum \frac{SAC_i}{DCGL_i} \qquad \text{Equation 6-4}$$

Where:

SOF = *Sum of the fractions for all radionuclides of concern*

SAC_i = *Surface Activity Concentration of radionuclide i, dpm/100cm^2*

$DCGL_i$ = *Derived Concentration Guideline Level of radionuclide i, dpm/100cm^2*

The SOF must be less than or equal to 1 for the material to meet the release criterion.

6.11 Reefing

Sections 6.1, 6.3 and 6.3.1 of this chapter describe artificial reefing in sufficient detail to explain why this disposition scenario is unlikely to be exercised, especially given current MARAD policy to restrict vessels built before 1985 from this method of disposal. As noted in section 6.3.1, fewer than two (2) percent of MARAD vessels have been disposed via artificial reefing and none since FY 2010. As part of the NHPA consultation, MARAD has requested public comment regarding disposition options for NSS. Reefing the ship represented a very small fraction of the responses, fewer than five percent, and no state has expressed interest to MARAD in obtaining NSS for reefing. Nevertheless, because reefing of any vessel based on its age is not restricted by statute, for the purposes of the LTP MARAD considers artificial reefing to be a Less Likely but Plausible (LLBP) scenario[27] and has evaluated potential exposures to determine if the DCGLs presented in Table 6-17 are limiting.

All else being equal with respect to the normal preparations of a ship for reefing, there are two potential end-state conditions for NSS if it is reefed. The structures and components that contain residual radioactivity are either retained in-situ and result in potential exposures in the marine environment and aquatic habitat, or they are removed prior to sinking the ship. In the case where the materials containing residual radioactivity are removed, the immediate shipbreaking scenario governs that activity and is bounding. Therefore, this section evaluates exposures in the case where the materials remain in-situ.

6.11.1 Diver on a Reefed Ship

For this evaluation, the most likely exposed individual is a diver. The only exposure mode is through external exposure to structures and components. The same model used for external exposure to the Remediation and Component workers was used for the divers with the 1-foot air gap changed to water. The resulting dose rates were again multiplied by 2 for conservatism. The exposure time was conservatively assumed to be 1000 hours in a year. The calculated dose rate coefficients and 15 mrem/year activity concentrations are presented in Table 6-18.

Table 6-18 Reefing Diver Dose Rate Coefficients and DCGLs

Radionuclide	(mrem/year)/ (dpm/cm^2)	15 mrem/year limit (dpm/100cm^2)
Ag-108m	2.19E-03	6.85E+05
Co-60	4.40E-03	3.41E+05
Cs-137	7.95E-04	1.89E+06

These DCGLs are approximately 14 to 16 times greater than those for the shipbreaking scenarios. Therefore, the shipbreaking DCGLs are bounding.

6.11.2 Consuming Fish from a Reefed Ship

A potential exposure pathway in the reefing scenario is through the consumption of contaminated seafood. Current nuclide bio-accumulation models require site specific measurements of the biomasses of various functional groups and key species. The undetermined location and biota composition of the reefing scenario make such modeling impossible. However, using conservative assumptions of the composition of residual nuclides at the time of reefing, it can be shown that concentrations of the residual nuclides in the environment would be less than the NRC unrestricted effluent release limits. By demonstrating concentrations lower than the unrestricted effluent release limits, it is also demonstrated that concentrations of the nuclides would not create deleterious environmental effects including bioaccumulation of nuclides in species eaten by humans. Ultimately this demonstrates that an exposure pathway via contaminated seafood is not of concern.

The following conservative assumptions are required to make this demonstration:

[27] NUREG-1757, Volume 2 Section I.3.3.3.6 states "Even if offsite use is not considered reasonably foreseeable, offsite scenarios may be less likely but plausible scenarios and should be analyzed as scenarios, to understand the robustness of the analysis."

1) The total residual man-made radioactive material is equal to 1 curie.

2) All contamination is located within the containment vessel (CV).

3) All contamination is located within 1mm or less of the exposed surface of the containment vessel's structural steel.

4) Structural steel deterioration is dependent on multiple factors. Two primary factors are water temperature and depth of the steel below the water surface.

   a) Higher water temperatures are correlated with faster rates of corrosion. This scenario will assume that the vessel in reefed in US controlled tropical waters and thus subject to the higher rates of corrosion. Corrosion rates derived from measurements collected from the USS Arizona in Pearl Harbor, Hawaii will be used for this scenario [Reference 6-15].

   b) Lower depth is associated with slower rates of corrosion. The reefing scenario will assume the highest point of the reefed vessel is 30 feet below sea level. Correspondingly, the reactor hatch will be at a depth of 56 feet and lowest portion of the reactor compartment at a depth of 110 feet. Corrosion rates for a depth of 56 feet, the highest portion of the containment vessel, will be used for these calculations.

   c) The corrosion rate (C) of 0.0026 mm/yr will be used as calculated from the following formula,

$$C = i_{corr} * 0.0254 \text{ mm/yr}$$

$$i_{corr} = -0.051\ (WD) + 2.96$$

Where: i_{corr} is the corrosion rate in mpy (mils per yr)

1 mil/yr = 0.0254 mm/yr

5) The volume of the containment vessel is estimated at 1.13E9 mL

6) A 100% exchange of water within the CV is achieved AT LEAST once every five years. This is perhaps the most conservative of assumptions. All openings on the Savannah would be opened or removed prior to reefing. In such a scenario water exchange in the CV would likely be multiple times daily.

Using the assumption of 1 Curie of residual material and the fractions of the total activity of the nuclides (From LTP Table 6-5), the total remaining microcuries of each nuclide can be calculated and are presented in Table 6-19.

Based on the assumption of 0.0026 mm per year deterioration of steel in seawater and the assumption of 1mm depth of contamination, the release rate each nuclide every five years can be calculated as follows:

$$0.0026\text{mm/year} * 5 \text{ years} = 0.013\text{mm}$$

$$0.013 \text{ mm of contaminated steel/ 1mm of total contamination} = 1.3\ \%\text{ of contaminated steel}$$

Based on this rate of deterioration, we can show the concentration of the nuclides after five years of sustained release with no water exchange. The method to calculate the concentration of the nuclides in the containment vessel water is shown below.

$$Conc_i = A_i/V*0.013$$

Where: A_i *=Nuclide activity in* μCi

$V = CV\ volume = 1.13E9\ ml$

Those activity concentrations were then compared to the 10CFR20 Appendix B, Table 2, Column 2 Effluent Concentrations. The results of the calculations are shown in the last two columns of Table 6-19. Note that all water concentrations are less than the effluent release concentrations in Appendix B of 10CFR20.

The low levels of nuclide concentrations show that the potential is below a level considered deleterious to humans and the environment. Given these concentrations are based on an extremely low rate of water exchange, consuming seafood from the reefed vessel is an unlikely potential pathway of exposure to individuals.

Table 6-19 Concentrations of the Residual Nuclides in the Containment Volume

Nuclide	Fraction of Total Activity	Activity (Ci)	Activity (µCi)	Concentration in CV water after 5 years (µCi/ml) Note these were calculated for $Conc_i = A_i/V*0.013$	NRC unrestricted release limit (µCi/ml)
C-14	5.61E-03	5.61E-03	5.61E+03	6.44E-08	3E-05
Co-60	1.63E-02	1.63E-02	1.63E+04	1.87E-07	3E-06
Ni-63	9.00E-01	9.00E-01	9.00E+05	1.03E-05	1E-04
Sr-90	1.54E-06	1.54E-06	1.54E+00	1.77E-11	5E-07
Tc-99	2.11E-05	2.11E-05	2.11E+01	2.42E-10	6E-05
Ag-108m	1.38E-04	1.38E-04	1.38E+02	1.58E-09	9E-06
Cs-137	7.61E-02	7.61E-02	7.61E+04	8.73E-07	1E-06
H-3	3.21E-04	3.21E-04	3.21E+02	3.68E-09	1E-03
Fe-55	1.35E-03	1.35E-03	1.35E+03	1.55E-08	1E-04
Am-241	9.66E-08	9.66E-08	9.66E-02	1.11E-12	2E-08
Pu-239/240	8.01E-08	8.01E-08	8.01E-02	9.19E-13	2E-08

6.12 Area Factors

An area factor is the magnitude by which the concentration within a small area of elevated activity can exceed DCGL while maintaining compliance with the release criterion. Area factors are only applicable to Class 1 survey units where the residual activity may be non-uniform. Also, area factors are only applicable to land areas and building surfaces. Based upon the characterization of the surfaces within the CV and RC, we do not expect the need to calculate or use area factors during the FSS.

6.13 Radionuclides of Concern and Insignificant Dose Contributors

NUREG-1757, Vol 2, Section 3.3 [Reference 6-8] states:

> *NRC staff considers radionuclides and exposure pathways that contribute no greater than 10 percent of the dose criteria to be insignificant contributors. Because the dose criteria are performance criteria, this 10 percent limit for insignificant contributors is an aggregate limitation only. That is, the sum of the dose contributions from all radionuclides and pathways considered insignificant should be no greater than 10 percent of the dose criteria.*

After removal of the radionuclides considered as Insignificant Dose Contributors (IDC), the remaining radionuclides are designated as the Radionuclides of Concern (ROC). The IDCs are determined by calculation of the Relative Dose Fraction. The Relative Dose Fraction, $RDF_{i,k}$, for radionuclide i in each sample is calculated using the radionuclide fractions from Table 6-4, the applicable DCGLs from Table 6-17 and Equation 6-5.

$$RDF_{i,k} = \frac{fA_{i,k}}{DCGL_i}\left[\frac{1}{\sum i\frac{fA_{i,k}}{DCGL_i}}\right] \qquad \text{Equation 6-5}$$

Where:

$RDF_{i,k}$ = *Relative Dose Fraction for radionuclide i in the sample*

$fA_{i,k}$ = *Activity fraction of radionuclide i in the sample*

$DCGL_i$ = *DCGL for radionuclide i*

The results of the activity fraction/DCGL calculations are presented in Table 6-19.

Table 6-20 Activity Fractions/DCGLs

Nuclide	SG composite frac/DCGL	PZR composite frac/DCGL	PD-T4 composite frac/DCGL	RC Exhaust Vent composite frac/DCGL	RC IX Piping composite frac/DCGL	Makeup Storage Tank frac/DCGL
Ag-108m	0.00E+00	0.00E+00	4.55E-09	1.17E-08	1.10E-09	1.99E-09
C-14	1.09E-13	1.28E-11	2.66E-12	2.99E-10	2.93E-11	1.12E-11
Co-60	4.47E-08	9.80E-07	8.27E-07	4.00E-07	6.49E-07	1.23E-06
Cs-137	3.42E-11	1.54E-09	1.26E-07	3.60E-06	1.42E-09	7.17E-08
Ni-63	3.95E-09	3.86E-09	3.82E-09	2.09E-09	3.88E-09	3.76E-09
Sr-90	0.00E+00	6.38E-13	0.00E+00	0.00E+00	2.51E-13	6.88E-12
Tc-99	3.63E-12	2.32E-11	1.09E-10	0.00E+00	4.87E-11	1.20E-11
H-3	-	-	-	-	-	3.00E-11
Fe-55	-	-	-	-	-	1.76E-11
Am-241	-	-	-	-	-	1.38E-10
Pu-239/240	-	-	-	-	-	1.11E-10
Totals	**4.87E-08**	**9.86E-07**	**9.61E-07**	**4.02E-06**	**6.56E-07**	**1.30E-06**

The Relative Dose Fractions (RDFs) were calculated by dividing the fractions presented in Table 6-19 for each radionuclide by the totals for each sample presented in Table 6-19. The RDFs for each radionuclide in each sample are presented in Table 6-20.

Table 6-21 Relative Dose Fractions

Nuclide	SG composite RDF	PZR composite RDF	PD-T4 composite RDF	RC Exhaust Vent composite RDF	RC IX Piping composite RDF	Makeup Storage Tank RDF
Ag-108m	0.00E+00	0.00E+00	4.73E-03	2.92E-03	1.68E-03	1.53E-03
C-14	2.24E-06	1.30E-05	2.77E-06	7.45E-05	4.46E-05	8.60E-06
Co-60	9.18E-01	9.94E-01	8.60E-01	9.96E-02	9.90E-01	9.40E-01
Cs-137	7.03E-04	1.56E-03	1.31E-01	8.97E-01	2.16E-03	5.50E-02
Ni-63	8.12E-02	3.92E-03	3.97E-03	5.21E-04	5.92E-03	2.89E-03
Sr-90	0.00E+00	6.47E-07	0.00E+00	0.00E+00	3.83E-07	5.28E-06
Tc-99	7.46E-05	2.36E-05	1.14E-04	0.00E+00	7.42E-05	9.18E-06
H-3	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.30E-05
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.35E-05
Am-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.06E-04
Pu-239/240	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.50E-05

The RDFs for Co-60 and Cs-137 were summed for each sample. The RDFs for the remaining radionuclides were summed for each sample. The results of the calculations are presented in Table 6-21.

Table 6-22 Radionuclides of Concern (ROCs) and Insignificant Dose Contributors (IDCs)

Nuclide	SG composite	PZR composite	PD-T4 composite	RC Exhaust Vent composite	RC IX Piping composite	Makeup Storage Tank sample
ROCs (Co-60 and Cs-137)	9.19E-01	9.96E-01	9.91E-01	9.96E-01	9.92E-01	9.95E-01
Remainder (IDCs)	8.13E-02	3.95E-03	8.82E-03	3.51E-03	7.72E-03	4.67E-03

Based upon this evaluation, the only two ROCs are Cs-137 and Co-60. The IDC dose in five out of the six samples is less than 1 percent. The sixth sample is 8 percent of the dose. Because this evaluation was based on a dose limit of 15 mrem/year, no adjustment (reduction) in the DCGLs is necessary or will be performed.

6.14 Additional Considerations Concerning Preservation

As described in Section 6.1, the analyses presented in this chapter consider the scenario in which NSS is scrapped immediately after license termination to be the worst-case scenario from a dose perspective. In adopting a building reuse analogy for the preservation scenario, MARAD considers the 70-year exposure timeframe postulated in NUREG-1496 to be applicable to NSS. Given that NSS will be nearly 70-years old at the anticipated license termination date, and that the typical lifespan of oceangoing ships is less than half of NSS' current age, it is prudent to consider whether NSS can be preserved for such a lengthy period. This discussion focuses on experiences with preserved vessels and does not consider examples of long-lived operating commercial ships, such as on the Great Lakes, where at least one commercially active hull is over 115 years old.[28]

The first oceangoing ship to be constructed with an all-iron hull is reputedly the SS *Great Britain* of 1845. *Great Britain* was constructed by the renowned British engineer Isambard Kingdom Brunel. In addition to its wrought iron hull construction (wood was used for decks and masts), *Great Britain* was principally propelled by steam. Here again the ship was innovative, featuring a rudimentary screw propeller vice paddle wheels mounted to the ship's sides. Important to our consideration of the potential longevity of NSS, *Great Britain* survives to this day and is displayed as a static museum ship in the drydock in Bristol, where it was built. Although restored and with many features recreated, its original hull is substantially intact. Thus, the world's first iron-hulled oceangoing ship is also its oldest surviving iron-hulled ship.

The more common use of iron in the early 19th century was the composite hull, in which iron frames support a wood-sheathed hull. This form of construction gave way to a widespread use of all-iron hulls beginning in the 1850s. The use of steel as a primary hull material began in the mid-1870s but took a period of some 30 years before becoming the dominant form of large merchant and naval ship construction. There are numerous preserved vessels from this period, spanning the full range of vessel types. These ships are preserved both afloat and ashore. The oldest such ship that is in seagoing condition today is *Star of India*, maintained by the San Diego Maritime Museum, and originally built in 1863. This ship has always been waterborne.

There are much older preserved ships in existence. Among the oldest of these are Egyptian funeral barges recovered from archeological sites – these are some 4,000 years old. There are many Viking longboats, also recovered from archeological sites, that are some 1,000 years old. Shipwrecks recovered from the sea and restored as museums include *Mary Rose* (1511) and *Vasa* (1628). These examples are not fully relevant to NSS because they involved some aspect of loss of the vessel and exposure to the environment (note – this also applies to *Great Britain*). They do, however, serve to indicate an interest by the public in preserving old ships. The key factor in any individual ship's longevity, particularly for large vessels preserved afloat, is proper and continuing maintenance. This especially includes periodic drydocking to allow for inspection and repair of the underwater hull.

Concentrating on more modern examples of preserved steel-hulled ships, there are numerous US warships preserved around the country, most of which are afloat. The oldest of these is the 1895-built USS *Olympia*, flagship of Admiral Dewey at the 1898 Battle of Manilla Bay. This ship is preserved in Philadelphia. The next oldest preserved major warship afloat is battleship *Texas*, of 1914. From 1948 to present, *Texas* was preserved in a basin adjacent to the upper reaches of the Houston Ship Channel. The ship will be relocated, most likely to a seawater berth, after its current drydocking period is completed.

[28] The *St. Marys Challenger* was originally constructed in 1906 as *William P. Snyder*. The vessel operated until 2013 as a self-propelled steam-powered freighter. Its machinery was removed over the winter of 2013-2014, with the hull converted into a barge. The barge, keeping the name *St Marys Challenger*, is paired with a tug to form an articulated tug-barge unit, and is still operating in 2023.

These vessels offer a realistic present-day scenario for the potential longevity of NSS, which was completed in 1962. *Olympia* is 67 years older than NSS, while *Texas* is 48 years older. These vessels were preserved soon after the end of their active service lives, which was 1948 for *Texas*, and 1922 for *Olympia*. *Olympia* has not been drydocked since 1945, whereas *Texas* was drydocked in 1988, and is again on drydock in 2023. By comparison, NSS ended service in 1970, at the conclusion of a drydocking availability. The ship has been maintained in protective storage and active decommissioning from 1970 to present, with a 13-year period of public display; all in seawater locations. Since 1970 the ship has been drydocked four (4) times: 1975, 1994, 2008, and 2020.

On any static vessel, the area of the hull which is most subject to deterioration is the wind-water interface (waterline). This area is similar to the splash zone of fixed marine structures, and is subject to continuous wave action, leading to repeated wetting of the steel surfaces. Inevitably this action leads to corrosion, causing both erosion and pinholing of the steel surface if the ship is not maintained. Both *Olympia* and *Texas* have suffered significant problems in this area, and both have required substantial repairs to their hull at the waterline area. There are generally two approaches to maintaining a museum ship hull; the first involving periodic drydocking for inspection and repairs, and the second involving repairs with the ship afloat, typically using cofferdams to isolate those sections of the hull needing repair. *Olympia* has followed the second approach, whereas *Texas* has used a combination of both methods.

The service history of NSS' hull structure is mixed. Hull pitting was observed early in the ship's career. Sometime prior to 1970, the ship was fitted with a first-generation impressed current cathodic protection system, which was intended to minimize further deterioration of the underwater hull plating. The system was groomed and deemed to be functioning satisfactorily at the 1975 drydocking. That drydocking was performed to prepare the hull for a fifty-year retention period in a MARAD reserve fleet, in accordance with the mothballing criteria of RG 1.86. Among other things, all the ship's through-hull connections were fitted with welded steel blanks. From the onset of protective storage, *SAVANNAH'S* hull was better prepared for extended drydocking intervals than most museum ships in normal experience. Despite this good preparation, there was a significant cathodic protection system failure while the ship was in museum service, resulting in extensive pitting and leaks. The hull leaks were the proximate cause for the ship's removal and drydocking in 1994. Over the course of the three protective storage drydockings in 1994, 2008 and 2020, MARAD: a) prepared the ship's underwater hull surfaces to bare metal and repainted using highly-effective marine coatings; b) effected long-term weld repairs to deteriorated shell plating and rivets, c) removed the ship's propeller to reduce the potential for electrolytic corrosion of the hull, and d) replaced and upgraded the components and controllers for the ship's cathodic protection system. MARAD completed scantling assessments and evaluations of the ship's hull structure in 2008 and again in 2020 to ensure that the hull is in suitable condition for the static services of protective storage or future preservation. MARAD also conducts annual diver-based underwater surveys to monitor the condition of the hull.

Considering the extensive work performed on NSS' hull, and the future option for afloat hull maintenance and repair, combined with the experiences of existing museum ships with afloat preservation service lives of at least seventy-five (75) years, MARAD believes that a seventy-year preservation period for NSS after license termination is reasonable. Table 6-23 projects the effects of radiological decay over the seventy years, in ten-year increments for each radionuclide of interest.

Table 6-23 Activity Fractions Remaining Over Time

Radionuclide	LT	10 y	20 y	30 y	40 y	50 y	60 y	70 y
H-3	1	5.69E-01	3.23E-01	1.84E-01	1.05E-01	5.95E-02	3.38E-02	1.92E-02
Fe-55	1	7.67E-02	5.89E-03	4.52E-04	3.47E-05	2.66E-06	2.04E-07	1.57E-08
C-14	1	9.99E-01	9.98E-01	9.96E-01	9.95E-01	9.94E-01	9.93E-01	9.92E-01
Co-60	1	2.68E-01	7.21E-02	1.94E-02	5.19E-03	1.39E-03	3.74E-04	1.01E-04
Ni-63	1	9.33E-01	8.71E-01	8.12E-01	7.58E-01	7.07E-01	6.60E-01	6.16E-01
Sr-90	1	7.85E-01	6.16E-01	4.83E-01	3.79E-01	2.98E-01	2.34E-01	1.83E-01
Tc-99	1	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Ag-108m	1	9.83E-01	9.67E-01	9.51E-01	9.35E-01	9.19E-01	9.04E-01	8.88E-01
Cs-137	1	7.95E-01	6.32E-01	5.02E-01	3.99E-01	3.17E-01	2.52E-01	2.00E-01
Am-241	1	9.84E-01	9.68E-01	9.53E-01	9.38E-01	9.23E-01	9.08E-01	8.94E-01
Pu-239/240	1	1.00E+00	9.99E-01	9.99E-01	9.99E-01	9.99E-01	9.98E-01	9.98E-01

Another evaluation was performed to estimate the effects of decay over time. Cobalt-60 and Cs-137 are the dominant dose contributors at the ship. Table 6-23 shows that the fraction of Co-60 remaining after only ten years decay is equal to 26.8%. This evaluation assumes that the decay over time is dominated by Cs-137. Assuming that the DCGL from residual radioactivity at License Termination is equal to an annual dose of 15 mrem, the annual dose will decrease over time with the half-life of Cs-137. Table 6-24 presents the results of the calculation.

Table 6-24 Annual Dose Over Time (mrem)

LT	10 y	20 y	30 y	40 y	50 y	60 y	70 y
15	11.9	9.5	7.5	6.0	4.8	3.8	3.0

6.15 References

6-1 Regulatory Guide 1.179, *Standard Format and Contents for License Termination Plans for Nuclear Power Reactors*, Rev. 2, July 2019

6-2 NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plan*s, Rev. 2, April 2018

6-3 NUREG-1496, *Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities,* Volume 1, July 1997

6-4 NUREG-1640, *Radiological Assessments for Clearance of Materials from Nuclear Facilities*, Volume 1, 2003 and Volume 2, 2004

6-5 Joint MARAD-EPA *National Guidance: Best Management Practices for Preparing Vessels Intended to Create Artificial Reefs*, May 2006.

6-6 CR-109, RSCS TSD 19-031, *N.S. SAVANNAH RCCV Characterization*, Revision 1, February 2020

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6-8 NUREG-1757, *Consolidated NMSS Decommissioning Guidance*, Volume 2, 2003

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6-13 Federal Guidance Report No.12, *External Exposure to Radionuclides in Air, Water and Soil*, EPA-402-R-93-081, 1993

6-14 *Assessment of dust exposure in a steel plant in the eastern coast of peninsular Malaysia*, Work, Vol. 55, Issue No. 3, pp. 655-662, 2016

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7 UPDATE of the SITE-SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(F), the guidance of *Regulatory Guide 1.179, Standard Format and Content for License Termination Plans for Nuclear Power Reactors* [Reference 7-1] and the guidance in NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, [Reference 7-2], this chapter provides a summary of MARAD's site-specific decommissioning costs as of September 30, 2023.

In accordance with Reference 7-1, the LTP should address two principal factors: a) the estimated remaining costs to complete decommissioning at the time of LTP submittal,[29] and b) a comparison of those remaining costs to the funds available (set aside) for decommissioning. If there is a deficit, the licensee must indicate the means to provide adequate funding to complete decommissioning. The projected costs for decommissioning were determined in Calendar Year (CY) 2021 when the fixed price primary decommissioning services contract (aka TSIM)[30] was awarded. MARAD was then able to address the factors in its CY 2021 Decommissioning Funds Status Report [Reference 7-3]. In addressing the factors, MARAD described the scope of work covered by the TSIM and other minor contracts related to decommissioning and projected the costs of them all out to license termination. No deficit was identified. MARAD updated these projections in its CY 2022 Decommissioning Funds Status Report [Reference 7-4], and again, no deficit was identified. In each case, MARAD made allowance for potential requests for equitable adjustment (REA, aka change orders) when assessing whether the available balance of funds was adequate to complete decommissioning and license termination. This chapter further updates the projections, bringing them forward from December 31, 2022, to September 30, 2023. During this period, MARAD has settled several requests for equitable adjustment, and directed several change orders that noticeably reduced the available balance of funding. Nevertheless, MARAD still considers the available balance of funds described in the LTP to be adequate to complete decommissioning and license termination.[31] As a federal licensee, MARAD relies on federal appropriations for decommissioning funding, and the full faith and credit of the United States for financial assurance (see Reference [7-5]). This mechanism will remain in effect until the license is terminated.

As stated in Section 2.3 of Reference 7-3, the fixed price TSIM contract, with certain exceptions noted:

> *... is intended to cover all remaining MARAD requirements to complete DECON-LT.*

The fixed price nature of the TSIM contract is such that any estimate of remaining costs on any given date is simply the value of future obligations or future invoices as of that date. Consequently, MARAD believes that new estimates are not required to meet the underlying intent of the regulation and has therefore not prepared a new engineering cost estimate. The following sections of this chapter apply this logic to the content requirements of References 7-1 and 7-2.

Note that MARAD will continue to submit CY Decommissioning Funds Status Reports on an annual basis until the license is terminated, which will provide NRC with updated decommissioning estimates if

[29] For the purposes of this chapter, MARAD defines "time of LTP submittal" as September 30, 2023. This date is the end of federal FY 2023 and occurs within a reasonable span of the expected LTP submittal.
[30] This contract is in the same general form and scope as previous Technical Support for Integrated Management of Licensed Activities contracts awarded by MARAD and known by the acronym TSIM. Where it appears in the LTP, TSIM refers to the 2021 primary decommissioning services contract.
[31] By letter dated September 26, 2022, NRC provided its analysis of Reference 7-3, and concluded that MARAD had satisfied the 10 CFR 50.82 decommissioning funding assurance requirements as of December 31, 2021.

project costs change based on newly discovered requirements, or resolution of contract change orders / requests for equitable adjustment.

7.2 Estimated Remaining Decommissioning Costs

7.2.1 Previously Docketed Decommissioning Estimates

MARAD's Decommissioning Cost Estimate (DCE) was provided in Revision 1 to its Post Shutdown Decommissioning Activities Report (PSDAR) [Reference 7-6]. To-date, there have been no substantive changes to the previously docketed DCE, other than routine escalation over time, and reporting of actual costs.

7.2.1.1 PSDAR Cost Estimate

In Reference 7-6 MARAD described its 2006 and 2008 Rough Order of Magnitude (ROM) estimates and provided a summary of the 2008 ROM as its DCE. That DCE was prepared for budget planning purposes, and not as an engineering estimate. It did, however, employ engineering estimate methodologies, including unit cost factors, commercial nuclear industry standards, scaling factors, and benchmarking of other power reactor decommissioning projects (e.g., Yankee Rowe, Connecticut Yankee and Big Rock Point). The 2006 ROM addressed each of the seven cost elements[32] whose evaluation is required in Reference 7-1. It was subjected to three (3) separate and independent[33] Verification & Validation studies, and its final revision incorporated the comments and recommendations received. The overall contingency factor was thirty (30) percent, and these adjustments carried forward into subsequent estimates. After the withdrawal of PSDAR Rev. 0, the 2006 ROM became the basis for the 2008 ROM included as the DCE in Reference 7-6.

Reference 7-6 also described in summary form the inherent difficulties present in the federal budgeting and Congressional appropriations processes. Beginning with the 2006 ROM, the DCE applied extremely conservative cost assumptions, with a direct intent to overestimate the cost of the project, as a means of assuring that appropriations would be sufficient to complete the work.

7.2.1.2 Decommissioning Funds Status Reports

Beginning with CY 2009, MARAD has submitted Decommissioning Funds Status Reports on an annual basis. Each report updated the DCE using the NRC-specified escalation factors, and the DCE itself received a complete review and revision every five years. The most recent five-year DCE revision was deferred to the LTP (see Section 7-4 of this chapter).

In Section 4.1(B) of Reference 7-4, MARAD provided estimated costs to complete decommissioning as of December 31, 2022. This section is repeated in part in Section 7.2.2 of this chapter and is revised to make it current as of September 30, 2023. Those revisions in Section 7.2.2 are indicated by **bold text**. This revision accounts for the following additional obligations made during the nine month period of the update, and refined projections of future costs: a) settled TSIM requests for equitable adjustment; b) MARAD-directed TSIM change orders; c) layberthing services for the 2023 – 2024 period; d) additional costs in minor support contracts; e) additional obligations to support NRC fee recoveries; and f) refined salary and overhead projections.

[32] The elements include: a) cost assumptions used including contingency factors; b) major decommissioning activities and tasks; c) unit cost factors; d) estimated costs of decontamination and removal of equipment and structures; e) estimated costs of waste disposal including any applicable disposal site surcharges; f) estimated final survey costs, and g) estimated total costs.
[33] V&Vs of the 2006 ROM were performed by the U.S. Department of Energy (Argonne National Lab), BAE Systems, and consultants affiliated with the Massachusetts Institute of Technology.

7.2.2 Current Site-Specific Decommissioning Cost Estimate

As noted in Section 7-1, MARAD has not prepared a new engineering estimate to support the LTP. Instead, the LTP summarizes the estimated remaining, unobligated costs associated with the TSIM and other service contracts that support decommissioning. The italicized text below is repeated and revised from Reference 7-5. Revisions are shown in **bold text.** Table 7-1 summarizes the estimates.

*Subject to any negotiated change orders, the total costs to complete the decommissioning activities contained in the TSIM contract was **$64,527,170.97** as of **September 30, 2023**. The awarded amount as of December 31, 2022, was **$60,067,459.97**. The difference **of $4,459,711.00** is the value of the outyear baseline services CLINs for performance years **3** through 5 of the contract,*[34] *and represents the primary estimate of remaining decommissioning costs.* ***Table 7-1 adjusts this upward as per footnote 34.***

*The value of the T&M line item was **$5,261,410.42** as of **September 30, 2023**. MARAD estimates that an additional **$1.0** million will be obligated to this line item over the course of the TSIM contract.*

*The estimated future costs for engineering review, independent oversight, field inspection, and independent regulatory analysis and review services through the projected License Termination is **$0.5** million. The estimated future costs for MARAD payroll, {travel} and public affairs activities through the projected license termination is **$2.1** million* (note that travel was not included in Reference 7-3). *The estimated cost of layberthing and utility services through the projected License Termination timeframe is **$1.53** million.*

Table 7-1 Estimated Remaining Decommissioning Costs as of September 30, 2023

Cost Category by Appropriation	Amount
MARAD Annual Appropriations	
TSIM Baseline Services (CLIN 3-5)	\$6.164,964
Layberthing and Utility Services **(to 03/31/2026)**	\$1,500,000
Subtotal	\$7,664,964
MARAD DECON Appropriation	
TSIM T&M Projection	\$1,000,000
REAs	\$3,000,000
MARAD Support Contracts	\$750,000
MARAD Payroll, Travel & Public Affairs	\$2,200,000
Interagency Agreement (NRC)	\$500,000
Subtotal	\$7,450,000

[34] CLINs 003, 004, 005 are among the five line items funded by MARAD's annual protective storage appropriation. Future funding for these CLINs will be provided by the FY 2024, and 2025 appropriations. CLIN 003 was partially funded during FY 2023; the table reflects the outstanding balance to be funded in FY 2024. The line also includes the estimate to fund CLIN 005 to the March 31, 2026, end of the contract performance period; it is currently priced only to September 30, 2025.

Total	$15,114,964

7.2.2.2 Cost Elements of the Decommissioning Estimate in relation to the Fixed Price Services Contract

Both Reference 7-1 and Reference 7-2 require confirmation that the estimate of remaining decommissioning costs evaluate seven elements, which are not meant to be all-inclusive. The seven elements are:

(1) Cost assumptions used, including a contingency factor;

(2) Major decommissioning activities and tasks;

(3) Unit cost factors;

(4) Estimated costs of decontamination and removal of equipment and structures;

(5) Estimated costs of waste disposal, including applicable disposal site surcharges;

(6) Estimated final survey costs; and,

(7) Estimated total costs.

As noted in Section 7.2.1.1, MARAD's confirms that its DCE included the above elements. Although MARAD has not prepared a new engineering estimate for the LTP, the balance of this section discusses how the seven elements are considered within the context of MARAD's current activities. Note that element (7) is addressed in the immediately preceding subsection.

As described in Reference 7-3, the TSIM contract is made up of nine (9) contract line items (CLIN). Eight (8) of the CLINs are fixed price. CLIN 0009 is a Time & Material item that provides for supplemental labor, materials, and services to perform work within the scope of the contract, but not included in the fixed price award. To date, the item has provided for, among other things, COVID-19 facility protocols, including enhanced cleaning and disinfection, and maintenance and repairs beyond the fixed price baseline of preventive maintenance, inspections and surveys that support facility operations during decommissioning. Five (5) of the fixed price CLINs provide baseline services and labor; each has a one-year period of performance that corresponds to the contract anniversary date. The remaining three (3) fixed price CLINs provide the bulk of decommissioning services and cover the entire contract period of performance.

The Contractor's price proposal addresses elements (1) and (3). Note that element (3) is subsumed within the cost estimating process employed by the Contractor to develop its fixed price proposal. Except for element (7), the three fixed price decommissioning CLINs address the remainder of these elements and are described below. Note that cost elements may be supported by more than one line item.

7.2.2.2.1 CLIN 0006 TSIM Decommissioning Support

This line item provides labor, materials, and supplies to perform dismantlement activities in Phase II of the DECON-LT project. It also includes all costs to package Low Level Radioactive Waste (LLRW) and remove it from the ship to the transport conveyance. In the case of the Reactor Pressure Vessel (RPV), this includes the personnel, materials, and logistics services associated with landing the RPV package onto land and transferring it to rail as the final transportation mode. This line item supports elements (2) and (4).

7.2.2.2.2 CLIN 0007 Decommissioning – License Termination

This is an activity-based line item that specifically addresses elements (2), (4), and (6) through performance milestones. The seven major activity groups defined in this CLIN are:

- Mobilization and D&D Preparations
- Energy (to the extent not covered by MARAD in its layberth contract)
- Major Component Removal
- Major Component and Structure Decontamination
- Minor Component Removal
- Decommissioning Equipment and Tools
- License Termination and Demobilization

The seven major activity groups have been subdivided into 36 discrete payment milestones, which are paid upon milestone completion. The License Termination and Demobilization activity group includes the comprehensive costs for preparation, submittal, review and approval of the LTP, planning, performance and submittal of Final Status Surveys, and support for NRC confirmatory surveys.

7.2.2.2.3 CLIN 0008 Waste Disposal

This line item provides for the transportation and disposal of all waste generated by decommissioning activities. This line item supports element (5).

7.3 Decommissioning Funding

7.3.1 Funds Available to Complete Decommissioning

As of September 30, 2023, the MARAD DECON-LT appropriation balance of funds available is $18,516,532.71.

As of September 30, 2023, the MARAD annual baseline activities (protective storage) appropriation balance of funds available is $700,000. Projected appropriations in FY 2024, 2025 and 2026 are $9.0 million, for total projected baseline activities funding of approximately $9.7 million. MARAD currently anticipates continuing to request protective storage funding through FY 2027.

For both the DECON and protective storage accounts, funds will carryover and will be available until expended.

7.3.2 Comparison to Estimated Remaining Costs

With respect to the DECON-LT appropriation, the funds available exceed the estimated remaining costs. After deducting the estimated remaining DECON costs from Table 7-1, the approximate balance of funds available for decommissioning is $11.0 million.

With respect to the baseline activities appropriation, future appropriations are expected to be adequate for the estimated costs.

7.4 Regulatory Commitments

In Reference 7-3, MARAD committed to provide its revised DCE in the LTP. With submission of the LTP, this commitment is completed.

7.5 Future Escalation of Estimates

As required by 10 CFR 50.75(f)(1), MARAD submitted Reference 7-3 on March 29, 2022. Section 3.4 of that report addresses the seven required reporting items of the regulation, including the formula-based escalation of estimated decommissioning costs. For CY 2021, MARAD temporarily deferred performing the escalation calculation, as described in Sections 3.1 and 3.4 of Reference 7-3 and stated that it would address future formula-based escalations in the LTP.[35] Based on the discussions contained in this chapter, MARAD believes it has met the underlying intent of 10 CFR 50.75(a), which is to provide reasonable assurance that resources are available for decommissioning. Because MARAD is relying upon the contract fixed price to calculate remaining decommissioning costs when such costs are required to be reported, and because the TSIM contract does not include escalation or inflation-adjustment clauses, MARAD believes that there is no added-value to performing formula-based escalation calculations for the few years remaining before license termination.

7.6 References

7-1 Regulatory Guide 1.179, *Standard Format and Content for License Termination Plans for Nuclear Power Reactors*, Revision 2 – July 2019

7-2 NUREG-1700, *Standard Review Plan for License Termination Plan*, Revision 2, April 2018

7-3 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated March 29, 2022 - *Submittal of Decommissioning Funds Status Report for Calendar Year (CY) 2021*

7-4 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated March 29, 2023 - *Submittal of Decommissioning Funds Status Report for Calendar Year (CY) 2022*

7-5 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated March 31, 2011 – *Submittal of Decommissioning Funds Status Report for CY 2010 and updated Governmental Statement of Intent for Decommissioning Financial Assurance* (ML1109400076)

7-6 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated December 11, 2008, *Submittal of Post Shutdown Decommissioning Activities Report*, Revision 1

[35] When the CY 2021 decommissioning funds status report was submitted, MARAD anticipated that it would submit the LTP before the end of CY 2022. This was based on the further expectation that the NHPA PA would be executed in CY 2021, and that the PA was a prerequisite to the LTP submittal. The PA was not executed in CY 2022, and therefore the CY 2022 decommissioning funds status report was submitted before the LTP. Consequently the CY 2022 decommissioning funds status report continued the deferral of the formula-based escalation of the decommissioning cost estimate.

8 SUPPLEMENT to the ENVIRONMENTAL REPORT

8.1 Introduction

In accordance with the underlying intent of 10 CFR 50.82(a)(9)(ii)(G), the guidance of *Regulatory Guide 1.179, Standard Format and Contents for License Termination Plans for Nuclear Power Reactors* [Reference 8-1] and the guidance in NUREG-1700, *Standard Review Plan for License Termination Plan*, [Reference 8-2], this chapter provides a summary of MARAD's environmental analyses in support of decommissioning and license termination activities, with emphasis on MARAD's 2019 Supplemental Environmental Assessment (SEA).

References 8-1 and 8-2 describe the content required for this chapter. Summarizing from these references, the LTP must submit a supplement to the environmental report (ER) describing any new information or significant environmental changes associated with the site-specific termination activities. As described in 10 CFR 51.53, the term ER refers to the licensee's "Environmental Report—Operating License Stage" (see 10 CFR 51.53(b)), and the supplement refers to the separate 10 CFR 51.53(d) document, entitled "Supplement to Applicant's Environmental Report—Post Operating License Stage". The supplement must:

a. Describe in detail the environmental impact of the site-specific termination activity.

b. Compare the impact with previously analyzed termination activities (see NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," Supplement 1, "Regarding the Decommissioning of Nuclear Power Reactors," issued November 2002) (Ref. 17).

c. Analyze the environmental impact of the site-specific activity. Include alternative actions and any mitigating actions.

The NSS was constructed and operated when there were no AEC regulations for environmental protection. On July 18, 1974, the AEC promulgated 10 CFR Part 51-Licensing and Regulatory Policy and Procedures for Environmental Protection. The regulations were intended to implement the revised Guidelines of the Council on Environmental Quality published in the FEDERAL REGISTER on August 1, 1973, pertaining to preparation of environmental impact statements pursuant to the National Environmental Policy Act of 1969 (NEPA). Promulgation of these regulations was over three years after the NSS's final shutdown in November1970. The impact of the new Part 51 was recognized when MARAD proposed the first significant licensing action for the NSS which was a license amendment request for a Possession-only License in 1976. This request was developed after both permanent cessation of NSS operations on December 3, 1971, and promulgation of 10 CFR Part 51 in 1974.

It is instructive to provide the following timeline that shows the timing relationship of AEC's development of 10 CFR Part 51 and MARAD's development of NEPA compliance documents as shown below:

- August 1, 1973 – The Council on Environmental Quality (CEQ) publishes the first guidelines for preparation of Environmental Impact Statements (EIS) pursuant to NEPA.
- June 1974 – The AEC publishes Regulatory Guide 1.86 Termination of Operating Licenses for Nuclear Reactors. This document contains no discussion of NEPA compliance.
- August 19, 1974 – The effective date of the original 10 CFR Part 51 Licensing and Regulatory Policy and Procedures for Environmental Protection.

- March 17, 1976 – MARAD submits its Possession-only License amendment request, the first major licensing action by MARAD after promulgation of 10 CFR Part 51.
- April 13, 1976 – MARAD submits an Environmental Assessment (EA) to NRC for consideration as part of the Possession-only License.[36] The Possession-only License is issued on May 19, 1976. This EA is repurposed and updated for subsequent license amendments and renewals until MARAD submits its revised PSDAR in 2008. The exception was License Amendment 9 in 1981. That amendment request included a site-specific EIS related to operation of NSS as a museum, hotel, and restaurant.
- March 12, 1984 – NRC publishes the revised 10 CFR 51 regulations in the Federal Register with a proposed effective date not later than June 7, 1984.
- October 16, 2003 – MARAD contractor WPI produces a screening-level bounding analysis of NSS decommissioning against the NRC Final Generic Environmental Impact Statement on Decommissioning Nuclear Facilities (1988) and its Supplement (2002).
- March 2008 – MARAD publishes an EA and Finding of No Significant Impact (FONSI – dated May 6, 2008) regarding NSS Decommissioning. MARAD submits these documents to NRC on October 3, 2008 [Reference 8-3].
- December 11, 2008 – MARAD submits its revised PSDAR.
- April 2019 – MARAD publishes SEA and FONSI regarding NSS Decommissioning.

Both MARAD and NRC are federal agencies, each with its own NEPA responsibilities. The 1974 AEC regulations acknowledged this in a footnote, which stated "*Where the "applicant", as used in this part, is a Federal agency, different arrangements for implementing NEPA may be made, pursuant to the Guidelines established by the Council on Environmental Quality.*" The 1974 AEC regulations established the requirements for Environmental Reports at the Construction Permit Stage (10 CFR 51.20) and Operating License Stage (10 CFR 51.21). Retroactive ERs were not required, and consequently MARAD never docketed an ER for either stage. The absence of an Operating License Stage ER goes directly to the LTP acceptance criteria. Existing licensees were required to submit an ER at the next major licensing action. For NSS, the first major licensing action after 10 CFR 51 became effective occurred in 1976 with the application for a Possession-only License. At that time, MARAD was completing work under RG 1.86 to place the NSS facility into a condition of mothballed protective storage. It is not clear, and there is no direct statement in any of the license amendments issued from 1976 onwards, that the ER submitted as part of the Possession-only License is considered an Operating License Stage ER. Internal MARAD correspondence surrounding the 1976 amendment suggests that MARAD subrogated its NEPA compliance to NRC's licensing actions; a position the agency maintained until contemporary DECON planning began (see Chapter 2 of this LTP for a description of the DECON planning effort).

Among other things, Section 102 of NEPA requires all federal agencies to include in every recommendation or report on proposals for … major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on --

(i) reasonably foreseeable environmental effects of the proposed agency action;

[36] No copy of this letter has yet been found in MARAD files, ADAMS, or the NRC public document room. Its content is inferred, with confidence, from available internal agency correspondence of the same timeframe contained in folders related to the Possession-only License amendment request.

(ii) any reasonable foreseeable adverse environmental effects which cannot be avoided should the proposal be implemented;

(iii) a reasonable range of alternatives to the proposed agency action, including an analysis of any negative environmental impacts of not implementing the proposed agency action in the case of a no action alternative, that are technically and economically feasible, and meet the purpose and need of the proposal;

(iv) the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and,

(v) any irreversible and irretrievable commitments of Federal resources which would be involved in the proposed agency action should it be implemented.

It is certainly true that NEPA compliance evolved and became more sophisticated over time. By subrogating its NEPA compliance to NRC licensing actions in 1976, MARAD acted in accordance with the common understanding of NEPA of that time. When MARAD resumed decommissioning consideration in 2002, it was apparent that the older interpretation was no longer appropriate, and that the agency would require its own NEPA analysis and conclusions to support the decommission action. This approach was consistent with contemporary MARAD and DOT NEPA compliance requirements. In 2005 MARAD entered into a reimbursable working agreement with the DOT Volpe Center to produce an EA in support of decommissioning planning. This work reached the stage of public comment in August 2006, with public meetings in the port cities of Norfolk, VA, Wilmington, NC and North Charleston, SC as part of the decommissioning site selection process. Shortly thereafter, the effort was forestalled by the same conditions which prompted the withdrawal of PSDAR Rev. 0. It was eventually completed in March 2008, with a corresponding FONSI issued in May 2008. Section 8.3 of this chapter describes the 2008 EA and FONSI in more detail.

The 2008 EA analyzed effects at several possible decommissioning locations. The PSDAR Rev. 1 submittal included a commitment to prepare a supplement to the EA once a decommissioning site was selected. NSS was moved to Baltimore in May 2008, within the context of preparing the ship for extended protective storage (SAFSTOR) – consistent with the EA and with the PSDAR Rev. 1 submitted later in that year (see Section 8.4 of this chapter). With respect to DECON, it would be nearly ten years before MARAD was in a position to consider potential sites for the major industrial dismantlement activities leading towards termination of the NS-1 license. This length of time itself necessitated a new look at the 2008 EA.

The funds appropriated in FY 2017 supported development of CR-137, *Supplemental Environmental Assessment and Finding of No Significant Impact* [Reference 8-4]. The supplement was specifically prepared to support DECON, and included considerations not previously addressed, such as the modification of the ship's cargo holds and other spaces to support DECON requirements for waste material handling and packaging, as opposed to performing those functions on adjacent land. The SEA and FONSI were published in April 2019 and are described in Section 8.5 of this chapter.

Taken together, the 2008 EA/FONSI and 2019 SEA/FONSI adequately describe the site-specific impacts of NSS license termination activities. MARAD requests that these documents together be considered the 10 CFR 51.53(d) "Supplement to Applicant's Environmental Report—Post Operating License Stage." In this context, MARAD affirms that there is no new information or significant environmental changes associated with the site-specific termination activities up to the submission of the LTP; nor does MARAD expect there to be any new information or significant environmental changes in the period leading to license termination.

8.2 Purpose

This chapter summarizes MARAD's previously docketed environmental report (the 2008 EA and FONSI), the environmental considerations described in its PSDAR, and describes the 2019 SEA and FONSI. Collectively, these documents are considered the 10 CFR 51.53(d) "Supplement to Applicant's Environmental Report—Post Operating License Stage." The chapter concludes that there is no new information or significant environmental change associated with the site-specific decommissioning and license termination activities presented in this LTP.

8.3 Initial Environmental Assessment

The 2008 EA was derived from the incomplete draft of the 2006 DECON EA and was completed for the purpose of supporting MARAD's decision-making on decommissioning. While decommissioning, as a condition of the NS-1 license, was not a discretionary act by MARAD, the timing and methodology to be employed were actions which required independent NEPA analysis by MARAD. The actions themselves were limited to the NRC decommissioning methods of SAFSTOR, DECON and ENTOMB, as well as the statutorily-required No Action[37] alternative. The EA was prepared in accordance with MARAD and DOT NEPA compliance requirements. The EA drew heavily on the NRC GEIS and its supplement. For each decommissioning method, the environmental impacts were considered based on locating NSS in an east coast port city with existing industrial facilities suitable for receiving the ship, and performing the activities necessary to complete the method chosen – to include transportation of waste packages to disposal facilities at Clive, UT and Barnwell, SC. In this respect the EA, even though prepared by MARAD, mirrored the requirements for an NRC-prepared EA, as described in 10 CFR 51.30. The EA also met the generic requirements for an applicant's environmental report, as described in 10 CFR 51.45. The EA considered the following range of impacts: air quality, water quality, navigation, hazardous materials, public health & safety, socioeconomics & environmental justice, coastal resources, wildlife & vegetation, and historic resources (NHPA) within the port complexes of Hampton Roads, VA, Baltimore, MD, and Charleston, SC. The EA also included a Section 4(f) analysis pursuant to the Department of Transportation Act (49 U.S.C. Sec. 303(c)).

MARAD prepared and published a Finding of No Significant Impact based on the EA. The FONSI documented MARAD's conclusion that the proposed federal action to decommission the NSS was consistent with existing national environmental policies and objectives set forth in Section 101(a) of the NEPA. MARAD concluded that the proposed action would not significantly affect the quality of the human environment or otherwise include any condition requiring consultation pursuant to Section 102(2)(c) of NEPA. As noted previously, MARAD submitted the EA and FONSI to NRC in October 2008 for its information and use.

8.4 Post-Shutdown Decommissioning Activities Report

The PSDAR [Reference 8-5] was submitted on December 11, 2008. Within the PSDAR, MARAD described its intention to return NSS to a condition of protective storage based on contemporary SAFSTOR criteria. The option to transition to DECON was discussed in the PSDAR but was not the focus of agency efforts at the time. The PSDAR described MARAD's planned decommissioning operations and summarized the conclusions of the Initial EA. In Section 1.3, the PSDAR states the following:

[37] Under the No Action alternative, all work on NSS would cease and the ship would be returned to MARAD's James River Reserve Fleet site for indefinite retention (note, the retention time was indeterminate, but remained bounded by the 60-year rule).

On May 14, 2008 (see Reference d), the Maritime Administration published notice of the availability of the Finding of No Significant Impact (FONSI), Reference (e), derived from the March 2008 Environmental Assessment (EA) regarding the Decommissioning of the Nuclear Ship Savannah, Reference (f). The FONSI documents the Agency's conclusion that the proposed federal action to decommission the NSS is consistent with existing national environmental policies and objectives as set forth in Section 101(a) of the National Environmental Policy Act of 1969, as amended (NEPA). The Agency concluded that the proposed action will not significantly affect the quality of the human environment or otherwise include any condition requiring consultation pursuant to Section 102(2)(c) of NEPA.

Because the NSS is a federally-owned facility, NEPA required that the Maritime Administration prepare an evaluation of the available alternatives for the NSS prior to the Agency making an executive decision on decommissioning. A draft EA was released for public comment in 2006. Like Revision 0 of the PSDAR, that EA emphasized the DECON approach. The 2008 final EA incorporates public comments received, and expands the discussion and evaluation of the SAFSTOR decommissioning alternative. The EA has been independently evaluated by the Maritime Administration and determined to adequately and accurately discuss the environmental issues and impacts of the proposed project. Because the Agency concluded that the proposed action will not significantly affect the quality of the human environment or otherwise include any condition requiring consultation, preparation of an Environmental Impact Statement, pursuant to NEPA, was not required. The FONSI was published instead.

Section 7.3 of the PSDAR summarizes the Environmental Impacts Conclusions as follows:

... the potential environmental impacts associated with decommissioning the NSS have already been postulated in and will be bounded by the Finding of No Significant Impact (FONSI), Reference (e), which based on the Environmental Assessment (EA) regarding the Decommissioning of the Nuclear Ship Savannah, Reference (f).

The EA documents the available decommissioning alternatives for the NSS. The EA has been independently evaluated by the Maritime Administration and determined to adequately and accurately discuss the environmental issues and impacts of the proposed project.

8.5 Supplement Environmental Assessment

In 2017, the President's Budget Request included funding for DECON and license termination. Prior to the 2017 budget approval, MARAD affirmed internally that a supplement to the 2008 EA (SEA) would be required to: a) account for the passage of ten years from the initial EA, b) more fully evaluate the effects of the DECON process, and c) identify suitable locations to carry out the DECON work. Decommissioning funds were appropriated in March 2017 and allowed MARAD to contract for the analysis. In April 2019,[38] MARAD completed and published CR-137, *Supplemental Environmental Assessment and Finding of No Significant Impact* [Reference 8-4]. This document clearly states that the proposed action is to decommission NSS and terminate the NS-1 license using the DECON method, and that the preferred location to carry out these activities is Baltimore, MD. It describes:

[38] Numerous reports to NRC state that Phase I officially began on October 1, 2017. The decommissioning activities carried out by MARAD from the effective date of appropriations availability (May 2017) were limited to those activities determined to be within the scope of the 2008 EA and the PSDAR. DECON-specific activities commenced only after the 2019 EA/FONSI was complete and published.

- An analysis of the potential environmental consequences that may result from implementation of the alternatives for proposed decommissioning actions and all reasonably foreseeable, connected actions; and,
- The identified and analyzed potential effects on the natural and human environment in sufficient detail to determine the significance of impacts on the affected environment so that a preferred alternative and location may be selected and the decommissioning of NSS's nuclear power plant may be implemented.

Section 8.5.1 describes the Proposed Action and Alternatives evaluated in the SEA. Section 8.5.2 describes the conclusion of the SEA and FONSI.

8.5.1 Proposed Action and Alternatives Evaluated in the SEA

As described in the SEA, its purpose is to evaluate the Proposed Action:[39]

... to reduce residual radioactivity to levels that allow termination of the NRC license. Low Level Radioactive Waste (LLRW) would be segregated and enclosed while still onboard the vessel, removed from the vessel via crane directly onto the transportation mode (rail, highway, barge), and transported to licensed/permitted facilities for final disposal following Federal and/or State regulations.

The SEA assumes the project would be completed in three phases:

- Phase 1 would include pre-decommissioning planning, engineering, hazardous materials abatement, infrastructure preparation, and license amendment actions (which would be completed at Pier 13, Baltimore, MD) that takes about two years.
- Phase 2 would include the removal of the systems, structures, and components related to the nuclear power plant and disposal of these items at licensed radioactive waste disposal facilities in the United States, which takes about four years.
- Phase 3 would include a final status survey conducted by MARAD and a confirmatory survey conducted by an independent verification contractor for the NRC. Following these survey activities, the NRC would review the results of these surveys and if LTP requirements were met, the NRC would terminate the license.

The SEA concentrated its review on Phase 2 activities. The DECON approach in the SEA is refined from the less-specific approach described in the 2008 EA. Significantly, the SEA considered the impacts of waste material handling and packaging (i.e., waste management) occurring: 1) solely on the vessel; 2) solely on land adjacent to the vessel; or 3) partially on the vessel and partially on land. Rigging, i.e., crane service, was not included in this impact consideration as it was separately considered under the site screening criteria. The SEA describes MARAD's choice of Option 1. In selecting this option, MARAD minimized impacts to adjacent property and avoided any need to extend the licensed envelope beyond the existing boundary. This option also necessitated the conversion of cargo holds 3 and 4 as described in the SEA. The SEA differs from the 2008 EA in several other areas and again reflects a more mature project approach gained from experience during NSS protective storage operations. Among the differences are:

[39] Again, as in the 2008 EA, the statutory "No Action" alternative is also evaluated. MARAD did not include an alternative to seek an exemption to the decommissioning requirement.

- The Barnwell, SC waste repository is dropped from consideration; the Waste Control Specialists repository in Andrews, TX is added.
- Site screening criteria were refined. The principal screening criteria included: existing industrial facility characteristics and permitted uses, adequate waterway access without need for dredging, laydown space for a 100 ton landside crane and adjacent space for decommissioning activities (principally parking and truck access), adequate infrastructure for a 1,000 ton crane (either floating or land/pierside polar), multiple transportation options available (road, rail, barge), remote access, isolated from residential housing, and proximity to Baltimore to minimize relocation costs.

Under these factors, the port of Charleston, SC was dropped from consideration, and the port of Philadelphia, PA was added. Each of the three ports considered, Baltimore, Philadelphia and Hampton Roads, are considered a Proposed Action Alternative in the SEA.

The SEA presents a review and analysis of the potential environmental impacts associated with the three Proposed Action Alternative locations, as well as the No-Action Alternative. The SEA documents MARAD's evaluation and assessment of the potential environmental impacts associated with the decommissioning of NSS's nuclear power plant.

NEPA, CEQ regulations, and MARAD's procedures for implementing the NEPA specify that an EA should only address those resource areas potentially subject to impacts. In addition, the level of analysis should be commensurate with the anticipated level of environmental impact.

The location of all three Proposed Action Alternatives was in developed areas along a waterfront with restricted access. Given these locations, the SEA described that construction of new facilities and dredging would not be required because all three locations have existing infrastructure and deep water to accommodate NSS and support decommissioning. Therefore, the proposed Federal action would not be expected to involve major construction activities at the alternative locations; instead, there would only be minor alterations to the NSS itself to aid in decommissioning actions.

MARAD determined that five environmental resources would be reasonably foreseeable and potentially affected by the Proposed Action. The SEA evaluated the following five resources in detail:

- Water Resources;
- Biological Resources;
- Air Quality;
- Waste Management; and,
- Health and Safety.

The environmental consequences associated with implementation of the Proposed Action and the No-Action Alternative are compared below in Table 8-1.

Table 8-1 Summary of Impacts

Resource Area	Baltimore, MD, Preferred Alternative	Hampton Roads, VA, Alternative	Philadelphia, PA, Alternative	No-Action Alternative
Water Resources	Minimal adverse impacts	Minimal adverse impacts	Minimal adverse impacts	No significant impacts
Biological Resources	No reasonably foreseeable takes are expected for marine mammals. No effect on Essential Fish Habitat.	No reasonably foreseeable takes are expected for marine mammals. No effect on Essential Fish Habitat.	No reasonably foreseeable takes are expected for marine mammals. No effect on Essential Fish Habitat.	No significant impacts
Air Quality	Insignificant temporary impacts	Insignificant temporary impacts	Insignificant temporary impacts	No impacts
Waste Management	No significant impacts	No significant impacts	No significant impacts	No impacts
Health and Safety	No significant impacts	No significant impacts	No significant impacts	No impacts

Using the same considerations that in all three potential locations, the NSS would be decommissioned at a commercial facility with all actions taking place on coastal land with controlled and limited access, MARAD determined that because no major construction or modifications to facilities would reasonably be required, the SEA required no detailed evaluation of the following resources:

- *Cultural Resources - There would be no effects to cultural resources at any industrial facility; Section 106 for the vessel is ongoing in a separate coordinated action;*
- *Land Use - There would be no change in land use as a result of the Proposed Action;*
- *Geology, Soils and Seismicity - There would be no effects to these resources;*
- *Aesthetics and Visual Resources - The vessel does not have aesthetic value that would be negatively affected. The Proposed Action does not have an effect on the existing visual character or quality of the possible decommissioning sites and their surroundings;*
- *Socioeconomics - The project would not have a negative effect on the state, local and regional economy, housing, or community services;*
- *Environmental Justice - This addresses environmental and human health conditions in minority and low-income communities; the Proposed Action would occur at an existing facility and would not require construction of new facilities within minority or low income communities. Waste disposal routes are discussed in Chapter 3 and would not have an impact on environmental justice. Thus, environmental justice concerns are not applicable;*

- *Transportation - The Proposed Action would not result in increased traffic or number of personnel at the vessel's current location or the decommissioning facilities' locations; waste transportation is part of decommissioning and discussed under waste management;*
- *Noise - The Proposed Action is considered a routine vessel movement and the decommissioning of its nuclear power plant would not generate any noise above and beyond what is routinely generated at these facilities;*
- *Utilities - There is no need to provide additional utilities for the Proposed Action;*
- *Emergency Services - There would be no effect on emergency services resulting from the Proposed Action; and,*
- *Wetlands and floodplains - The Proposed Action would not affect wetlands or floodplains.*

The SEA also analyzed cumulative impacts within the regions associated with each port complex. The analysis combined the Proposed Action with the impacts of other known past, present, and reasonably foreseeable future actions for each region, and concluded that there would be no cumulative effects in any analysis category.

NEPA requires consideration of possible conflicts between the Proposed Action and the Objectives of Federal, State, Regional and Local Land Use Plans, Policies and Controls. MARAD considered the following Federal Acts, Executive Orders, Policies and Plans: a) NEPA, b) Clean Water Act, c) Clean Air Act and General Conformity Rule, d) Coastal Zone Management Act, e) Endangered Species Act, f) Migratory Bird Treaty Act, g) Marine Mammal Protection Act, h) NHPA, and i) Executive Order 12372 (Intergovernmental Review of Federal Programs). State, Local and Regional Plans, Policies and Controls considered included Coastal Zone Management, Endangered Species, and Air Quality Management.

Other impacts considered included Energy Requirements and Conservation Potential, Irreversible or Irretrievable Commitment of Natural or Depletable Resources, the Relationship between Local Short-Term Use of the Human Environment and Maintenance and Enhancement of Long-Term Natural Resource Productivity, the Means to Mitigate and/or Monitor Adverse Environmental Impacts, and Any Probable Adverse Environmental Effects that cannot be Avoided and are not Amenable to Mitigation

8.5.2 Conclusion of the SEA and FONSI

The SEA concluded the following:

> *Overall, no significant environmental impacts are expected to occur as a result of the Proposed Action. NSS is listed in the National Register of Historic Places. Through consultation with the NRC, the National Park Service, the Advisory Council on Historic Preservation, and the Maryland Historical Trust, which serves as the SHPO [State Historic Preservation Office], a Programmatic Agreement [PA] will be implemented as mitigation efforts for DECON-LT. MARAD is in the process of finalizing the details of the PA, which will formally document the agreed upon mitigation measures required for Section 106 compliance.*
>
> *The Proposed Action would comply with all Federal and state regulations, guidelines, and agreements. All Proposed Action Alternatives are environmentally equal. However, Baltimore, MD is the Preferred Alternative because the vessel is already there and may not need towing. There would be minor differences with respect to towing distances and waste transportation and disposals depending on the alternatives; however, none of the differences would produce significant impacts. Based on the findings from this EA, a FONSI shall be prepared.*

The FONSI concluded:

> *Conclusion and Approval: After careful and thorough consideration of the facts contained herein, and in the Supplemental EA, the undersigned finds that the proposed federal action is consistent with existing national environmental policies and objectives set forth in Section 101(a) of NEPA and that it will not significantly affect the quality of the human environment of [SIC or] otherwise include any condition requiring consultation pursuant to Section 102(2)(c) of NEPA. Therefore, a FONSI is warranted, and preparation of an EIS, pursuant to NEPA is not required. This FONSI is based on the attached Supplemental EA, which has been independently evaluated by MARAD and determined to adequately and accurately discuss the environmental issues and impacts of the proposed project. MARAD takes full responsibility for the accuracy, scope, and content of the attached Supplemental EA.*

8.6 Impact to PSDAR

Despite some refinement to decommissioning plans between the 2008 submittal of the PSDAR [Reference 8-5] and the 2017 commencement of activities, the DECON project remains within the scope of the PSDAR. The 2019 SEA did not identify any significant impacts beyond those already identified in the PSDAR. Thus, the 2019 SEA did not have any material impact to the PSDAR environmental considerations, and MARAD has concluded that completing decommissioning and license termination will not significantly affect the quality of the human environment.

As described in Chapter 3 of the LTP, MARAD will complete dismantlement activities prior to submittal of the LTP to NRC. Based on its experience conducting the Phase 2 activities described in the PSDAR and SEA, MARAD affirms that there is no new information or significant environmental change associated with the site-specific termination activities, nor is any such information or change foreseeable during the conduct of Phase 3 activities.

8.7 Environmental Considerations Regarding Vessel Disposition

Chapter 6 of the LTP discusses in some detail the potential end-state conditions of NSS, and the disposition alternatives that MARAD will pursue after the license is terminated. Although the disposition actions are not subject to NRC jurisdiction, they are obviously of interest to NRC when considering the LTP, and the request to terminate the license. The actions are, however, subject to NHPA consultation through the NSS PA, to which NRC is a signatory party. MARAD published a Final Programmatic Environmental Assessment covering its Ship Disposal activities in August 2009. This EA considers the environmental effects of vessel disposition via Ship Recycling, Ship Donation, Ship Sale for Use, Artificial Reefing and SINKEX. MARAD carries out its current ship disposal activities within the context of this EA and is preparing a comprehensive review and update to the EA in 2023. The disposition of NSS will be performed in accordance with the provisions of the updated EA.

8.8 Conclusion

Overall, decommissioning the NSS complies with all Federal and state regulations, guidelines, and agreements. No significant environmental impacts are expected to occur as a result of decommissioning and terminating the license of the NSS.

The SEA adequately described and analyzed the potential environmental impacts associated with decommissioning NSS and experience in Phase 2 to demonstrate that there is no new information or significant environmental changes associated with decommissioning and terminating the license of the NSS. Therefore, no new supplement is required, at this time.

8.9 References

8-1 Regulatory Guide 1.179, *Standard Format and Content for License Termination Plans for Nuclear Power Reactors*, Revision 2 – July 2019

8-2 NUREG-1700, *Standard Review Plan for License Termination Plan*, Revision 2, April 2018

8-3 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated October 3, 2008, - *Submittal of Finding of No Significant Impact and Environmental Assessment* (ML082810182)

8-4 CR-137, *Supplemental Environmental Assessment and Finding of No Significant Impact*, April 2019

8-5 Letter from Mr. Erhard W. Koehler (MARAD) to U.S. Nuclear Regulatory Commission (NRC), dated December 11, 2008, *Submittal of Post Shutdown Decommissioning Activities Report*, Revision 1

9 PORTIONS of FACILITY RELEASED prior to LTP APPROVAL

No parts of the facility were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H).

10 LTP AREAS that cannot be changed without NRC APPROVAL

Chapter 10 lists the LTP areas that cannot be changed without NRC approval. In Section 1.2, NUREG-1700 [Reference 10-1] states, in part:

> *In accordance with 10 CFR 50.82(a)(10), the LTP is approved by license amendment. Recognizing that there may be a need to make changes to the LTP following its approval by the NRC, the licensee should include a provision in the LTP that concerns such changes. Appendix 2* [(sic) B], *"LTP Areas That Cannot Be Changed Without NRC Approval," sets out such a provision that the NRC finds acceptable.*

For the NSS, the LTP areas that cannot be changed without NRC approval are derived from Appendix B of the NUREG.

MARAD will update the LTP in accordance with 10 CFR 50.71(e). After NRC approval, MARAD may make changes to the LTP, without prior NRC approval, in accordance with the criteria in 10 CFR 50.59, 10 CFR 50.82(a)(6), and 10 CFR 50.82(a)(7). Additionally, MARAD may make changes to the LTP without prior approval provided the proposed changes do not meet any of the following criteria:

- Require Commission approval under 10 CFR 50.59.
- Result in the potential for significant environmental impacts that have not previously been reviewed.
- Detract or negate the reasonable assurance that adequate funds will be available for decommissioning.
- Decrease a survey unit area classification (i.e., impacted to not impacted; Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3) without providing NRC a minimum 14 calendar day notification before implementing the change in classification.
- Increase the derived concentration guideline levels. Nominal values for the minimum detectable concentrations (MDCs) have been presented in Table 5-4 in the LTP. Using the methodology for calculating MDCs presented in Chapter 5 in the LTP, the actual MDCs will be calculated prior to performing the FSS. Therefore, increasing the MDCs does not require NRC approval.
- Increase the radioactivity level, relative to the applicable derived concentration guideline level, at which an investigation occurs.
- Change the statistical test applied to a test other than the Sign test. Note that the Wilcoxon Rank Sum test will not be used at the NSS.
- Increase the approved Type I decision error. Only Scenario A will be used in the FSS of the NSS. Therefore, changing the Type II error when using Scenario B is not applicable and does not require NRC approval.
- Change the approach used to demonstrate compliance with the dose criteria (e.g., change from demonstrating compliance using derived concentration levels to demonstrating compliance using a dose assessment that is based on final concentration data).
- Change parameter values or pathway dose conversion used to calculate the dose such that the resultant dose is lower than in the approved LTP and if a dose assessment is being used to demonstrate compliance with the dose criteria.

10.1 References

10-1 NUREG-1700, *Standard Review Plan for License Termination Plan*, Revision 2, April 2018

U.S. Department
of Transportation

**Maritime
Administration**

Office of Ship Operations

1200 New Jersey Ave., SE
Washington, DC 20590

Docket No. 50-238; License No. NS-1; N.S. *SAVANNAH*

ENCLOSURE 5 ACCEPTANCE CRITERIA REVIEW MATRIX

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
1	Chapter 1, General Information	RG 1.179	The licensee's name, address, license number, and docket number should agree with the most recent license.	Y	Section 1.6
2	Chapter 1, General Information	RG 1.179	The LTP should address each of the criteria from 10 CFR 50.82(a)(9) and 10 CFR 50.82(a)(10), and the related radiological criteria from Subpart E of 10 CFR Part 20 for unrestricted or restricted release of the site.	Y	Section 1.2, 1.3 and 1.4 all
3	Chapter 1, General Information	RG 1.179	These are the following seven chapters. The LTP should provide any supporting information necessary to address the criteria, including the following:	y	see lines 4 to 13 below
4	Chapter 1, General Information	RG 1.179	a. Describe the site characteristics.	y	Chapter 2
5	Chapter 1, General Information	RG 1.179	b. Identify remaining site dismantlement activities.	y	Chapter 3
6	Chapter 1, General Information	RG 1.179	c. Discuss plans for site remediation.	y	Chapter 4
7	Chapter 1, General Information	RG 1.179	d. Provide detailed plans for the final radiation survey for release of the site.	y	Chapter 5
8	Chapter 1, General Information	RG 1.179	e. Detail a method for demonstrating compliance with the radiological criteria for license termination.	y	Chapter 6
9	Chapter 1, General Information	RG 1.179	f.(1) Update site-specific estimates of remaining decommissioning costs	y	Chapter 7
10	Chapter 1, General Information	RG 1.179	f.(2) Include the estimated volume of radiological waste and	y	Chapter 7 see also Chap 3 and Chap 7, line 390
11	Chapter 1, General Information	RG 1.179	f.(3) Proposed disposal methods.	y	Chapter 7 see also Chap 3 and line 391

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
12	Chapter 1, General Information	RG 1.179	g. Provide a supplement to the environmental report, in accordance with 10 CFR 51.53, "Postconstruction Environmental Reports," that describes any new information or significant environmental change associated with the licensee's proposed termination activities.	y	Chapter 8
13	Chapter 1, General Information	RG 1.179	h. Identify parts, if any, of the facility that were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H).	y	Chapter 9
14	Chapter 1, General Information	NUREG 1700, SRP	The LTP is submitted in the form of a supplement to the FSAR or equivalent and the LTP has preceded or is accompanied by an application for license termination.	y	LAR cover letter
15	Chapter 1, General Information	NUREG 1700, SRP	The LTP is submitted 2 years or more before the proposed termination date of the license.	y	LAR cover letter
16	Chapter 1, General Information	NUREG 1700, SRP	The LTP is submitted in the form of a license amendment request.	y	LAR cover letter
17	Chapter 1, General Information	NUREG 1700, SRP	The LTP lists the name and address of the licensee;	y	Section 1.1, 1.6
18	Chapter 1, General Information	NUREG 1700, SRP	license number;	y	Section 1.1, 1.6
19	Chapter 1, General Information	NUREG 1700, SRP	docket number;	y	Section 1.1, 1.6
20	Chapter 1, General Information	NUREG 1700, SRP	facility name and address;	y	Section 1.1, 1.6

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
21	Chapter 1, General Information	NUREG 1700, SRP	size of the site in acres or square meters;	y	Ships are not sized in acres or square meters; Table 1-1, Figures 1-4 through 1-7
22	Chapter 1, General Information	NUREG 1700, SRP	the State and county in which the site is located;	y	Section 1.6
23	Chapter 1, General Information	NUREG 1700, SRP	the names of and distances to nearby communities, towns, and cities;	y	Section 1.6.2
24	Chapter 1, General Information	NUREG 1700, SRP	a description of the contours and features of the site;	y	Section 1.6 Figures 1-4 through 1-7
25	Chapter 1, General Information	NUREG 1700, SRP	the elevation of the site;	y	Section 1.6.1, Figure 1-4 and Table 1-1
26	Chapter 1, General Information	NUREG 1700, SRP	a description of property surrounding the site, including the location of all off-site wells used by nearby communities or individuals;	y	Section 1.6.2 Figures 1-8 and 1-9
27	Chapter 1, General Information	NUREG 1700, SRP	the location of the site relative to prominent features such as rivers and lakes;	y	Figure 1-9
28	Chapter 1, General Information	NUREG 1700, SRP	a map that shows the detailed topography of the site using a contour interval;	y	Section 1.6 Figures 1-4 through 1-7
29	Chapter 1, General Information	NUREG 1700, SRP	the location of the nearest residences and all significant facilities or activities near the site; and	y	Section 1.9
30	Chapter 1, General Information	NUREG 1700, SRP	a description of the facilities (buildings, parking lots, fixed equipment, etc.) at the site.	y	Section 1.6 Figures 1-4 through 1-7

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
31	Chapter 1, General Information	NUREG 1700, SRP	The LTP identifies all changes to the site boundaries (as defined in 10 CFR 20.1003, "Definitions") that have occurred. 10 CFR 50.75(g) requires licensee's to keep records that document any changes to the original site boundary such as any partial site release.	y	Section 1.6
32	Chapter 1, General Information	NUREG 1700, SRP, App A	Licensee Name and Address:	y	Section 1.6
33	Chapter 1, General Information	NUREG 1700, SRP, App A	Docket Number:	y	Section 1.6
34	Chapter 1, General Information	NUREG 1700, SRP, App A	Facility: name and address of the facility	y	Section 1.6
35	Chapter 1, General Information	NUREG 1700, SRP, App A	Facility: location and address of the site	y	Section 1.6
36	Chapter 1, General Information	NUREG 1700, SRP, App A	Facility: brief description of the site and immediate environs	y	Section 1.6
37	Chapter 1, General Information	NUREG 1700, SRP, App A	Facility: brief description of any changes to the original site bou	y	Section 1.6
38	Chapter 1, General Information	NUREG 1700, SRP, App A	Facility: summary of the licensed activities that occurred at the site	y	Section 1.6.3, Table 1-2
39	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: size of the site in acres or square meters	y	Ships are not sized in acres or square meters; Table 1-1, Figures 1-4 through 1-7
40	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: State and county in which the site is located	y	Section 1.6
41	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: names and distances to nearby communities, towns and cities	y	Section 1.6.2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
42	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: description of the contours and features of the site	y	Section 1.6 Figure 1-4 through 1-7
43	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: elevation of the site	y	Section 1.6.2, Figure 1-4 and Table 1-1
44	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: description of property surrounding the site, including the location of all off-site wells used by nearby communities or individuals	y	Section 1.6, Table 1-1
45	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: location of the site relative to prominent features such as rivers and lakes	y	Figures 1-8 and 1-9
46	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: a map that shows the detailed topography of the site using a contour interval	y	Figures 1-4 through 1-7
47	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: the location of the nearest residences and all significant facilities or activities near the site	y	Section 1.6.2
48	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: description of the facilities (buildings, parking lots, fixed equipment, etc.) at the site and the nature and extent of contamination at the site	y	Section 1.6.1 and 1.6.2 See Chapter 2 for contamination discussion
49	Chapter 1, General Information	NUREG 1700, SRP, App A	Site Description: decommissioning objective proposed by the licensee (i.e., restricted or unrestricted use)	y	Section 1.4, 1.5.

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
50	Chapter 2, Site Characterization	RG 1.179	The purpose of the site characterization is to ensure that the licensee conducts final radiation surveys in all areas where contamination existed, remains, or has the potential to exist or remain. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," issued August 2000 (Ref. 13), provides guidance on developing a site characterization program, and NUREG-1757 contains additional guidance.	y	Chapter 2
51	Chapter 2, Site Characterization	RG 1.179	The licensee can submit the entire site characterization package separately at any time before submitting the LTP and reference it in the LTP, or the licensee can submit the site characterization as an integral part of the LTP.	y	Section 2.1
52	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of structures,	y	Tables 2-9 through 2-13, Tables 2-21 and 2-22
53	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of systems	y	Tables 2-9 through 2-23
54	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of sewer systems	y	Section 2.1.4.

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
55	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of [rad] waste plumbing systems,	y	Section 2.1.4.; Table 2-14
56	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of floor drains,	y	Section 2.1.4.
57	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of ventilation ducts, and	y	Table 2-18
58	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of piping and	y	Tables 2-9, 2-10, 2-11, 2-14 and 2-19.
59	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of embedded piping,	y	Section 2.1.4.
60	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of rubble	y	Section 2.1.4.
61	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of paved parking lots - surface	y	Section 2.1.4.

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
62	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of paved parking – buried beneath the site.	y	Section 2.1.4.
63	Chapter 2, Site Characterization	RG 1.179	Ground water data	y	Section 2.1.4.
64	Chapter 2, Site Characterization	RG 1.179	Surface water data	y	Section 2.1.4.
65	Chapter 2, Site Characterization	RG 1.179	Components data	y	Tables 2-9 through 2-23
66	Chapter 2, Site Characterization	RG 1.179	Residues data	y	Tables 2-5, 2-6 and 2-7; Section 2.1.4, 2.4.2
67	Chapter 2, Site Characterization	RG 1.179	Environment data	y	Section 2.1.4.
68	Chapter 2, Site Characterization	RG 1.179	Maximum contamination levels - structures	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
69	Chapter 2, Site Characterization	RG 1.179	Maximum contamination levels - equipment	y	Tables 2-9 through 2-23
70	Chapter 2, Site Characterization	RG 1.179	Maximum contamination levels - soils	y	Section 2.1.4.
71	Chapter 2, Site Characterization	RG 1.179	Average contamination levels - structures	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
72	Chapter 2, Site Characterization	RG 1.179	Average contamination levels - equipment	y	Tables 2-9 through 2-23

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
73	Chapter 2, Site Characterization	RG 1.179	Average contamination levels - soils	y	Section 2.1.4.
74	Chapter 2, Site Characterization	RG 1.179	Ambient exposure rate measurements - structures	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
75	Chapter 2, Site Characterization	RG 1.179	Ambient exposure rate measurements - equipment	y	Tables 2-9 through 2-13 Tables 2-21 through 2-23
76	Chapter 2, Site Characterization	RG 1.179	Ambient exposure rate measurements - soils	y	Section 2.1.4.
77	Chapter 2, Site Characterization	RG 1.179	The site characterization should contain sufficiently detailed data to support planning for all remaining decommissioning activities and the final status survey program.	y	Tables 2-9 through 2-23
78	Chapter 2, Site Characterization	RG 1.179	The LTP should describe historic events (including dates, types of occurrences, and locations inside and outside the facility), such as radiological spills, onsite disposals, or other radiological accidents or incidents, that resulted or could have resulted in the contamination of structures, equipment, letdown areas, or soils and ground water beneath buildings and in outside areas.	y	Section 2.2.3
79	Chapter 2, Site Characterization	RG 1.179	Describe the survey instruments	y	Table 2-2 Section 2.3.2, 2.3.4, 2.3.5, 2.3.6

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
80	Chapter 2, Site Characterization	RG 1.179	Describe the supporting quality assurance (QA) practices used in the site characterization program.	y	Table 2-2 Section 2.3.2, 2.3.4, 2.3.5, 2.3.6
81	Chapter 2, Site Characterization	RG 1.179	Describe the how MARAD applied the data quality objectives discussed in NUREG-1575 during site characterization.	y	Section 2.3.4, 5.4.1
82	Chapter 2, Site Characterization	NUREG 1700, SRP	The LTP identifies all locations, both inside and outside the facility, where radiological spills, disposals, operational activities, or other radiological accidents and or incidents occurred and could have resulted in contamination. This identification should be done on a room-by-room or area-by-area basis as necessary, including equipment, laydown areas, or soils (subfloor and outside area).	y	Section 2.2.3
83	Chapter 2, Site Characterization	NUREG 1700, SRP	The LTP describes, in summary form, the original shutdown	y	Section 2.1
84	Chapter 2, Site Characterization	NUREG 1700, SRP	The LTP describes, in summary form, the current radiological status of the site	y	Tables 2-9 through 2-23
85	Chapter 2, Site Characterization	NUREG 1700, SRP	The LTP describes, in summary form, the current non-radiological status of the site	y	Table 2-9, Table 2-10, Table 2-11
86	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of structures,	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
87	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of systems	y	Tables 2-8 and 2-9, Tables 2-14 through 2-18
88	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of sewer systems	y	Section 2.1.4.
89	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of [rad] waste management systems,	y	Table 2-14
90	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of floor drains,	y	Section 2.1.4.

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
91	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of ventilation ducts, and	y	Table 2-18
92	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of piping and	y	Tables 2-9 through 2-23
93	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of embedded piping,	y	Section 2.1.4.
94	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of rubble	y	Section 2.1.4.
95	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of ground water	y	Section 2.1.4.
96	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of surface water	y	Section 2.1.4.
97	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of components	y	Tables 2-9 through 2-23
98	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of residues	y	Section 2.1.4.
99	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of the environment	y	Section 2.1.4.
100	Chapter 2, Site Characterization	NUREG 1700, SRP	Maximum contamination levels - structures	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
101	Chapter 2, Site Characterization	NUREG 1700, SRP	Maximum contamination levels - equipment	y	Tables 2-9 through 2-23
102	Chapter 2, Site Characterization	NUREG 1700, SRP	Maximum contamination levels - soils	y	Section 2.1.4.

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
103	Chapter 2, Site Characterization	NUREG 1700, SRP	Average contamination levels - structures	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
104	Chapter 2, Site Characterization	NUREG 1700, SRP	Average contamination levels - equipment	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
105	Chapter 2, Site Characterization	NUREG 1700, SRP	Average contamination levels - soils	y	Section 2.1.4.
106	Chapter 2, Site Characterization	NUREG 1700, SRP	Ambient exposure rate measurements - structures	y	Tables 2-9 through 2-13, Tables 2-21 through 2-23
107	Chapter 2, Site Characterization	NUREG 1700, SRP	Ambient exposure rate measurements - equipment	y	Section 2.3.3 Tables 2-10 through 2-12
108	Chapter 2, Site Characterization	NUREG 1700, SRP	Ambient exposure rate measurements - soils	y	Section 2.1.4.
109	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of paved parking lots - surface	y	Section 2.1.4.
110	Chapter 2, Site Characterization	NUREG 1700, SRP	Extent and range of rad contamination of paved parking – buried beneath the site.	y	Section 2.1.4.
111	Chapter 2, Site Characterization	NUREG 1700, SRP	Describe the survey instruments	y	Table 2-2 Section 2.3.2, 2.3.4, 2.3.5, 2.3.6
112	Chapter 2, Site Characterization	NUREG 1700, SRP	Describe the supporting quality assurance (QA) practices used in the site characterization program.	y	Section 2.3.4, 2.3.5, 2.3.6

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
113	Chapter 2, Site Characterization	NUREG 1700, SRP	Identify the survey instruments used in the site characterization program.	y	Table 2-2 Section 2.3.2, 2.3.4, 2.3.5, 2.3.6
114	Chapter 2, Site Characterization	NUREG 1700, SRP	Identify the supporting quality assurance practices used in the site characterization program.	y	Section 2.3.4, 2.3.5, 2.3.6
115	Chapter 2, Site Characterization	NUREG 1700, SRP	Identify the background levels used during scoping or characterization surveys.	y	Table 2-13
116	Chapter 2, Site Characterization	NUREG 1700, SRP	Describe in detail the areas and equipment that need further remediation to allow the reviewer to estimate the radiological conditions that will be encountered during remediation of equipment, components, structures, and outdoor areas.	y	Section 2.4
117	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Background Levels Used During Characterization Surveys	y	Section 2.3.5; Table 2-13
118	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radionuclides Present at Each Location - maximum radionuclide activities (in dpm/100cm2, pCi/gm or pCi/l)	y	Tables 2-9 through 2-23
119	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radionuclides Present at Each Location - average radionuclide activities (in dpm/100cm2, pCi/gm or pCi/l	y	Tables 2-9 through 2-23
120	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radionuclides Present at Each Location - maximum radionuclide ratios, if multiple radionuclides are present (in dpm/100cm2, pCi/gm or pCi/l)	y	Tables 2-9 through 2-23
121	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radionuclides Present at Each Location - average radionuclide ratios, if multiple radionuclides are present (in dpm/100cm2, pCi/gm or pCi/l)	y	Tables 2-9 through 2-23

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
122	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radiological Contamination - List or description of all structures, systems, and equipment at the facility where licensed activities occurred that contain residual radioactive Material in excess of site background levels	y	Tables 2-9 through 2-23
123	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radiological Contamination - Summary of the structures, systems, equipment, and locations at the facility that the licensee or responsible party has concluded have not been affected by licensed operations, and the rationale for the conclusion	y	Tables 2-3 through 2-6
124	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radiological Contamination - List or description of each room or area, and equipment within each of the contaminated structures	y	Table 2-8, Section 2.3.5 Table 2-12, and Section 2.3.6
125	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radiological Contamination - Summary or map of the locations of contamination in each room or work area	y	Tables 2-6, 2-9, 2-10, 2-11, & 2-12
126	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Radiological Contamination - Mode of contamination for each surface (i.e., whether the radioactive material is present only on the surface of the material or if it has penetrated the material)	y	Section 2.3.5, Table 2-10 and Table 2-11
127	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Characterization Surveys: description and justification of the survey measurements for affected media	y	Section 2.1.4.
128	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Characterization Surveys: survey results, including tables or charts of the concentrations of residual radioactivity measured	y	Tables 2-9 through 2-23
129	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Characterization Surveys: maps or drawings of the site, area, or building showing areas classified as impacted or not impacted, with justification for considering areas to be not impacted	y	Section 2.4.2, Table 2-25

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
130	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Surface and Subsurface Soil Contamination: list or description of all locations at the facility where surface and subsurface soil contains residual radioactive material in excess of site background levels	y	Section 2.1.4.
131	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Surface and Subsurface Soil Contamination: scale drawing or map of the site showing the locations of subsurface soil contamination	y	Section 2.1.4.
132	Chapter 2, Site Characterization	NUREG 1700, SRP, App A	Surface Water and Ground Water: summary of all surface water bodies and aquifer(s) at the facility that contain residual radioactive material in excess of site background levels	y	Section 2.1.4.
133	Chapter 2, Site Characterization	NUREG-1757, Vol 2 4.1.3	The characterization survey provides sufficient information to permit planning for site remediation that will be effective and will not endanger the remediation workers	y	Chap 2
134	Chapter 2, Site Characterization	NUREG-1757, Vol 2 4.1.3	The characterization survey provides sufficient information to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected	y	Section 2.1.4, 2.3.4, 2.3.5, 2.3.6
135	Chapter 2, Site Characterization	NUREG-1757, Vol 2 4.1.3	The characterization survey provides sufficient information to provide information that will be used to design the FSS	y	Chap 2
136	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	The characterization survey design is adequate to determine the radiological status of the facility.	y	Section 2.4
137	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	Describe the radiation characterization survey design	y	Section 2.1

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
138	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	Describe the results of the survey, including the following:	y	See Lines 139 to 152.
139	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description and justification of the survey measurements for impacted media (for example, building surfaces, building volumetric, surface soils, subsurface soils, surface water, ground water, sediments, etc., as appropriate)	y	Section 2.4
140	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description of the field instruments that were used for measuring concentrations	y	Section 2.3.2, 2.3.4, 2.3.5, 2.3.6
141	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description of the methods that were used for measuring concentrations	y	Section 2.3.4
142	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description of the sensitivities of those field instruments and methods;	y	Section 2.3.4, 2.3.5 Table 2-13
143	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description of the laboratory instruments that were used for measuring concentrations	y	Section 2.3.4, 2.3.5, 2.3.6
144	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description of the laboratory methods that were used for measuring concentrations	y	Section 2.3.4, 2.3.5
145	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A description of the sensitivities of those laboratory instruments and methods	y	Section 2.3.4, 2.3.5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
146	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	The survey results including tables or charts of the concentrations of residual radioactivity measured	y	Tables 2-9 through 2-23
147	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	Maps or drawings of the site, area, or building showing areas classified as non-impacted or impacted and visually summarizing residual radioactivity concentrations in impacted areas;	y	Table 2-25
148	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	The justification for considering areas to be non-impacted;	y	Section 2.4.1, 2.4.2
149	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A discussion of why the licensee considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected;	y	Section 2.1.4, 2.3.4, 2.3.5, 2.3.6
150	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A discussion of how areas and surfaces were surveyed	y	Section 2.3.4, 2.3.5, 2.3.6
151	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	A discussion of why areas and surfaces did not need to be surveyed - for areas and surfaces that were considered to be inaccessible or not readily accessible; and	y	Section 2.1

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
152	Chapter 2, Site Characterization	NUREG-1757, Vol 2, 4.2.3.1.3	For sites, areas, or buildings with multiple radionuclides, a discussion justifying the ratios of radionuclides that will be assumed in the FSS or an indication that no fixed ratio exists and each radionuclide will be measured separately (note that this information may be developed and refined during decommissioning and licensees may elect to include a plan to develop and justify final radionuclide ratios in the LTP).	y	Tables 2-14 to 2-19, Section 2.3.5
153	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include a discussion of the remaining tasks associated with the decontamination and dismantlement	y	Section 3.2
154	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include an estimate of the quantity of radioactive material to be released to unrestricted areas	y	Section 3.3.
155	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include the proposed control mechanisms (to prevent recontamination)	y	Chap 5.4.4 and added 3.1.4 referring to 5.4.4.

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
156	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include the proposed dose estimates	y	Section 3.4; Table 3-3
157	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include the proposed radioactive waste characterization.	y	Section 3.3; Table 3-2
158	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Identify any decommissioning tasks that require coordination with other Federal or State regulatory agencies and explain how that coordination will occur.	y	Section 3.1.3.
159	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Describe the areas and equipment that need further remediation in sufficient detail to allow the reviewer to predict the radiological conditions that will be encountered during remediation. The details in this Section should be sufficient for the NRC to identify any inspection or technical resources needed during the remaining dismantlement activities	y	Section 3.2
160	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	List the remaining activities that do not involve unreviewed safety questions or changes in a facility's technical specifications. This list should be sufficiently detailed for the NRC staff to confirm that remedial activities may in fact be carried out under 10 CFR 50.59	y	Section 3.1.3

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
161	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP discusses the remaining tasks associated with decontamination and dismantlement	y	Section 3.2
162	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP estimates the quantity of radioactive material to be shipped for disposal or processing	y	Section 3.3 Table 3-2
163	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP describes the proposed control mechanisms to ensure that areas are not re-contaminated	y	Chap 5.4.4 and added 3.1.4 refering to 5.4.4.
164	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP contains occupational exposure estimates	y	Section 3.4; Table 3-3
165	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP contains radioactive waste characterization	y	Section 3.3; Table 3-2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
166	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP describes the remaining dismantlement activities in sufficient detail for the NRC staff to identify any associated inspection or technical resources that will be needed.	y	Section 3.2
167	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP is sufficiently detailed to provide data for use in planning further decommissioning activities.	y	Section 3.2
168	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP includes decontamination techniques	y	Section 4.2
169	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP includes projected schedules	y	Section 3.5, Table 3-4
170	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP includes costs	y	Chap 7

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
171	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP includes waste volumes	y	Section 3.3; Table 3-2
172	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP includes dose assessments (including groundwater assessments)	y	Section 3.4; Table 3-3
173	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP includes health and safety considerations	y	Section 3.2, 5.7
174	Chapter 3, Identification of Remaining Site Dismantlement Activities	NUREG 1700, SRP	The LTP lists the remaining activities that do not require any additional licensing action	y	Section 3.2
175	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Gantt or PERT chart detailing the proposed remediation tasks in the order in which they will occur	y	Figure 3-1
176	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Statement acknowledging that circumstances can change during decommissioning	y	Section 3.5 and 4
177	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Statement acknowledging that if the licensee determines that the decommissioning cannot be completed as outlined in the schedule, the MARAD will provide an updated schedule to NRC	y	Section 3.5 and 4

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
178	Chapter 4, Remediation Plans	RG 1.179	Summarize any changes from the previously approved radiological control program that the licensee will use for the control of radiological contamination associated with the remaining decommissioning and remediation activities described in LTP Chapter 3	y	Section 4
179	Chapter 4, Remediation Plans	RG 1.179	Summarize changes to the radiation protection program, but these details should be provided in either periodic updates to the final safety analysis report or the LTP.	y	Section 4
180	Chapter 4, Remediation Plans	RG 1.179	Discuss in detail the remediation methods and techniques that the licensee will use to demonstrate that the facility and site areas meet the NRC criteria for license termination in Subpart E of 10 CFR Part 20	y	Section 4.2
181	Chapter 4, Remediation Plans	RG 1.179	Use of new techniques should be reviewed under the 10 CFR 50.59 criteria and described sufficiently for the NRC to perform a safety evaluation	y	Section 4 and 4.2
182	Chapter 4, Remediation Plans	NUREG 1700, SRP	Address any changes in the radiological controls to be implemented to control radiological contamination associated with the remaining decommissioning and remediation activities	y	Section 4 and 4.3
183	Chapter 4, Remediation Plans	NUREG 1700, SRP	Discuss in detail how facility and site areas will be remediated to meet the proposed residual radioactivity levels (DCGLs) for license termination. Discussions should focus on any unique techniques or procedures used to evaluate whether the DCGLs have been met including the following:	y	Section 4 and 4.5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
184	Chapter 4, Remediation Plans	NUREG 1700, SRP	Summarize the techniques that will be used to remediate building structures and components (e.g., scabbling, hydrolazing, grit blasting, etc.).	y	Section 4.2
185	Chapter 4, Remediation Plans	NUREG 1700, SRP	Summarize the equipment that will be decontaminated and how the decontamination will be accomplished.	y	Section 3.2 Section 4.2
186	Chapter 4, Remediation Plans	NUREG 1700, SRP	Summarize the radiation protection methods and control procedures that will be employed including a summary of the procedures already authorized under the existing license.	y	Section 4.3
187	Chapter 4, Remediation Plans	NUREG 1700, SRP	Commit to conduct decommissioning activities in accordance with approved written procedures.	y	Section 4
188	Chapter 4, Remediation Plans	NUREG 1700, SRP	Include a detailed description of the techniques that will be employed to remove or remediate surface and subsurface soils, groundwater, and surface water and sediments.	y	Section 1.6, 2.1.4, 4, 5.2, 5.8, 6.1
189	Chapter 4, Remediation Plans	NUREG 1700, SRP	Describe plans, if any, for onsite disposal of decommissioning waste.	y	Section 4
190	Chapter 4, Remediation Plans	NUREG 1700, SRP	Include a schedule that demonstrates how and in what time frames MARAD will complete the interrelated decommissioning activities.	y	Figure 3-1
191	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of the radiation protection methods and control procedures that will be employed	y	Section 4.3
192	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of the procedures already authorized under the existing license to conduct decommissioning activities in accordance with approved written procedures	y	Section 4.3

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
193	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of the procedures for which approval is being requested in the LTP to conduct decommissioning activities in accordance with approved written procedures	y	Section 4.3
194	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of any unique safety issues associated with remediating contaminated structures	y	Section 4.5
195	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of any unique safety issues associated with remediating contaminated systems	y	Section 4.5
196	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of any unique safety issues associated with remediating contaminated equipment	y	Section 4.5
197	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of any remediation issues associated with remediating contaminated structures	y	Section 4.5
198	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of any remediation issues associated with remediating contaminated systems	y	Section 4.5
199	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of any remediation issues associated with remediating contaminated equipment	y	Section 4.5
200	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of the remediation tasks planned for each room, area and/or system in the order in which they will occur	y	Section 4.5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
201	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Description of the remediation techniques that will be employed in each room, area, or system	y	Section 4.2
202	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of the removal and remediation tasks planned for surface and subsurface soil at the site in the order in which they will occur, including which activities will be conducted by licensee staff and which will be performed by a contractor	y	Section 4
203	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Description of the techniques that will be employed to remove or remediate surface and subsurface soil at the site	y	Section 4
204	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Summary of the remediation tasks planned for ground and surface water, in the order in which they will occur, including which activities will be conducted by licensee staff and which will be performed by a contractor	y	Section 4
205	Chapter 4, Remediation Plans	NUREG 1700, SRP, App A	Description the remediation techniques that will be employed to remediate the ground or surface water	y	Section 1.6, 2.1.4, 4, 5.2, 5.8, 6.1

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
206	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 4.3.1	The purpose of the review of the description of the remedial action support surveys is to verify that the licensee has designed these surveys appropriately and to assist the licensee in determining when remedial actions have been successful and that the FSS may commence. In addition, information from these surveys may be used to provide the principal estimate of residual radioactivity variability that will be used to calculate the FSS sample size in a remediated survey unit.	y	Sections 4 through 4.5 Section 5.2, 5.3 and 5.7
207	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 4.3.1.1.3	Describe the remedial action support survey field screening methods	y	Section 4.2. Section 5.2, 5.3 and 5.7
208	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 4.3.1.1.3	Describe the remedial action support survey field screening instrumentation	y	Section 5.2, 5.3 and 5.7.2
209	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 4.3.1.1.3	Demonstration that field screening should be capable of detecting residual radioactivity at the $DCGL_W$.	y	Section 5.2, 5.3 and 5.7.2
210	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 4.3.1.1.4	Describe the remedial action support survey field screening survey instrument sensitivity.	y	Section 4.2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
211	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the final status survey (FFS). The FSS is the radiation survey performed after an area has been fully characterized and remediated, and MARAD believes that the area is ready to be released. The purpose of the final status survey is to demonstrate that the plant and site meet the radiological criteria for license termination in Subpart E of 10 CFR Part 20.	y	Chap 5
212	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all equipment. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.	y	Section 5.1 -5.7
213	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all systems. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.	y	Section 5.1 -5.7
214	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all structures. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.	y	Section 5.1 -5.7
215	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all soils. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.	y	Section 1.5.4
216	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the method for ensuring that sufficient data are included for a meaningful statistical survey.	y	Section 5.8
217	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods the licensee will use to establish background radiation levels.	y	Sections 5.5.1, 5.7.4.2, 5.9.1. Tables 5-4 and 5-5
218	Chapter 5, Final Radiation Survey Plan	RG 1.179	Discuss variances in background radiation that can be expected	y	Sections 5.5.1, 5.7.4.2, 5.9.1. Tables 5-4 and 5-5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
219	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program to support field survey work	y	DQAP 4.0, 15.0, 16.0 17.0 18.0, and 19.0 Section 5.7, 5.7.2.2, 5.8, 5.11
220	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program to support laboratory analysis	y	DQAP 4.0, 13.0, 14.0, 15.0, 16.0 17.0 18.0, and 19.0 Section 5.7, 5.7.2.2, 5.8, 5.11
221	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA organization	y	DQAP 2.3
222	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for training and qualification requirements;	y	DQAP 3.3
223	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for survey instructions and procedures, including water, air, and soil sampling procedures;	y	DQAP 13.0 and 14.0 Section 5.11
224	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for document control;	y	DQAP 6.0 and 7.0
225	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of purchased items;	y	DQAP 6.0 and 7.0
226	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for inspections;	y	DQAP 11.0 and 15.0
227	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of survey equipment - handling,	y	DQAP 13.0 and 14.0 Section 5.3, 5.7.2
228	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of survey equipment - storage,	y	DQAP 14.0 Section 5.3, 5.7.2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
229	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of survey equipment - response checks;	y	DQAP 13.0 Section 5.3, 5.7.2
230	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for shipping of survey equipment	y	DQAP 13.0 Section 5.3
231	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for shipping of survey laboratory samples;	y	DQAP 14.0 Section 5.3
232	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for disposition of nonconformance items;	y	DQAP 16.0 Section 5.11
233	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for corrective action;	y	DQAP 17.0 Section 5.3, 5.7.1.4, 5.11
234	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for QA records; and	y	DQAP 18.0 Section 5.3; 5.7.2.2; 5.11
235	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for survey audits, including methods to be used for reviewing, analyzing, and auditing data.	y	DQAP 19.0 and 5.11
236	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the verification surveys and evaluations used to support the delineation of radiologically affected (contaminated) areas	y	Section 5.3, 5.5.3.5;
237	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the verification surveys and evaluations used to support the delineation of unaffected (uncontaminated) areas	y	Section 5.3, 5.5.3.5;
238	Chapter 5, Final Radiation Survey Plan	RG 1.179	Identify the major radiological contaminants.	y	Section 2.3, 6.4

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
239	Chapter 5, Final Radiation Survey Plan	RG 1.179	Discuss methods used for addressing hard-to-detect radionuclides.	y	Section 5.4.5, 6.4
240	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe access control procedures to avoid recontamination of clean areas.	y	Section 5.4.5
241	Chapter 5, Final Radiation Survey Plan	RG 1.179	Identify survey units having the same area classification.	y	Section 5.4.2, Table 2-24 and 2-25
242	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe scanning performed to locate small areas of elevated concentrations of residual radioactivity.	y	Section 5.3, 5.5
243	Chapter 5, Final Radiation Survey Plan	RG 1.179	Discuss levels established for investigating significantly elevated concentrations of residual radioactivity. Include survey instrument calibration and efficiency calculations.	y	Sections 5.3, 5.5.3
244	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the reference coordinate system established for the site areas.	y	Section 5.4.3
245	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Identify the major radiological contaminants	y	Section 2.3, 6.4
246	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Methods used for addressing hard-to-detect radionuclides	y	Section 5.4.5
247	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Access control procedures to control recontamination of clean areas	y	Section 5.4.4
248	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program to support field survey work	y	DQAP 4.0, 15.0, 16.0 17.0 18.0, and 19.0 Section 5.7, 5.7.2.2, 5.8, 5.11

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
249	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program to support laboratory analysis	y	DQAP 4.0, 13.0, 14.0, 15.0, 16.0 17.0 18.0, and 19.0 Section 5.7, 5.7.2.2, 5.8, 5.11
250	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA organization	y	DQAP 2.3
251	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for training and qualification requirements;	y	DQAP 3.3
252	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for survey instructions and procedures, including water, air, and soil sampling procedures;	y	DQAP 13.0 and 14.0 Section 5-11
253	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for document control;	y	DQAP 6.0 and 7.0
254	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for control of purchased items;	y	DQAP 6.0 and 7.0
255	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for inspections;	y	DQAP 11.0 and 15.0
256	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for control of survey equipment - handling,	y	DQAP 13.0 and 14.0 Section 5.3, 5.7.2
257	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for control of survey equipment - storage,	y	DQAP 14.0 Section 5.3, 5.7.2
258	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for control of survey equipment - calibration (NOT in RG 1.179),	y	DQAP 13.0 Section 5.3, 5.7.2
259	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for control of survey equipment - response checks;	y	DQAP 13.0 Section 5.3, 5.7.2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
260	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for shipping of survey equipment	y	DQAP 13.0 Section 5.3, 5.7.1.4
261	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for shipping of survey laboratory samples;	y	DQAP 14.0 Section 5.2, 5.7.1.4
262	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for nonconformance items;	y	DQAP 16.0 Section 5.11
263	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for corrective action;	y	DQAP 17.0 Section 5.3, 5.7.1.4, 5.11
264	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for QA records; and	y	DQAP 18.0 Section 5.3; 5.7.2.2; 5.11
265	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Describe the QA program for survey audits, including methods to be used for reviewing, analyzing, and auditing data.	y	DQAP 19.0 and 5.11
266	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Methods for surveying embedded and buried piping	y	Section 5.2
267	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP	Final survey plan meets the evaluation criteria defined in Section 4 of NUREG-1757, Vol. 2. Included below lines in 335 to 349	y	Section 5.4.5
268	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Summary table or list of the DCGLw for each radionuclide and affected media of concern	y	Section 2.3, 6.4
269	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	If Class 1 survey units are present, a summary table or list of area factors that will be used to determine the DCGLemc for each radionuclide and media of concern	y	Section 5.4.5.3

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
270	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	If Class 1 survey units are present, the DCGLemc for each radionuclide and medium of concern	y	Section 5.4.5.3
271	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	If multiple radionuclides are present, the appropriate DCGLw for the survey method to be used	y	Sections 5.8.1 and 5.8.2
272	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Discussion of why the licensee considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected	y	Section 2.1.4, 2.3.4, 2.3.5, 2.3.6
273	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	For areas and surfaces that are inaccessible or not readily accessible, a discussion of how they were surveyed or why they did not need to be surveyed	y	Section 5.5.1.4
274	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	For sites, areas, or buildings with multiple radionuclides, a discussion justifying the ratios of radionuclides that will be assumed in the final status survey or an indication that no fixed ratio exists and each radionuclide will be measured separately	y	Sections 5.8.1 and 5.8.2
275	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Remediation Survey: description of field screening methods and instrumentation	y	Sections 5.2, 5.4.1, 5.4.5, 5.5, Table 5-2
276	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Remediation Survey: demonstration that field screening should be capable of detecting residual radioactivity at 10-50 percent of the DCGL	y	Sections 5.2, 5.4.1, 5.4.5, 5.5.1.2, 5.5.3.3, 5.5.3.5 Table 5-2
277	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: brief overview describing the final status survey design	y	Section 5.5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
278	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description and map or drawing of affected areas of the site, area, or buildings classified by residual radioactivity levels (Class 1, Class 2, or Class 3) and divided into survey units with an explanation of the basis for division into survey units	y	Section 2.3, Sections 5.3- 5.7
279	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of the background reference areas and materials, if they will be used, and a justification for their selection	y	Section 5.8
280	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: summary of the statistical tests that will be used to evaluate the survey results • description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide	y	Sections 5.7 and 5.8
281	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: for in situ sample measurements made by field instruments, a description of the instruments, calibration, operational checks, sensitivity, and sampling methods with a demonstration that the instruments and methods have adequate sensitivity	y	Sections 5.7 and 5.8
282	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of the analytical instruments for measuring samples in the laboratory, including their calibration, sensitivity, and methods with a demonstration that the instruments have adequate sensitivity	y	DQAP 13.0 Section 5.2
283	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of how the samples to be analyzed in the laboratory will be collected, controlled, and handled	y	DQAP 4.0 Section 5.2,

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
284	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of the final status survey investigation levels and how they were determined	y	Section 5.5.3
285	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: summary of any significant additional residual radioactivity that was not accounted for during site characterization	y	Sections 5.3, 5.4,
286	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: summary of direct measurement results and/or soil concentration levels in units that are comparable to the DCGL, and whether data are used to estimate or update the survey unit	y	Section 5.2, Section 5.5.3
287	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of performance of confirmatory surveys	y	Section 5.5.3
288	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of performance of split sampling	y	Section 5.8 (note after Table 5-5)
289	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: description of performance of side by side measurements	y	Section 5.7, 5.11
290	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Design: summary of the direct measurements or sample data used to evaluate the success of remediation and estimate the survey unit variance	y	Section 5.5.3

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
291	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Quality Assurance Program to Support Final Surveys: description of the QA program management organization, the duties and responsibilities of each unit within the organization, how delegation of responsibilities is managed within the decommissioning program, and how work performance is evaluated	y	Section 5.11 and DQAP 2.3
292	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Quality Assurance Program to Support Final Surveys: description of the authority of each unit within the QA program	y	Section 5.11 and DQAP 2.3
293	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Quality Assurance Program to Support Final Surveys: organization chart of the QA program	y	Section 5.11 and DQAP 2.3
294	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Quality Assurance Program to Support Final Surveys: commitment that activities affecting the quality of site decommissioning will be subject to the applicable controls of the QA program, and activities covered by the QA program are identified in program-defining documents	y	Section 5.11 and DQAP 3.2
295	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Quality Assurance Program to Support Final Surveys: description of the self-assessment program to confirm that activities affecting quality comply with the QA program	y	Section 5.11 and DQAP 3.2.2 and 17.0
296	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Quality Assurance Program to Support Final Surveys: commitment that persons performing self-assessment activities will not have direct responsibilities in the area they assess	y	Section 5.11 and DQAP 3.2.2 and 17.0
297	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: overview of the results of the final status survey	y	Section 5.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
298	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: discussion of any changes that were made in the final status survey from what was proposed in the LTP	y	Section 5.10
299	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: description of the method by which the number of samples was determined for each survey unit	y	Section 5.7
300	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: summary of the values used to determine the number of samples and a justification for these values	y	Section 5.7
301	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: survey results for each survey unit including the number of samples taken for the survey unit, and a map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and Class 2 survey units and random locations for Class 3 survey units and reference areas	y	Section 5.7, 5.10
302	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: measured sample concentrations	y	Section 5.7
303	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: statistical evaluation of the measured concentrations, survey instrument calibration procedures, and survey instrument efficiency calculations	y	Section 5.8, 5.10
304	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: Final Status Survey Report: judgmental and miscellaneous sample data sets, reported separately from those samples collected for performing the statistical evaluation	y	Section 5.8, 5.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
305	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: discussion of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of DCGLW	y	Section 5.8, 5.10
306	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: statement that a given survey unit satisfied the DCGLW and the elevated measurement comparison if any sample points exceeded the DCGLW	y	Section 5.10
307	Chapter 5, Final Radiation Survey Plan	NUREG 1700, SRP App A	Final Status Survey Report: if survey unit fails, description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure and the effect that the failure has on the conclusion that the facility is ready for final radiological surveys; and if a survey unit fails, a discussion of the effect of the failure has on other survey unit information	y	Section 5.5.3; 5.10
308	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol 2 4.1.3	NRC staff should review the FSS design to determine whether the survey design is adequate for demonstrating compliance with the radiological criteria for license termination.	y	Section 5.5
309	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol 2 4.1.3	NRC staff should review the results of the FSS to determine whether the survey demonstrates that the site, area, or building meets the radiological criteria for license termination.	y	Section 5.10 NRC will review FSS reports

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
310	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol 2 4.1.3	NRC staff should note that NRC regulations require that LTPs include a description of the planned final radiological survey. Recognizing the flexible approach discussed in Section 2.2 of NUREG 1757, Vol 2 and that the MARSSIM approach allows certain information needed to develop the final radiological survey to be obtained as part of the remedial activities at the site, a licensee or responsible party may submit information on facility radiation surveys in one of two ways, as summarized below. Section 2.2 of NUREG 1757, Vol 2 provides additional relevant guidance.	y	Section 5.3 - 5.7
311	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.1.3	MARAD should list the DCGL(s) that will be used to design the surveys and to demonstrate compliance with the radiological criteria for release, including … (next 4 lines)	y	Tables 2-15 through 2-19
312	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.1.3	Include a summary table or list of the $DCGL_W$ for each radionuclide and impacted medium of concern;	y	Tables 2-15 through 2-19
313	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.1.3	Include a summary table or list of area factors that will be used for determining a $DCGL_{EMC}$ for each radionuclide and media of concern if Class 1 (refer to Appendix A.1 of this volume for classification of site areas) survey units are present;	y	Section 5.4.5.3, 6.12
314	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.1.3	Include the $DCGL_{EMC}$ for each radionuclide and medium of concern if Class 1 survey units are present; and	y	Section 5.4.5.3, 6.12

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
315	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.1.3	Include the appropriate $DCGL_W$ for the survey method to be used if multiple radionuclides are present.	y	Section 5.3, 5.4.5.3, 5.5
316	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.2	NRC staff should verify that, for each radionuclide and impacted media of concern, MARAD has provided a DCGLW and, if Class 1 survey units are present, a table of area factors.	y	Section 5.3, 5.4.5.3, 5.5, 6.7, 6.12
317	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.2	NRC staff should verify that the values presented are consistent with the values developed pursuant to the dose modeling, as discussed in Chapter 5 of NUREG 1715, Vol 2.	y	Section 5.3, 5.4.5.3, 5.5, 6.12
318	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, 4.1.4.2	If multiple radionuclides are present, MARSSIM Sections 4.3.2, 4.3.3, and 4.3.4 of NUREG 1575, Vol 2 describe acceptable methods to determine DCGLs appropriate for the survey technique	y	Section 5.5
319	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4	FSS primary object 1 of 3: verify survey unit classification,	y	Section 5.3
320	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4	FSS primary object 2 of 3: demonstrate that the potential dose from residual radioactivity is below the release criterion for each survey unit, and	y	Section 5.3
321	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4	FSS primary object 3 of 3: demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit.	y	Section 5.3
322	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4	Data provided by the FSS can demonstrate that all radiological parameters satisfy the established guideline values and conditions	y	Chap 5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
323	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1	The purpose of NRC staff's review is to verify that the design of the FSS is adequate to demonstrate compliance with the radiological criteria for license termination	y	Section 5.3 - 5.7
324	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	The information supplied by MARAD should be sufficient to allow NRC staff to determine that the FSS design is adequate to demonstrate compliance with the radiological criteria for license termination.	y	Chap 5
325	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a brief overview describing the FSS design;	y	Section 5.3
326	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description and map or drawing of impacted areas of the site, area, or building classified by residual radioactivity levels (Class 1, 2, or 3) and divided into survey units, with an explanation of the basis for division into survey units (maps should have compass headings indicated);	y	Section 5.5
327	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description of the background reference areas and materials, if they will be used, and a justification for their selection;	y	Table 5-5
328	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a summary of the statistical tests that will be used to evaluate the survey results, including the elevated measurement comparison, if Class 1 survey units are present; a justification for any test methods not included in MARSSIM; and the values for the decision errors (α and β) with a justification for α" values greater than 0.05;	y	Section 5.3, 5.4.1, 5.5, 5.8.1, 5.8.3

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
329	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide;	y	Section 5.7.2
330	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description of the instruments, calibration, operational checks, sensitivity, and sampling methods for *in situ* sample measurements, with a demonstration that the instruments and methods have adequate sensitivity;	y	Section 5.7.2
331	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description of the analytical instruments for measuring samples in the laboratory, including the calibration, sensitivity, and methodology for evaluation, with a demonstration that the instruments and methods have adequate sensitivity;	y	See note following Table 5-5
332	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description of how the samples to be analyzed in the laboratory will be collected, controlled, and handled; and	y	See note following Table 5-5
333	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.1.3	FSS design includes a description of the FSS investigation levels and how they were determined.	y	Section 5.5
334	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: Appendix A of NUREG 1757 Vol. 2, for general guidance on implementing the MARSSIM approach for conducting FSSes;	y	Section 5.1, 5.4, 5.5, 5.6, 5.8
335	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: Appendix B of NUREG 1757 Vol. 2, for guidance on alternative methods of FSS for simple situations;	y	Section 5.7

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
336	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Sections 4.4 and 4.6 for classifying areas by residual radioactivity levels and dividing areas into survey units of acceptable size;	y	Section 5.4
337	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Section 4.5 for methods to select background reference areas and materials;	y	Section 5.8
338	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: NUREG–1505, Chapter 13, for a method to account for differences in background concentrations between different reference areas;	y	Section 5.8
339	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Section 5.5.2 for statistical tests;	y	Section 5.4.1, 5.5.1.1, 5.8.1, 5.8.3
340	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: Appendix A of NUREG 1757 Vol. 2, Section A.7.2 for decision errors;	y	Section 5.4.1, 5.5.1.1
341	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Sections 6.5.3 and 6.5.4 for selection of acceptable survey instruments, calibration, and operational checkout methods;	y	Section 5.7.2
342	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Section 6.7 for methods to determine measurement sensitivity;	y	Section 5.5, 5.7
343	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: NUREG–1507 for instrument sensitivity information;	y	Section 5.7, 5.7.2.2
344	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Sections 5.5.2.4, 5.5.2.5, 5.5.3, 7.5, and 7.6 for scanning and sampling;	y	Section 5.7

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
345	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Section 7.7 for sample analytical methods (Table 7.2 of Section 7.7 provides acceptable analytical procedural references);	y	Section 5.7; 5.8
346	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Sections 7.5 and 7.6 for methods for sample collection;	y	Section 5.7; 5.8
347	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: MARSSIM Section 5.5.2.6 for survey investigation levels; and	y	Section 5.5.3
348	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Vol. 2, Rev.1 4.4.1.2	Evaluation Criteria: Appendix G of NUREG 1757 Vol. 2 for surveys for special structural or land situations.		N/A
349	Chapter 6, Dose Modeling	RG 1.179	The LTP should demonstrate that the dose from residual radioactivity that is distinguishable from background radiation per Subpart E of 10 CFR Part 20.	y	Section 6.4
350	Chapter 6, Dose Modeling	RG 1.179	The LTP should also demonstrate that residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA) (see 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use").	y	Chapter 4-Sections 4.1 and 4.4
351	Chapter 6, Dose Modeling	RG 1.179	The LTP should describe in detail the methods and assumptions used to demonstrate compliance with the 25-mrem (0.25-mSv)-per-year criterion.	y	Sections 6.6 to 6.8
352	Chapter 6, Dose Modeling	RG 1.179	NUREG-1757 Vol 2, Section 5 and its Appendix H provides additional guidance on how to demonstrate compliance with the unrestricted release. (See lines 367 to 385 below.)	y	See ACRM Seq # 367 to 385 below

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
353	Chapter 6, Dose Modeling	NUREG 1700, SRP	If MARAD desires an unrestricted release in accordance with the requirements of 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use," the LTP should describe the methods used to demonstrate compliance.	y	Sections 6.2 to 6.8
354	Chapter 6, Dose Modeling	NUREG 1700, SRP	The information that should be submitted in the LTP and the associated evaluation criteria are described in NUREG-1757.	y	See ACRM Seq # 367 to 385 below
355	Chapter 6, Dose Modeling	NUREG 1700, SRP	NUREG 1757, Vol 2, Group 4 [NSS and Waste Processors] and Group 5 [Power Plants, Fuel Facilities] Unrestricted release using site-specific information - 5.2 and Appendix I (see lines 367 to 385)	y	See ACRM Seq # 367 to 385 below
356	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: source term information, including nuclides of interest, configuration of the source, areal variability of the source	y	Section 6.4
357	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: description of the exposure scenario used to develop site-specific DCLGs, including a description of the critical group	y	Sections 6.6 and 6.9
358	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: description of the conceptual model of the site including the source term, physical features important to modeling the transport pathways, and the critical group	y	Sections 6.6 and 6.9

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
359	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: identification and description of the mathematical model used (e.g., hand calculations, DandD Screen v1.0, RESRAD v 5.81, etc.)	y	Sections 6.8 and 6.10
360	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: description of the parameters used in the analysis	y	Section 6.9
361	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: discussion about the effect of uncertainty on the results	y	Sections 6.7 and 6.9
362	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	For unrestricted release using site-specific information: input and output files or printouts, if a computer program was used	y	Sections 6.9.1 and 6.9.2 Attachments to CR-139, TSD No. 21-089 Calculations to Support NSS Surface Contamination DCGLs
363	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	ALARA Analysis: description of how the licensee or responsible party will achieve a decommissioning goal below the dose limit	y	Chapter 4-Sections 4.1 and 4.4
364	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	ALARA Analysis: quantitative cost-benefit analysis	y	Chapter 4-Sections 4.1 and 4.4
365	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	ALARA Analysis: description of how costs were estimated	y	Chapter 4-Sections 4.1 and 4.4
366	Chapter 6, Dose Modeling	NUREG 1700, SRP App A	ALARA Analysis: a demonstration that the doses to the average member of the critical group are ALARA	y	Chapter 4-Sections 4.1 and 4.4

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
367	Chapter 6, Dose Modeling	NUREG-1757, Vol. 1, Rev. 2 Pg 7-4	7.6 Group 4: Unrestricted Release with site-specific dose analysis and no ground water contamination Group 4 facilities have residual radiological contamination present in building surfaces and soils, but the licensee cannot meet, or chooses not to use, screening criteria, and the ground water is demonstrably not contaminated. The licensees are able to demonstrate that residual radioactive material may remain at their site but within the levels specified in NRC criteria for unrestricted use (10 CFR 20.1402, "Radiological Criteria for Unrestricted Use") by applying site-specific criteria in a comprehensive dose analysis. The LTP should characterize the location and extent of radiological contamination. The LTP should also identify the land use, exposure pathways, and critical group for the dose analysis.	y	Section 6.2
368	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3	Assess the potential doses resulting from exposure to residual radioactivity remaining at the end of the decommissioning process.	y	Section 6.4
369	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.4.2	Provide information (of sufficient detail) on the source term, exposure scenario(s), conceptual model(s), numerical analyses (e.g., hand calculations or computer models), and uncertainty.	y	Sections 6.2 to 6.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
370	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.5.2	the source term information including nuclides of interest, configuration of the source, areal variability of the source, and chemical form of residual radioactivity (i.e., provide possible chemical changes that may occur during the time period of interest). If the licensee used dose modeling to develop DCGLs, instead of estimating final concentrations and then entering them into the code, the licensee need not specifically address the spatial variability acceptance criteria at this time.	y	Sections 6.4, 6.8, 6.9, 6.10.
371	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Describe the compliance scenario including a description of the critical group (include Scenario Identification, Critical Group Determination and Exposure Pathways.	y	Section 6.9 and 6.10; Table 6-6
372	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Describe any other reasonably foreseeable or less likely but plausible scenarios considered;	y	Section 6.11
373	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Describe the conceptual model of the site including the source term, physical features important to modeling the transport pathways (the major assumptions in developing the model, both the hydrologic and environmental transport processes important at the site, the dimensions, location and spatial variability of the source term used in the model); and the critical group location and activities	y	Sections 6.1, 6.6 to 6.10
374	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Identify, describe and justify the mathematical model used (e.g., hand calculations, DandD v2.1, RESRAD v6.1);	y	Section 6.8, 6.9 and 6.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
375	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Describe the parameters used in the analysis;	y	Sections 6.9
376	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Describe about the effect of uncertainty on the results; and	y	Sections 6.7 and 6.9
377	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.3.2	Provide input and output files or printouts, if a computer program was used.	y	Sections 6.9.1 and 6.9.2 Attachments to CR-139, TSD No. 21-089 Calculations to Support NSS Surface Contamination DCGLs
378	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.5.1	Annual Dose is less than (or equal to) 0.25 mSv (25 mrem),	y	Section 6.10
379	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, 5.5.2	Commitment to use radionuclide-specific DCGLs and ensure that the total dose from all radionuclides will meet the requirements of Subpart E by using the sum of fractions.	y	Section 6.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
380	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Appendix I	Section I.2 presents NRC approaches for reviewing the conceptual representation of the radioactive source term at the site. This Section describes the areas of reviews pertaining to the existing radioactive material contamination and physical and chemical characteristics of the material. In addition, the Section presents recommended approaches for source-term abstraction for the purpose of performing the dose analysis.	y	Sections 6.4, 6.6 to 6.9
381	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Appendix I	Section I.3 focuses on areas of review and criteria for accepting modifications of pathways of the two generic critical group scenarios, the "resident farmer" and the "building occupancy" scenarios. Section I.3, also, along with Appendices L and M, discusses the information that should be provided for a licensee's justification for modifying default screening scenarios and associated pathways. It also presents approaches for establishing site-specific scenarios, critical groups, and/or sets of exposure pathways based on specific land use, site restrictions, and/or site-specific physical conditions.	y	Sections 6.1, 6.9 and 6.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
382	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Appendix I	Section I.4 provides approaches for developing site-conceptual models for dose analysis. This Section presents approaches—via the linkage of the source term with the critical group receptor and the use of applicable pathways and site-characterization data—for the assimilation of data to establish a site conceptual model. It also presents approaches for employing applicable mathematical models to simulate and calculate the release and transport of contaminants from the source to the receptor. This Section also presents discussions of the typical conceptual models used in the DandD and RESRAD codes. Additionally, the Section provides (a) information on the limitations of the DandD and RESRAD models and (b) review areas to ensure compatibility of the site conceptual model with the conceptual models embedded in the DandD and RESRAD codes.	y	Sections 6.1, 6.6 to 6.10

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
383	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Appendix I	Section I.5 presents approaches and criteria for NRC staff acceptance of computer codes/models. This Section discusses review aspects pertaining to specifications, testing, verification, documentation, and QA/QC of the licensee's codes/models. This Section also addresses reviews applicable to embedded numerical models for the source term, the exposure pathway models, the transport models, and the intakes or dose conversion models. In addition, the Section provides a discussion of the development of and a description of the DandD code, particularly the excessive conservatism of the Version 1 of the DandD code. Section I.5 also presents a generic description of the RESRAD/RESRAD–BUILD codes.	y	Section 6.8, 6.9
384	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Appendix I	Section I.6 describes approaches for the selection and modification of input parameters for dose modeling analysis and includes the use of default parameters from the DandD code in other models.	y	Sections 6.5 to 6.9

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
385	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Appendix I	Section I.7 addresses the acceptable criteria for treating uncertainties in the dose modeling analysis. Issues pertaining to uncertainty and sensitivity are described, and NRC staff recommended approaches for the resolution of these issues are addressed. Policy positions are presented regarding approaches both to uncertainty/sensitivity treatments and to specific percentile dose-distribution selection for the screening and site-specific analysis. NRC staff review of input parameter distributions for Monte Carlo analysis and generic description of sensitivity analysis, including statistical techniques, are also described.	y	Sections 6.7 to 6.9
386	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the LTP includes the following: a. Estimate the decommissioning costs remaining at the time of LTP submittal described in LTP Chapter 3 b. Compare the estimated remaining costs with the present funds set aside for decommissioning.	y	Sections 7.1, 7.2.2, 7.2.2.2, 7.3.1, 7.3.2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
387	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the decommissioning cost estimate evaluates the following seven cost elements, which are not meant to be all-inclusive: (1) cost assumptions used, including a contingency factor, (2) major decommissioning activities and tasks, (3) unit cost factors, (4) estimated costs of decontamination and removal of equipment and structures, (5) estimated costs of waste disposal, including applicable disposal site surcharges, (6) estimated final survey costs, and (7) estimated total costs.	y	Sections 7.1, 7.2.2, 7.2.2.2, 7.3.1, 7.3.2
388	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimate focuses on: • the remaining work and • provide details for each activity associated with the decommissioning, including the costs of labor, materials, equipment, energy, and services.	y	Sections 7.1, 7.2.2, 7.2.2.2, 7.3.1, 7.3.2
389	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimates is based on credible engineering assumptions that are related to all remaining major decommissioning activities and tasks.	y	Section 7.3.1

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
390	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimate includes: • the cost of the planned remediation actions, • the cost of transportation and disposal of the waste generated by the actions (from lines 10 and 11 estimated volume of radiological waste and proposed disposal methods), • and other costs that are appropriate for the planned actions. NUREG-1307, "Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities," issued January 2013, provides information on estimating waste disposal costs.	y	Sections 7.2.1, 7.2.2, 7.2.2.2, 7.3.1, 7.3.2
391	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimate includes no credit for the salvage value of equipment.	y	Chapter 7

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
392	Chapter 7, Site Specific Decommissioning Costs	NUREG 1700, SRP	Confirm the LTP decommissioning cost estimate includes an evaluation of the following cost elements associated with the remaining decommissioning and remediation activities described in LTP Chapter 3: • cost assumptions used, including a contingency factor (normally 25 percent) • major decommissioning activities and tasks • unit cost factors • estimated costs of decontamination and removal of equipment and structures • estimated costs of waste disposal, including applicable disposal site surcharges and transportation costs • estimated final survey costs • estimated total costs	y	Sections 7.1, 7.2.2, 7.2.2.2, 7.3.1, 7.3.2
393	Chapter 7, Site Specific Decommissioning Costs	NUREG 1700, SRP	Confirm the LTP focuses on detailed activity by activity cost estimates.	y	Sections 7.2.1, 7.2.2
394	Chapter 7, Site Specific Decommissioning Costs	NUREG 1700, SRP	Confirm the LTP also compares the funds available for decommissioning with the calculated total cost from the licensee's detailed cost analysis. In addition, Regulatory Guide 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors", explains in detail the methods for estimating decommissioning costs, as well as accepted financial assurance mechanisms.	y	Section 7.3

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
395	Chapter 7, Site Specific Decommissioning Costs	NUREG 1700, SRP	Confirm the LTP cost estimate is based on credible engineering assumptions, and the assumptions are related to all major remaining decommissioning activities and tasks and are consistent with the information identified in Sections A3 Identification of Remaining Site Dismantlement Activities and Remediation Plans and A.4 Final Radiation Survey Plan of this NUREG 1700, SRP.	y	Section 7.2
396	Chapter 7, Site Specific Decommissioning Costs	NUREG 1700, SRP	Confirm the LTP cost estimate includes the cost of the remediation action being evaluated, the cost of transportation and disposal of the waste generated by the action, and other costs that are appropriate for the specific case. The current version of NUREG-1307, "Report on Waste Burial Charges", provides guidance on estimating waste disposal costs	y	Section 7.2

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
397	Chapter 7, Site Specific Decommissioning Costs	NUREG 1700, SRP App A	Confirm the LTP cost estimate includes: • cost assumptions used, including a contingency factor and basis for each • cost estimate addressing the major decommissioning activities and tasks and their relationship to remaining dismantlement activities described in LTP Chapter 3 • description of the unit cost factors • estimated costs of decontamination and removal of equipment and structures • estimated costs of waste disposal, including applicable disposal site surcharges • estimated transportation costs • estimated final survey cost	y	Sections 7.1, 7.2.2, 7.2.2.2, 7.3.1, 7.3.2
398	Chapter 8, Supplement to the Environmental Assessment (EA)	RG 1.179	Confirm the EA supplement describes in detail the environmental impact of the site-specific termination activity.	y	8.5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
399	Chapter 8, Supplement to the Environmental Assessment (EA)	RG 1.179	Confirm the EA supplement compares the impact with previously analyzed termination activities (see NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," Supplement 1, "Regarding the Decommissioning of Nuclear Power Reactors," issued November 2002).	y	8.3 and 8.5
400	Chapter 8, Supplement to the Environmental Assessment (EA)	RG 1.179	Confirm the EA supplement analyzes the environmental impact of the site-specific activity. Include alternative actions and any mitigating actions.	y	8.5
401	Chapter 8, Supplement to the Environmental Assessment (EA)	NUREG 1700, SRP	Confirm the EA supplement describes changes to the data that have arisen since the licensee submitted its "Applicant's Environmental Report - Operating License Stage" or its "Applicant's Environmental Report - Operating License Renewal Stage," as appropriate.	y	Not applicable as described in Section 8.3
402	Chapter 8, Supplement to the Environmental Assessment (EA)	NUREG 1700, SRP	Confirm the EA supplement describes the potential environmental impacts associated with site specific termination activities from the time the LTP is submitted until the license is terminated.	y	8.5
403	Chapter 8, Supplement to the Environmental Assessment (EA)	NUREG 1700, SRP	Confirm the EA supplement states the licensee's determination regarding whether the activities and effects are bounded by the potential impacts described by any site-specific EIS or EA developed in support of licensing the facility, NUREG-0586 as supplemented, or the PSDAR. The EA supplement should also describe any proposed mitigation measures the licensee will take to avoid significant impact.	y	8.5

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP
404	Chapter 8, Supplement to the Environmental Assessment (EA)	NUREG 1700, SRP	Confirm the EA supplement identifies the parts, if any, of the facility or site that were released for use before approval of the license termination plan.	y	Chap 9
405	Chapter 8, Supplement to the Environmental Assessment (EA)	NUREG 1700, SRP App A	Confirm the EA supplement describes any new information or potential significant environmental impact(s) associated with the site-specific termination activities related to the end use of the site (the environmental evaluation does not have to address decommissioning activities but focuses on site end use)	y	8.5
406	Chapter 9, Portions of Facility Released Prior to LTP Approval	RG 1.179 1.h	Confirm identification of parts, if any, of the facility that were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H).	y	Chap 9