July 7, 2025

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SUBJECT: CONFIRMATORY SURVEY OF THE NUCLEAR SHIP SAVANNAH LOCATED IN BALTIMORE, MARYLAND DOCKET NUMBER 05000238; RFTA 24-003; DCN 5381-SR-01-0

Dear Ms. Hood:

The Oak Ridge Institute for Science and Education (ORISE) is pleased to provide the enclosed final report detailing the procedures and results of the independent confirmatory survey activities for the Nuclear Ship Savannah when it was docked in the Canton Marine Terminal in Baltimore, Maryland. NRC comments on the draft report were addressed in the final version.

Please feel free to contact me at Erika.Bailey@orau.org if you have any questions.

Sincerely,

Erika N. Bailey

Erika N. Bailey
Survey and Technical Projects Manager
ORISE

TJV:enb

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CONFIRMATORY SURVEY OF THE NUCLEAR SHIP SAVANNAH LOCATED IN BALTIMORE, MARYLAND

Timothy J. Vitkus, CHP

FINAL REPORT

Prepared for the
U.S. Nuclear Regulatory Commission

JULY 2025

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This document was prepared for the U.S. Nuclear Regulatory Commission (NRC) by the Oak Ridge Institute for Science and Education (ORISE) through Interagency Agreement 31310021S0033 between the NRC and the U.S. Department of Energy (DOE). ORISE is managed by Oak Ridge Associated Universities under DOE contract number DE-SC0014664.

CONFIRMATORY SURVEY OF THE
NUCLEAR SHIP SAVANNAH LOCATED IN
BALTIMORE, MARYLAND

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ABBREVIATIONS AND ACRONYMS

AEC	Atomic Energy Commission
Ag-108m	silver-108metastable
AL	action level
Am-241	americium-241
C-14	carbon-14
cm	centimeter(s)
cm^2	square centimeter
Co-60	cobalt-60
cpm	counts per minute
Cs-137	cesium-137
CV	containment vessel
DCGL	derived concentration guideline level
$DCGL_{EMC}$	elevated measurement comparison derived concentration guideline level
$DCGL_{Op}$	operational derived concentration guideline level
DER	duplicate error ratio
dpm/100 cm^2	disintegrations per minute per one hundred square centimeters
DQO	data quality objective
F-55	iron-55
FSS	final status survey
FTT	Fuel Transfer Tank
H-3	tritium
H_0	null hypothesis
H_A	alternative hypothesis
IL	investigation level
LSC	liquid scintillation counting
LTP	license termination plan
MARAD	U.S. Department of Transportation, Maritime Administration
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	minimum detectable activity
MDC	minimum detectable concentration
mrem/yr	millirem per year
MW_{th}	mega-watt thermal
NaI[Tl]	thallium-doped sodium iodide
Ni-63	nickel-63
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NSS	Nuclear Ship Savannah
NST	Neutron Shield Tank
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g, cm^2, of sample	picocurie per gram, picocurie per square centimeter, picocurie per sample
Pu-239/240	plutonium-239/240
PSP	project-specific plan
PSQ	principal study question
RA	remedial action
RCA	radiological control area
RESL	Radiological and Environmental Sciences Laboratory
ROC	radionuclide of concern
SOF	sum-of-fractions

Sr-90	strontium-90
SU	survey unit
Tc-99	technetium-99
TEDE	total effective dose equivalent

CONFIRMATORY SURVEY OF THE NUCLEAR SHIP SAVANNAH LOCATED IN BALTIMORE, MARYLAND

1. INTRODUCTION AND HISTORY

The Nuclear Ship Savannah (NSS) was the world's first nuclear-powered merchant ship developed and built as part of the "Atom for Peace" initiative. The ship's keel was laid in 1958 and then launched in 1959. The NSS was built to be powered by an 80-MW_{th} (mega-watt thermal) pressurized water nuclear reactor. The Atomic Energy Commission (AEC), predecessor to the U.S. Nuclear Regulatory Commission (NRC), authorized nuclear fueling and operation for testing and demonstration in July 1961. The AEC issued license NS-1 in 1965. NSS made several demonstration commercial voyages, and during what became the final voyage, evidence of fuel failure was detected where small amounts of fission products were released to the primary coolant. The ship docked in Galveston, Texas, during the latter half of 1968 for a fuel shuffle outage. By 1970, commercial operations had ended, the NSS's reactor was shut down, and the ship was removed from service. The reactor fuel was removed in 1971, the ship was rendered inoperable by 1976, and the license was amended to possession-only. The NSS was moved and docked for public display at the Patriots Point Naval and Maritime Museum in Mt. Pleasant, South Carolina, from 1981 through 1994. During that time, the ship was designated a National Historic Landmark in July 1991. The NSS was next placed in protective storage in the James River Reserve Fleet near Ft. Eustis, Virginia.

The NSS has been under the authority of the U.S. Department of Transportation's Maritime Administration (MARAD) throughout the ship's history, although, at times, management was shared with other federal agencies. Preliminary plans for NSS decommissioning began in 2003 when MARAD awarded a decommissioning planning contract, followed in 2004 by a contract awarded for radiological scoping and characterization surveys of the ship. A neutron activation analysis was also conducted in 2004 to determine the radionuclide activation product inventories within the reactor pressure vessel, reactor internals, and the Neutron Shield Tank. The ship was moved in 2008 from Ft. Eustis, Virginia, to its current location at the Canton Marine Terminal in Baltimore, Maryland. Characterization surveys of the ship began in 2008 and continued through 2020 and addressed radiologically controlled areas (RCAs). Remediation was performed where indicated, and radiological controls were discontinued.

MARAD submitted a license termination plan (LTP) to the NRC outlining the final plans to characterize, remediate, and then conduct final status surveys of the remaining radiologically impacted ship systems and structures (MARAD 2023). The LTP provides the processes that will be used to demonstrate that the NSS satisfies the NRC's Title 10 of the Code of Federal Regulations (10 CFR) 20.1402 criterion for license termination without restrictions and, thereby, providing reasonable assurance that the total effective dose equivalent (TEDE) from residual radioactivity distinguishable from background to the average member of the critical group will not exceed 25 mrem (0.25 mSv) per year. MARAD states within the LTP that all dismantlement activities have been completed. The current decommissioning activities are to identify any remaining locations of residual radioactivity in excess of guidelines, remediate such areas, then conduct the final status surveys (FSS) required to demonstrate compliance with the unrestricted release criteria. The criteria applied are to be based on final ship disposition which could include shipbreaking (dismantlement and scraping for salvage and/or disposal), reefing (sinking the ship for artificial marine habitat to enhance recreational and commercial fishing), or continued preservation as a historical site involving unrestricted public access.

Because the MARAD plans to request NSS release without radiological restrictions and to terminate the radiological license, the NRC requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory survey activities of NRC-selected areas within the NSS. The specific NRC-selected areas for confirmatory surveys were the Neutron Shield Tank/Fuel Transfer Tank, Steam Generators, and Containment Vessel Level 1 (Tank Top) and Level 3 (D - Deck). The safely accessible surfaces of these items/areas were surveyed as well as the Pressurizer. ORISE developed a project-specific plan (PSP) to support the confirmatory survey activities for these ship components (ORISE 2024).

2. SITE DESCRIPTION

The dock and work area for the NSS's decommissioning is located in Baltimore Harbor, Pier 13, in the harbor's Canton industrial district. The street address is given as 4601 Newgate Avenue, Baltimore, Maryland. Figure 2.1 shows an aerial view of Baltimore Harbor, and Figure 2.2 shows the location of 4601 Newgate Avenue and Pier 13 with the docked NS Savannah.

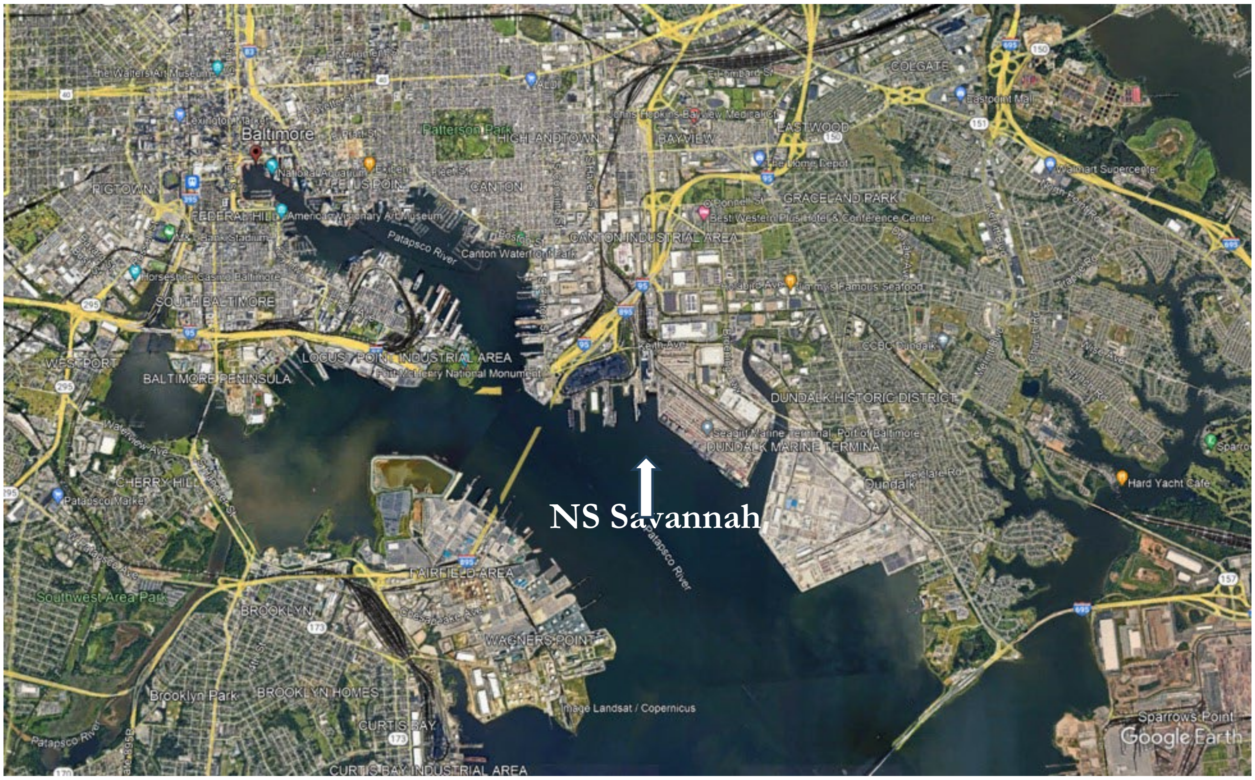


Figure 2.1. Aerial Image Showing Baltimore Harbor and Location of the NS Savannah (Google Earth Pro)

Figure 2.2. 4601 Newgate Avenue, Pier 13 with NS Savannah (Google Earth Pro)

The NSS itself is 181.5 meters (595.5 feet) in length, with a maximum beam width and height of 24 and 26 meters, respectively. The ship profile consists of multiple decks that are shown in Figure 2.3, which also indicates the radiological areas of the ship spanning from the Tank Top up to the Promenade Deck.

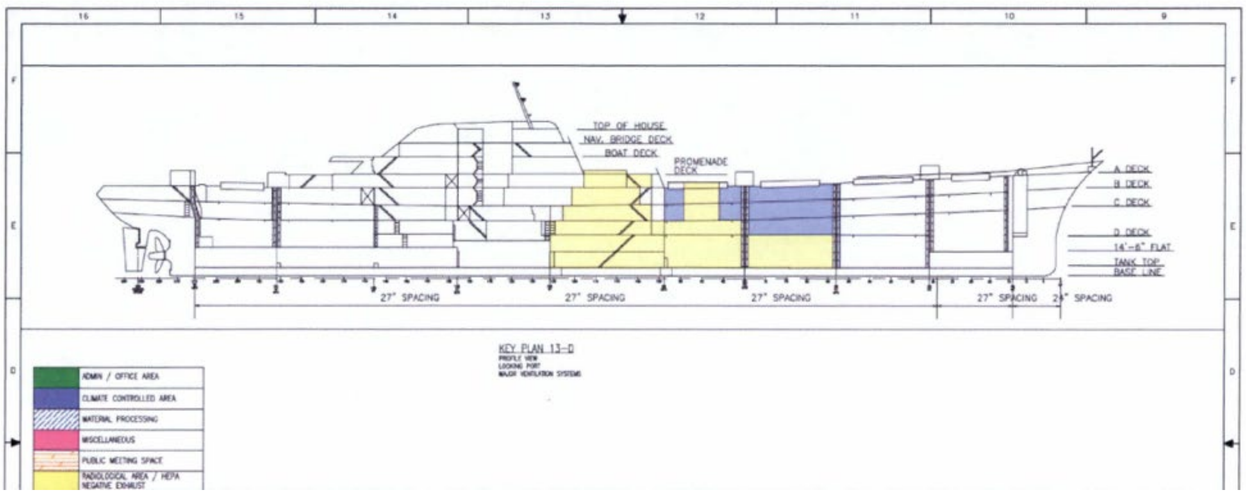


Figure 2.3. NS Savannah Profile
(yellow-coded areas are radiological/HEPA negative exhaust spaces)

Figure 2.4 provides a cutaway view of the reactor space and containment vessel, and Figure 2.5 illustrates the reactor components within the containment vessel.

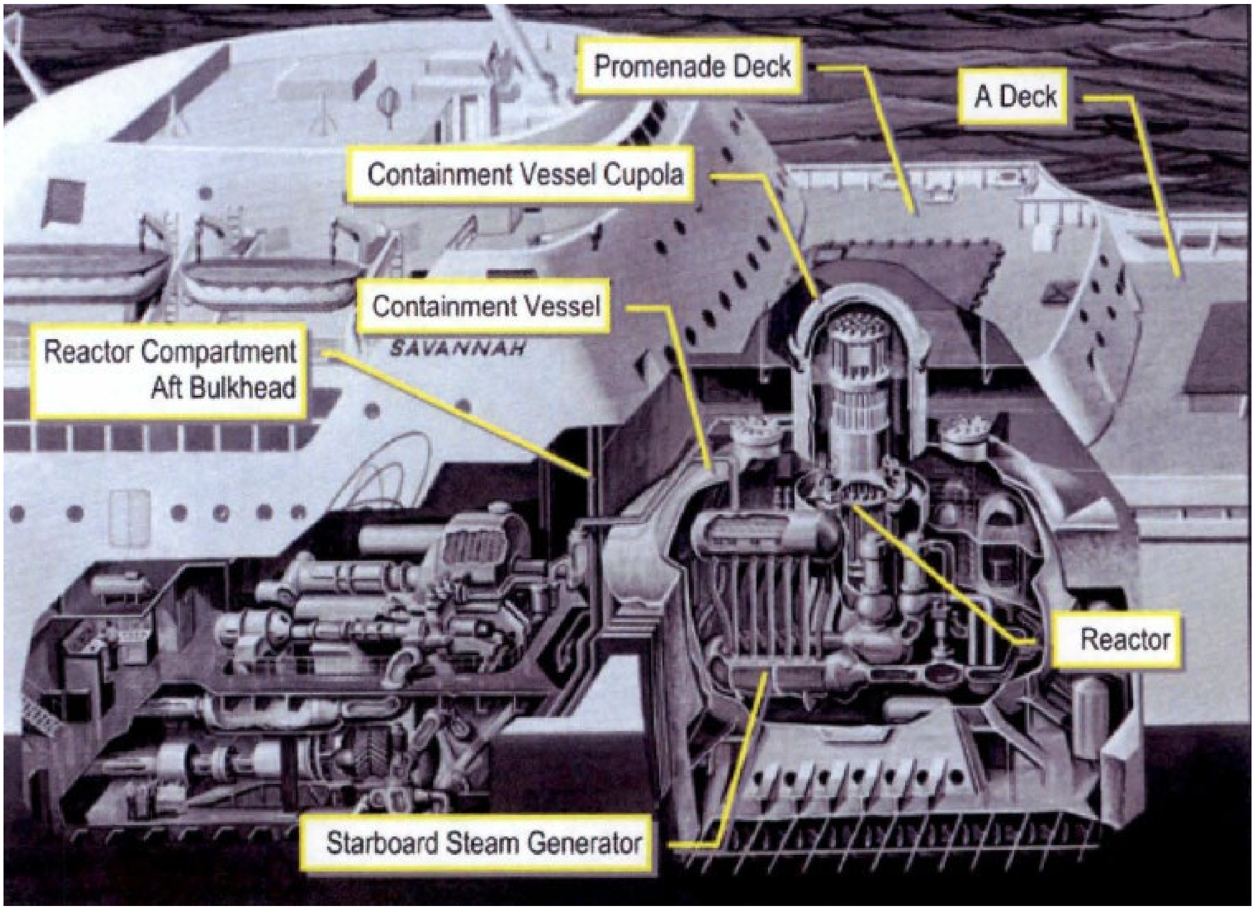
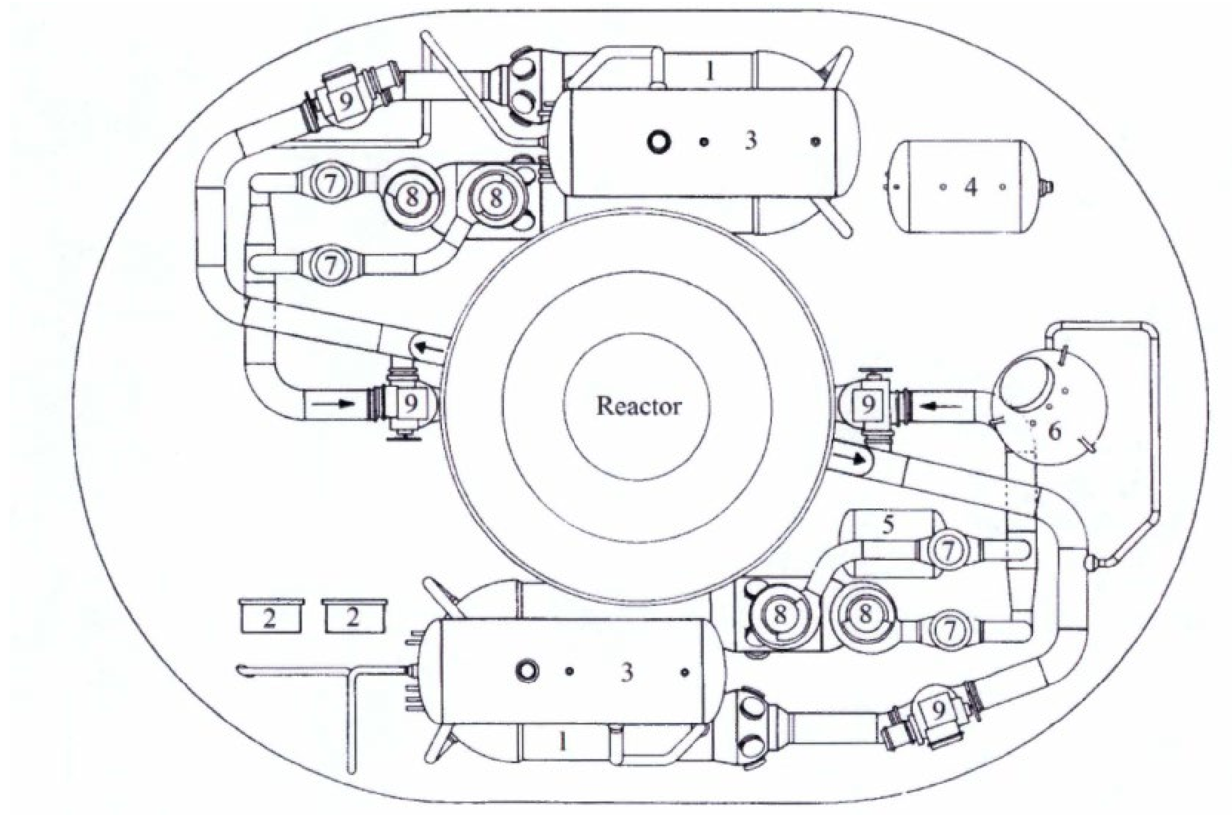


Figure 2.4. Cutaway View Showing Reactor and Components



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 2.5. Reactor and Components

Spaces in addition to the Reactor Compartment that had been designated as radiological control areas include, or at one point during the NSS's history included, the Hot Chemistry Lab, Port and Starboard Charge Pump Rooms, Port Stabilizer Room, Lower Secondary Room, Forward Control Room, the Health Physics Lab, the Reactor Compartment Upper Level, Containment Vessel, and Starboard Stabilizer Room. The LTP provides specific plot plans for these various spaces.

Of these spaces and components, the NRC is specifically targeting the Neutron Shield Tank/Fuel Transfer Tank (NST/FTT), Steam Generators (Figure 2.5), and Containment Vessel (Figure 2.4) for confirmatory survey. The Neutron Shield Tank is 5.4 meters in height and surrounds a portion of the reactor vessel illustrated in Figure 2.6.

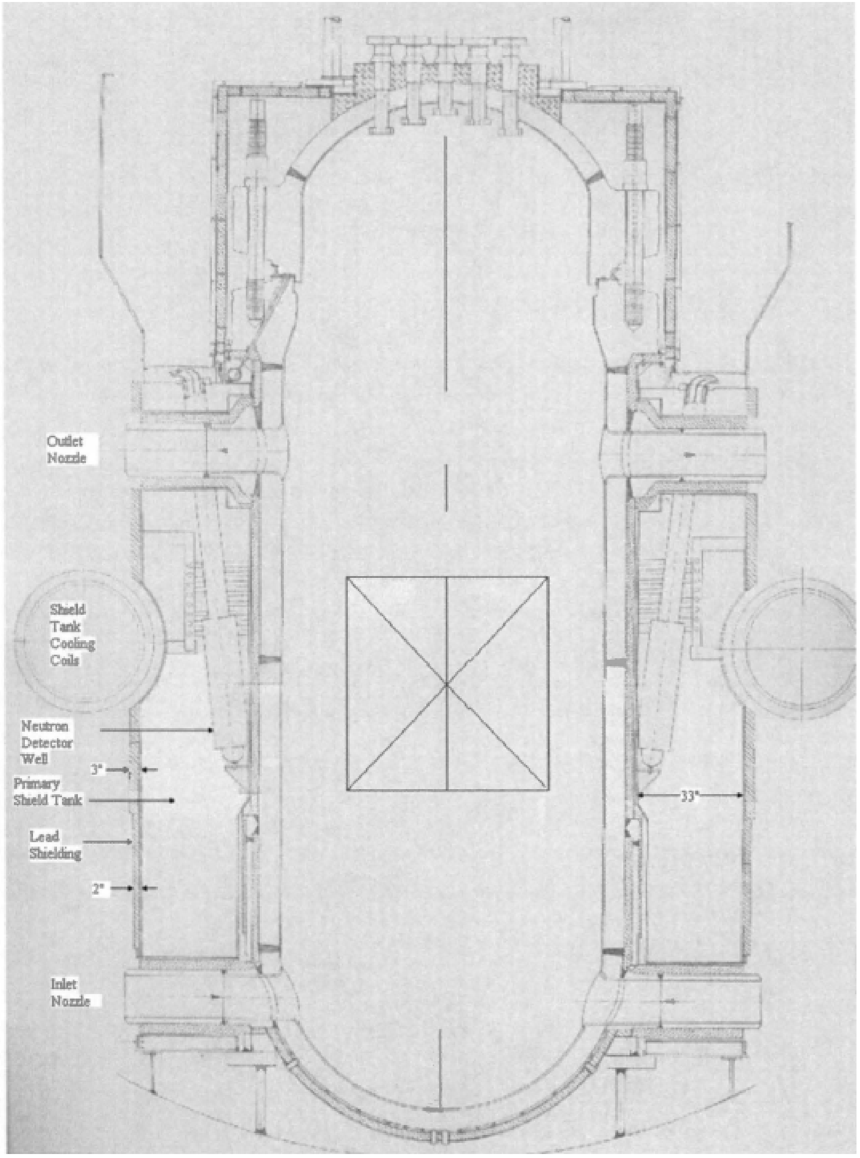


Figure 2.6. Reactor and Cutaway Illustration of Neutron Shield Tank

The outside diameter measures 4.7 meters. The purpose of the NST/FTT was to prevent excessive neutron activation of material inside the containment vessel and to reduce the neutron doses outside the secondary shield. The FTT is similar in dimensions to the NST.

3. DATA QUALITY OBJECTIVES

The data quality objectives (DQOs) described herein are consistent with the *Guidance on Systematic Planning Using the Data Quality Objectives Process* (EPA 2006) and provided a formalized method for planning radiation surveys, improved survey efficiency and effectiveness, and were developed to ensure that the type, quality, and quantity of data collected were adequate for the intended decision applications. The seven steps in the DQO process used in planning the NSS confirmatory survey were as follows:

1. State the problem
2. Identify the decision/objective
3. Identify inputs to the decision/objective
4. Define the study boundaries
5. Develop a decision rule
6. Specify limits on decision errors
7. Optimize the design for obtaining data

3.1 STATE THE PROBLEM

The first step in the DQO process defines the problem that necessitated the study, identifies the planning team, and examines the project budget and schedule.

MARAD committed in the LTP to perform FSS activities in all NSS areas where contamination existed in the past, or where there is presently, or at one time, the potential for contamination. The FSS, consisting of surface scans, surface activity measurements, and volumetric sampling, was to be conducted in accordance with the *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (NRC 2000). The FSS data generated were intended to demonstrate that the NSS satisfies radiological derived concentration guideline levels (DCGLs) and may, therefore, be released without radiological restrictions.

The objectives of the confirmatory survey activities were to provide independent data for NRC's evaluation of the licensee's FSS results. Confirmatory data collection was also required to evaluate whether the efficiencies developed for the FSS surface activity measurement instrumentation were appropriate and would account for volumetric activity resulting from neutron activation within the metal matrices. Based on this, the problem statement was:

Independent confirmatory survey activities are necessary to generate independent radiological data for NRC's use in assessing and evaluating the adequacy and accuracy of the FSS results and that the results account for surficial and potential volumetric residual activity to ensure compliance with the release criteria.

3.2 IDENTIFY THE DECISION/OBJECTIVE

The second step in the DQO process identifies the principal study questions (PSQs) and alternative actions, develops decision statements, and organizes multiple decisions, as appropriate. This second step was done by specifying alternative actions that could result from a "yes" response to the PSQs and combining the PSQs and alternative actions into decision statements. Given that the problem statement introduced in Section 3.1 is fairly broad, multiple PSQs arose. PSQs, alternative actions, and combined decision statements are presented in Table 3.1.

Table 3.1. Confirmatory Survey Decision Process

Principal Study Questions	Alternative Actions
PSQ1: Are split sample analytical results comparable within the statistical uncertainties of the results?	**Yes:** Compile a table of radionuclide-specific confirmatory and licensee data for each sample with results of the comparative assessment and provide to the NRC indicating that results agree within the statistical uncertainties of the procedure and/or where both results were non-detects. **No:** Compile a table of radionuclide-specific confirmatory and licensee data with results of the comparative assessment and provide to the NRC indicating which radionuclide results were outside of agreement parameters if both results were detectable and/or where detection discrepancies occur.
PSQ2: Has the licensee accurately or otherwise conservatively represented the radionuclides of concern (ROCs) and the fractional abundances used to calculate total activity results via either gross activity or surrogate methods?	**Yes:** Compile radionuclide-specific analytical results for each sample, indicate detectable and non-detectable results, calculate fractional estimates, compare with licensee's LTP assumptions, and provide to the NRC for their evaluation and determination that the licensee adequately represented the radionuclide mixture and relationship used in calculating FSS data. **No:** Compile radionuclide-specific analytical results in comparison to the licensee's assumptions and indicate

Table 3.1. Confirmatory Survey Decision Process	
Principal Study Questions	**Alternative Actions**
	parameters that are not in agreement between confirmatory results and licensee results, provide to the NRC for their determination of any non-conservative impact to reported FSS results and guideline compliance.
PSQ3: Are confirmatory survey results comparable with the FSS data for the areas investigated, and are confirmatory survey results below applicable unrestricted release limits?	**Yes:** Compile confirmatory data and report results to the NRC for their decision-making. Provide interpretation of confirmatory field surveys and verify that: 1) surveys did not identify anomalous areas of residual radioactivity, and 2) quantitative field and analytical data were less than the applicable unrestricted release limits. **No:** Compile confirmatory data and report results to the NRC for their decision-making. Provide interpretation of confirmatory field surveys and verify that: 1) confirmatory surveys identified anomalous areas of residual radioactivity, and 2) quantitative field and analytical data exceeded the unrestricted release criteria.
Decision Statements	
The confirmatory survey data agree/do not agree with the licensee's assumed inventory of ROCs and/or that fractional amounts used in developing surrogate ratios or gross activity do/do not accurately represent residual activity and that the FSS data are/are not adequate for demonstrating that the residual surface activity levels are below the unrestricted release limits.	

3.3 IDENTIFY INPUTS TO THE DECISION/OBJECTIVE

The third step in the DQO process identifies both the information needed and the sources of this information, determines the basis for action levels, and identifies sampling and analytical methods to meet data requirements. For this effort, information inputs included the following:

- FSS survey packages and final data.
- Licensee's site-specific DCGLs.
- Licensee prior sample results used to identify applicable ROCs, determine insignificant radionuclides (potential dose contributions of <10% of the TEDE), and estimated relative fractions used for detection efficiency determination and gross activity DCGLs.

- ORISE IV surface scans and surface activity data.
- ORISE and licensee sample results; some of which are split samples.

3.3.1 Radionuclides of Concern

NSS structures and systems were potentially impacted by reactor operation. Components or structures exposed to any neutron fluence during operation could have become radiologically contaminated due to neutron activation of the elemental makeup within the volume of the structural system matrix and sediment or sludge buildups within reactor systems. Volumetric contamination should be limited to materials within the containment vessel, and the intent of the previously described NST was to minimize activation beyond the reactor vessel. Impact to other surfaces could have occurred due to spills or leaks of coolant that had been contaminated from any fuel failures, such as the suspected failure noted during a fuel shuffle outage in 1968 or neutron activation of corrosion products within the coolant. MARAD developed a list of suspected ROCs. The anticipated primary ROCs consisted of the activation products tritium (H-3), carbon-14 (C-14), iron-55 (Fe-55), cobalt-60 (Co-60), nickel-63 (Ni-63), and silver-108m (Ag-108m) and fission products: strontium-90 (Sr-90), technetium-99 (Tc-99), and cesium-137 (Cs-137). The list was developed based on characterization campaigns and the collection of smear, paint, metal, concrete cores, residues, and water samples (MARAD 2023). Alpha emitters were not identified apart from americium-241 (Am-241) and plutonium-239/240 (Pu-239/240) detection in a sludge sample from a Makeup Storage Tank sludge. The licensee-determined ROC fractional abundances are provided in Table 3.2.

Table 3.2. Surface Contamination Activity Fractions	
ROC	**Normalized Activity Fractions[a]**
H-3	3.21E-04
C-14	5.61E-03
Fe-55	1.35E-03
Co-60	1.63E-02
Ni-63	9.00E-01
Sr-90	1.54E-06
Tc-99	2.11E-05
Ag-108m	1.38E-04
Cs-137	7.61E-02
Am-241	9.66E-08
Pu-239/240	8.01E-08

[a]Values from NSS LTP Table 6-5

Per NUREG 1757, Vol. 2, the "*NRC provides for an approach to dose assessment that accounts for the site-specific risk-significance of radionuclides and exposure pathways. The NRC staff allows a licensee to identify radionuclides and exposure pathways that may be considered insignificant, based on their contribution to risk, and remove them from further consideration*" (NRC 2022). Insignificant ROCs are those that collectively would contribute less than 10% of the potential total future dose. The licensee has since submitted revised information, from that included in the LTP, to the NRC regarding the significant ROCs and their relative fractions. The licensee determined that, of the ROCs provided in Table 3.2, only Co-60, Ni-63, and Cs-137 are significant. The licensee opted to retain Ni-63 as an ROC based on the radionuclide's significance to the total residual activity fraction, while from a potential future dose perspective, Ni-63 could have been considered insignificant in accordance with the NRC guidance above. Table 3.3 lists the relative abundances of the ROCs that were accounted for during the FSS. ORISE, therefore, also used the Table 3.3 activity fractions when determining the verification survey detector efficiencies used for calculating surface activity levels and confirming the gross activity DCGL for data comparisons that are described in later sections of this report.

Table 3.3. Significant ROC Surface Contamination Activity Fractions	
ROC	**Normalized Activity Fractions[a]**
Co-60	1.63E-02
Ni-63	9.00E-01
Cs-137	7.61E-02

[a]Values from NSS to NRC correspondence

3.3.2 Residual Radioactivity Derived Concentration Guideline Levels

MARAD developed site-specific DCGLs for each of the ROCs. Although the NRC's license termination criteria are based on 25 mrem/year, the DCGLs were reduced to the operational DCGLs, provided in Table 3.4, that were developed to correspond with a 15 mrem/year maximum dose (MARAD 2023).

Table 3.4. Surface Contamination DCGLs	
ROC	**DCGL (dpm/100 cm^2)[a]**
H-3	6.43E+07
C-14	9.47E+07
Fe-55	4.60E+08
Co-60	2.37E+04
Ni-63	2.53E+08
Sr-90	1.19E+06
Tc-99	6.43E+05
Ag-108m	4.29E+04
Cs-137	1.20E+05
Am-241	4.19E+03
Pu-239/240	4.34E+03

[a]Values from NSS LTP Table 6-17

Note: Per the LTP, 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. The revised information on the licensee's determination of insignificant vs. significant ROCs reduced Table 3.4 to only the applicable DCGLs for Co-60, Ni-63, and Cs-137. The NRC allowance for eliminating insignificant ROCs does not preclude the licensee from being required to account for potential dose from the insignificant ROCs. Generally, this is accomplished by reducing the significant ROC DCGLs by a factor of 10% or 2.5 mrem/yr for the 25 mrem/yr basic dose limit. Because the applicable DCGLs are based on a 15 mrem/yr limit, the licensee prospectively met this condition for demonstrating compliance with the NRC dose limit. Table 3.5 provides the reduced DCGL list, showing only significant ROCs.

Table 3.5. Significant ROC Surface Contamination DCGLs	
ROC	**DCGL (dpm/100 cm^2)[a]**
Co-60	2.37E+04
Ni-63	2.53E+08
Cs-137	1.20E+05

[a]Values from NSS LTP Table 6-17

3.4 DEFINE THE STUDY BOUNDARIES

The fourth step in the DQO process defines target populations and spatial boundaries, determines the timeframe for collecting data and making decisions, addresses practical constraints, and determines the smallest subpopulations, area, volume, and time for which separate decisions must be made.

NRC specified targeted investigation boundaries in the Request for Technical Assistance (NRC 2024). These areas and systems and the site-assigned survey unit number are provided in Table 3.6:

Table 3.6. Targeted Confirmatory Survey Areas			
System (SYS) Survey Unit ID	**Description**		**Radiological Classification**
SYS-116	Retained Portions of Steam Generators		Impacted
SYS-117	Neutron Shield Tank (NST)/Fuel Transfer Tank (FTT)		Impacted
Structure (STR) Survey Unit ID	**Description**	**Ship Deck**	**MARSSIM-Based Classification**
STR-101	Containment Vessel (CV)-1st Level	Tank Top	1
STR-103	Containment Vessel-3rd Level	D-Deck	1

Temporal and/or access restriction boundaries were also factors as to which of these areas, or portions thereof, had confirmatory data collected. The temporal boundary was the scheduled period budgeted for the onsite visit. Confirmatory survey boundaries were defined with discussion and concurrence with NRC once onboard the NSS and accessibility restrictions were fully assessed.

3.5 DEVELOP A DECISION RULE

The fifth step in the DQO process specifies appropriate population parameters (e.g., mean, median), develops action levels, assesses whether detection limits were less than action levels, and develops "if…then…" decision rule statements. The survey consisted of judgmental area selection and judgmental investigations. Therefore, statistically based estimations or decisions are not generally relevant except for split sample analytical results that are the subject of PSQ1.

3.5.1 PSQ1: Split Sample Results Comparison

The response to PSQ1 was to be determined based on agreement/disagreement between split sample results using the duplicate error ratio (DER) method, or, if the licensee does not report measurement uncertainty with their analytical results, the relative percent difference (RPD) method will be used to assess data. The DER was intended to examine the results together with the combined uncertainty using the following equation.

$$\text{DER} = \frac{|C_S + M_S|}{\sqrt{\mu^2(C_S) + \mu^2(M_S)}}$$

However, samples with ROC activity concentrations that are below analytical detection limits negate the usefulness or need of the DER examinations as DCGLs are magnitudes higher than anticipated laboratory detection limits and as such, results below detection limits clearly demonstrate that the DCGLs have been satisfied. The NRC used preliminary screening analytical results discussed in Section 6 to decide that further analyses and DER comparisons were not necessary.

3.5.2 PSQ2: ROC Detection in Samples

The NSS confirmatory PSQ2 decisions are generally more qualitative in nature, consisting of whether an ROC is/is not detected in samples that the licensee included in their assessments and that the observed ratios between ROCs are/are not consistent with the licensee's determination. This decision then is interrelated with decisions regarding PSQ3 and then used to determine whether the ratios between ROCs that are observed in confirmatory samples would result in less conservative surface activity calculation data relative to the DCGLs.

The PSQ2 decision rule is:

Higher ratios of hard-to-detect (HTD) radionuclides than those the licensee established provide evidence that surface activity levels may have been under-reported and, therefore, could lead to an incorrect conclusion that DCGLs have been satisfied when they have not been. Alternatively, the confirmatory sample analytical result may provide evidence that the licensee's ratios are conservative and tend towards higher calculations of residual surface activity yet are less than the DCGLs.

Similarly, as with PSQ1, NRC used screening analyses of samples to decide that further radionuclide-specific analyses to address PSQ2 formally were unnecessary to assess whether the licensee's assumed ratios were conservative.

3.5.3 PSQ3: Comparison of Surface Activity Levels with DCGLs

PSQ3 is a simple binary decision rule. Confirmatory surface activity levels from judgmental measurement locations are compared to gross surface activity DCGLs, calculated based on the ROC ratios in accordance with MARSSIM Equation 4-4, reproduced below, and are determined to be either less than or exceed the gross activity DCGL.

$$DCGL_{\text{Gross Activity}} = \frac{1}{\left(\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \cdots \frac{f_n}{DCGL_n}\right)}$$

Where:

$f_{1,\ 2,\ etc.}$ = fractions of total activity contributed by radionuclide 1, 2, etc. as per Tables 3.3

$DCGL_{1,\ 2,\ etc.}$ = the individual DCGLs for radionuclide 1, 2, etc. Table 3.5 would provide the applicable DCGLs as inputs to the equation.

Alternatively, methods were available had the confirmatory data indicated that surrogate DCGLs would be more advantageous for quantifying surface activity levels in combination with a gross activity DCGL. In that event, Equation 4-1 in MARSSIM could be applied to modify the detectable ROC DCGLs to account for HTDs.

$$DCGL_{\text{Adj Surr}} = \frac{1}{\left(\frac{1}{DCGL_{SURR}} + \frac{R_2}{DCGL_2} + \cdots \frac{R_n}{DCGL_n}\right)}$$

Where:

$DCGL_{Adj\ Surr}$ = the adjusted (lowered) DCGL for the surrogate radionuclide from Table 3.5

$R_{2,\ldots n}$ = are the expected ratios of the activities of the HTD radionuclides to the activity of the surrogate radionuclide of total activity contributed by radionuclide 1, 2, etc. The Table 3.3 relative fractions provided the inputs for ratios.

$DCGL_{2\ldots n}$ = the individual DCGLs for radionuclide 2…n as provided in Table 3.5

Ultimately, the licensee elected to use a surrogate approach in combination with a gross activity guideline to demonstrate compliance whereby Co-60 served as the surrogate ROC for Ni-63 and the Co-60 DCGL adjusted (modified) to account for the presence of any Ni-63. A total gross beta activity DCGL was then calculated using the adjusted Co-60 DCGL, the Cs-137 DCGL and their relative fractions to one another.

The PSQ3 decision rule is:

The parameter of interest is whether the confirmatory surface activity levels are less than or exceed the operational DCGL unrestricted release criterion. If confirmatory surface activity results are less than the gross activity/surrogate DCGL determined using ROC-specific operational DCGLs, then the surveyed areas satisfy 15 mrem/year free release criteria, and, by extension, the NRC's 25 mrem/year criteria. Confirmatory surface activity levels that exceed the operational DCGLs may be less than, or could potentially exceed, the NRC's license termination criteria. The Findings and Results section of this report provide NRC the confirmatory data for their assessment and decision on whether the residual surface activity levels satisfy the operational DCGLs and/or DCGLs adjusted to 25 mrem/year values.

3.6 SPECIFY LIMITS ON DECISION ERRORS

The sixth step in the DQO process examines the consequences of making an incorrect decision and establishes bounds on decision errors. Decision errors were controlled during the survey design, on-site field investigations, and during the data assessment.

3.6.1 Scan Surveys

The confirmatory survey ultimately relied on judgmental surface scanning to identify candidate locations for selecting confirmatory surface activity measurement locations and sample locations. The confirmation of ROCs present and the relative ratios between HTDs and detectable ROCs relied upon identifying sample locations that were likely to contain residual contamination where the activity levels were high enough to readily quantify concentrations and that minimized the analytical uncertainty. The preliminary step in sample location selection was the scanning surveys that were performed. There are relevant decision errors associated with radiation scanning. NUREG-1507 (NRC 2020) describes scanning decision errors used in the calculation of the scan minimum detectable concentration (MDC_{SCAN}). These errors are defined as the true positive proportion (correctly concluding that contamination is present) and false positive proportion (incorrectly concluding that contamination is present). The false positive proportion is often harder to assign, though the assigned statistic has a considerable impact on the survey design. That is, if the DCGL is difficult to detect, stakeholders wish to locate as much contamination as possible, and surveyors are instructed to investigate small anomalous responses, then a relatively large false positive proportion is assigned. Alternatively, if the DCGL is easily detected and it is not important for surveyors to pause often, then a relaxed false positive, i.e., fewer occurrences, can be acceptably tolerated. The primary NSS ROCs of Cs-137 and Co-60 are readily detectable via surface scanning, and the DCGLs for both are large, in which case a small false positive proportion in combination with a reasonable true positive proportion was assigned in the MDC_{SCAN} calculation that ensured the detection sensitivities were less than applicable DCGLs.

There were other considerations that the confirmatory survey surface scanning sensitivities considered. Specifically, one of the NSS confirmatory survey objectives is to verify HTD contribution to any residual contamination. The licensee had concluded that the HTD ROCs are insignificant—contributing less than 10% to the total potential residual dose—and, therefore, were not required to be directly accounted for in survey design and data assessments. The potential dose from ROCs that are deselected from further assessment must still be accounted for. The acceptable methods commonly used to ensure the dose criteria are met include a reduction in the DCGLs of the significant ROCs by up to 10%, equivalent to 2.5 mrem/year. An alternative to percentage reduction of the DCGL is to account for the HTDs being present using either surrogate DCGLs or gross activity DCGLs combined with appropriate weighted detection efficiencies. The licensee had satisfied the requirement because, as noted previously, the DCGLs had already been reduced by more than 10% to equate to 15 mrem/year versus the NRC 25 mrem/year limit. Furthermore, the LTP committed that where HTDs were likely, a representative number of samples were to be collected. Should those sample results show that HTDs in a given area were significant—where significance was defined as HTDs present at greater than 5% of the activity mixture and could contribute greater than 10% of the dose—then a surrogate DCGL was required to be determined. In areas where a surrogate DCGL were to be required, thereby reducing either the Co-60 and/or Cs-137 DCGLs, then the confirmatory MDC_{SCAN}s would need adjustments to reduce the MDC_{SCAN} accordingly. As an example, the NRC had informed ORISE that the licensee had established a Ni-63 to Co-60 ratio of 55.3:1 and used this ratio in MARSSIM Eq. 4-1 to modify (lower) the corresponding DCGL for Co-60 from 2.37E+04 to 2.36E+04 dpm/100 cm^2. Further surface scanning controls accounted for in the confirmatory survey planning accounted for the potential that higher HTD concentrations could exist, with corresponding reduced DCGLs as described above. As such, the MDC_{SCAN}s were adjusted accordingly by accepting larger false positive proportions to enhance the probability of identifying locations where HTD ROCs could be present and, if so, could contribute higher fractions to the total activity relative to the primary ROCs. Furthermore, the types of scans and investigation levels planned were also intended to maximize the probabilities for collecting representative samples. Specifically, this condition relates to assurance that locations where these potential differences in ROC ratios are identified. That was achieved through conducting both gamma and alpha-plus-beta scans. Locations where the primary dose-producing ROCs of Cs-137 and Co-60 dominate were expected to exhibit elevated gamma radiation levels and, if surficial contamination, beta radiation as well. Areas where the primary ROCs were less dominant would be more likely to exhibit elevated beta radiation with lesser gamma levels. Elevated levels of gamma radiation without corresponding elevated beta radiation, if identified during surface scans, was intended to serve as an indicator that contamination, if present, was volumetric rather than surface deposited.

For the NSS confirmatory survey, it was assumed that the NRC desired a high confidence that scanning locations identified for measurements/sampling represented the contamination profile(s). Both the radiation levels and radiation emission ratios were factors in selecting candidate anomalies for sampling such that samples collected would reduce decision errors in both assessments of volumetric versus surficial contamination and ratios of ROCs. Therefore, the confirmation scanning assessments for both gamma and beta scans were based on a high true positive proportion of 0.95, or 5% possibility of overlooking a small area of elevated activity that may provide such data. The design was also based on false positive proportion of 0.50, or 50% possibility that the surveyor would investigate a potential anomaly when there was no indication of elevated radiation due to the primary ROCs, yet HTDs concentrations may be elevated beyond the MARAD-determined ratios.

3.6.2 Split Samples

Error control on deciding whether the results agree between confirmatory and licensee analysis results of split samples was presented within Section 3.5.1. With the DER decision level of 3, there is less than a 0.3% chance of incorrectly concluding that results do not agree, a false negative error, and conversely, a 99.7% probability of correctly concluding the results agree. In the RFTA, NRC requested a minimum of 10 split samples. To evaluate the potential for heterogeneous contamination with split sampling, each sampling location was scanned with portable beta and gamma instrumentation to minimize the potential that a non-uniform location would be sampled.

3.6.3 Minimum Detectable Concentration

Field scanning and analytical minimum detectable concentration (MDCs) were minimized by following the procedures referenced in Sections 4 and 5, respectively. Any anomalies above background identified while performing the surveys or subsequent data assessments were investigated thoroughly and discussed with NRC staff prior to selecting locations for sampling.

3.7 Optimize the Design for Obtaining Data

The seventh step in the DQO process is to review DQO outputs, develop data collection design alternatives, formulate mathematical expressions for each design, select the sample size to satisfy DQOs, decide on the most resource-effective design of agreed alternatives, and document requisite details. Specific survey procedures are presented in Section 4.

4. PROCEDURES

The ORISE survey team performed visual inspections, measurements, and sampling activities within the accessible portions of the targeted SUs identified in Table 3.3, and areas specifically directed by NRC staff. Survey activities were conducted in accordance with the project-specific IV survey plan, the *Oak Ridge Associated Universities (ORAU) Radiological and Environmental Survey Procedures Manual,* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORISE 2024, ORAU 2016, ORAU 2024). Appendices C and D provide additional information regarding survey instrumentation and related processes discussed within this section.

4.1 Reference System

ORISE referenced confirmatory measurement/sampling locations to the MARAD reference system that included component name, deck level, forward/aft section, and port/starboard of center line or other prominent ship features, as appropriate. Measurement and sampling locations were documented on field forms, survey maps, and photographs.

4.2 Surface Scans

Medium to high density (approximately 50 to 100% scan coverage of accessible surfaces) gamma and alpha-plus-beta radiation surface scans were performed over surfaces of the targeted survey units. Access was based on the ability of surveyors to reach surfaces from deck level either directly or with detectors placed on extension poles and also based on access limitations where fall protection was required. Ludlum Model 44-10 sodium iodide-thallium activated [NaI(Tl)] detectors were used for gamma radiation, and 43-68 gas-flow proportional hand-held detectors were used for the alpha-plus-beta scans. Although Co-60 and Cs-137 had been identified as the primary ROCs, the gas proportional detectors were fitted with 0.4 mg/cm^2-thick Mylar windows to enhance detection of low-energy beta emitters, such as C-14 and Ni-63, and were operated in alpha-plus-beta high voltage mode. A Geiger-Muller (GM) detector (e.g., Eberline Model HP-260) was used to survey disassembled reactor components that were too small for the larger gas-flow proportional detectors. All detectors were coupled to Ludlum Model 2221 ratemeter-scalers with audible indicators. Locations of elevated direct radiation, suggesting the presence of residual contamination, were marked for further investigation that was to include direct measurement and potential sampling.

Alpha-only radiation scan capability was available; however, these scans were not performed based on findings as the survey progressed. Alpha-only direct measurements were performed, as described in Section 4.4, to document the undistinguishable from background of alpha surface activity levels.

4.3 SAMPLING/MEASUREMENT LOCATIONS

Confirmatory direct measurements of gamma, alpha-plus-beta, and alpha-only radiation surface count rates were made at 30 judgmental locations. Measurement locations were selected based on both the gamma and alpha-plus-beta surface scans results. These direct measurements served a dual purpose: to both quantify surface alpha-plus-beta and alpha-only surface activity levels and where both the alpha-plus-beta and gamma radiation components could be factored into selecting the intrusive sampling locations discussed in Section 4.5. Additionally, because of variations in ambient gamma radiation background that could influence determination of the relative ratios of observed alpha-plus-beta and gamma radiation of the measured surfaces that were used to select sampling locations, ambient in air background measurements were also made in confirmatory survey areas.

Table 4.1 provides the number of measurements made within the targeted survey areas. Measurements were not made on the steam generator components—that had been temporarily removed and retained as free release items to be used in future ship preservation work—due to the radiation level interference from co-located radiological waste storage.

Table 4.1. Direct Measurement Distribution

Description	Direct Measurement #
FTT, Pressurizer, and Port Steam Generator	5
Neutron Shield Tank Interior and Exterior	11
Containment Vessel-1st Level	14
Containment Vessel-3rd Level	--

Figures A.1 through A.9 are photographs showing the direct measurement locations and gross radiation count rates at each measurement location. The direct alpha-plus-beta and alpha-only static measurements were made using the Ludlum 43-68 gas-flow proportional detectors. Static gamma measurements were made using the Ludlum 44-10 NaI(Tl) detectors.

Static measurement count rates were evaluated based on total alpha-plus-beta counts and total gamma counts and examined for relative ratios between alpha-plus-beta and gamma radiations. Counts were adjusted for ambient in-air background radiation levels if the background count rates interfered between measurement locations with surface activity count ranking accuracy. The variability between the relative abundances of gamma count rates and alpha-plus-beta count rates provided the input for selecting which measurement locations would be further investigated by sampling. That is, locations provided a mix of where the observed gamma count rates were the primary elevated radiation in relation to alpha-plus-beta levels and vice versa.

A set of four wet smears, one smear for assessing removable levels for each of the HTD ROCs C-14, H-3, Fe-55, and Ni-63, were collected from 10 of the direct measurement locations.

4.4 METAL SAMPLING

The NRC had requested that split volumetric metal samples be collected for confirmatory and licensee analyses as the means to address PSQ1. The direct measurements rankings and ratios together with discussions with the NRC representatives were used to identify a population of 10 of the confirmatory direct measurement locations for this volumetric sampling. Due to temporal boundaries, ORISE was unable to collect the samples during the time scheduled on-site for the confirmatory survey. The licensee had also committed to collecting metal samples, had predetermined their locations, but had not yet collected the samples. The NRC requested that the licensee while sampling, also collect the confirmatory samples. The licensee agreed to do so with NRC's concurrence that the licensee could relocate an equivalent number of the predetermined locations to the 10 confirmatory survey-selected metal sample locations.

The sample matrix was metal shavings that the licensee collected using drills with metal drill bits. The PSP specified that in addition to the pre-sampling direct measurements that ORISE did collect—the data ORISE used to select the locations for sampling—post-sampling direct measurements were planned to assess whether count rate reductions occurred which could have indicated that elevated direct radiation was surficial versus volumetric from neutron activation. The surface area that was sampled was also to be recorded to assess any difference in the activity, in units of disintegrations per minute per square centimeter (dpm/cm^2) of sampled area, between the before and after sampling direct measurement. However, because ORISE was not able to collect the samples, these additional post-sampling data were not collected. Figures A.1 through A.9 include notations, as applicable regarding sample identification number, where the licensee was to collect the metal shavings samples.

5. SAMPLE ANALYSIS AND DATA INTERPRETATION

Data collected on site were transferred to the ORISE facility for analysis and interpretation. The licensee shipped confirmatory metal shavings samples under chain-of-custody to the Radiological and Environmental Sciences Laboratory (RESL) in Idaho Falls, Idaho. Sample analyses were performed in accordance with the laboratory's applicable procedures or other special instructions from NRC.

An aliquot of each of the metal shavings samples was analyzed via liquid scintillation counting (LSC) for gross activity, and the results reported as counts per minute (cpm) for the beta, alpha, and gamma energy regions of interest. NRC elected not to perform additional radionuclide-specific analyses of the metal shavings based on these preliminary sample screening results. Similarly, one smear from each set of four wet smears was screened via LSC, and the gross counts per region were reported. NRC selected a subset of the wet smears for further quantitative analyses and the results were provided to ORISE.

The gross alpha-plus-beta and gross alpha-only surface activity measurement count rates were converted to units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2) using appropriate efficiency and geometry correction factors. Efficiencies used were weighted efficiencies based on the significant ROC mixture fractions discussed in Section 3 as NRC elected to forego radionuclide-specific analyses of the confirmatory samples upon which a different fractional efficiency calculation could have been made. Because the DCGL levels are readily distinguishable from background, gross measurements were not corrected for background radiation.

6. FINDINGS AND RESULTS

The results of the confirmatory survey activities are discussed in the following subsections.

6.1 SURFACE SCANS

Surface scans did identify generally small—all less than one square meter in size—anomalous locations of elevated gamma and/or alpha-plus-beta radiation levels on all targeted structures and systems. Table 6.1 provides the observed scan ranges for the targeted confirmatory survey areas together with ambient, in-air observed background radiation levels.

Table 6.1. Surface Scan Ranges				
Targeted Area	Ambient Alpha-Plus-Beta Background	Alpha-Plus-Beta	Ambient Gamma Background	Gamma
	cpm			
CV- Level 1	160	100 to 3,200	800	400 to 8,000
NST- Level 1	130	100 to 1,200	600	300 to 4,100
CV-Level 3	130	150 to 1,200	1,000	500 to 1,300
NST and FTT-Level 3	--	200 to 7,000	900	600 to 2,300
Starboard Steam Generator	--	250 to 1,800	--	1,200 to 1,800

The anomalies with the maximum observed count rates, together with representative locations of lesser anomalies, were selected for direct alpha, alpha-plus-beta, and gamma measurements.

6.2 SURFACE ACTIVITY AND REMOVABLE GROSS ACTIVITY LEVELS

Total surface activity levels at each direct measurement location are provided in Table B.1. The minimum and maximum for each radiation type are summarized in Table 6.2.

Table 6.2. Surface Activity Summary Results					
Ranges	Gamma	Alpha Activity		Alpha-Plus-Beta Activity	
	cpm	cpm	dpm/100 cm^2	cpm	dpm/100 cm^2
Minimum	1,100	0	0	165	600
Maximum	7,800	6	43	6407	23,000

A gross beta activity DCGL was calculated using the individual significant ROC DCGLs from Table 3.5, the ROC fractions listed in Table 3.3 and further modified to correspond to Co-60 and Cs-137 fractions relative to their combined activity, both shown in Table 6.3—and the MARSSIM-based modified and gross activity DCGL equations given in Section 3.5.3.

Table 6.3. Gross Beta DCGL		
ROC	DCGL (dpm/100 cm^2)	Activity Fraction
Co-60 modified	2.36E+04	1.63E-02
Cs-137	1.20E+05	7.61E-02
Gross Beta Activity DCGL = 69,700 dpm/100 cm^2		

The maximum measured confirmatory alpha-plus-beta surface activity level was 23,000 dpm/100 cm^2, or 33% of the gross beta activity DCGL. The decision statement for PSQ3 is that if confirmatory surface activity results are less than the gross activity/surrogate DCGL determined using ROC-specific operational DCGLs, then the surveyed areas satisfy 15 mrem/year free release criteria.

Table 6.4 provides the summary range of gross activity for the LSC smear analysis and includes the blank smear gross counts results for comparison for qualitative assessment for the presence of removable contamination, with emphasis on whether there was evidence that HTD ROCs were present at concentrations much higher, and therefore less conservative, than the ratios the licensee determined through characterization.

Table 6.4. Liquid Scintillation Counts Summary for Wet Smear Samples

	Counts per Minute (cpm) per Counting Window/Energy (keV) Region			
	Beta (0-20)	Beta (0-2000)	Alpha (0-2000)	Gamma (0-2000)
Confirmatory Smears	6.0 to 102	6.7 to 103	4.7 to 16	190 to 240
LSC Smear Blank	3.5	4.4	9.5	250

Table B.2 includes the individual LSC results for the 10 smears analyzed. The results showed that beta activity was detectable above the blank counts but were a small fraction of the total surface activity measured at the respective locations. Nine out of 10 of the alpha results were less than the blank counts, and all gamma counts were below those of the blank. NRC selected a subset of the wet smears for further quantitative ROC-specific analyses. Table B.3 includes only the ROC-specific analytical results that are considered statistically positive for the smears that were analyzed. There were no ROC-specific removable activity results above the maximum recommended removable contamination levels of 10% of the total levels. Thus, the decision statement for PSQ2 is that the confirmatory sample analytical results did provide evidence that the licensee's ratios are conservative and tend towards higher calculations of residual surface activity.

6.3 RADIONUCLIDE CONCENTRATIONS IN METAL SAMPLES

Table 6.5 provides the summary gross activity LSC analytical cpm of the metal shaving samples and blanks and Table B.4 provides the individual sample results.

Table 6.5. Liquid Scintillation Counts Summary for Metal Shaving Samples

Sample	Counts per Minute (cpm) per Counting Window/Energy (keV) Region			
	Beta (0-50)	Beta (50-2000)	Alpha (0-2000)	Gamma (0-2000)
Metal Shavings	3.5 to 5.2	0.25 to 0.55	7.6 to 9.8	250 to 260
LSC Cocktail Blank	3.4	0.45	10.6	280
Boiling Chip Blank	3.3	0.42	10.9	270

These gross activity results indicated that there was minimal volumetric contamination, and therefore, NRC also determined that ROC-specific analyses were unnecessary for the metal shavings. In context of PSQ2, as with the smear samples the results do not contradict the licensee's characterization information and therefore the FSS results on residual ROC mixtures and distributions in the structures and systems.

7. SUMMARY AND CONCLUSIONS

Confirmatory survey activities on the NS Savannah were conducted during three days: July 29 through August 1, 2024. The focus of these confirmatory survey activities was to collect independent radiological scan, total surface activity, removable HTD activity, and volumetric sample data that would enable NRC to assess whether the licensee's final status survey data were adequate and representative of final radiological conditions as compared with the 15 mrem/year based operational DCGLs. Provided results demonstrated that the operational DCGLs were satisfied, then the NRC's 25 mrem/year would also be met, which would also account for any dose contribution from the insignificant ROCs, relative to their potential contribution to the 25 mrem/yr NRC limit. The confirmatory survey targeted remaining ship structural and system survey areas that were either closely associated with the ship's nuclear reactor that provided ship propulsion during the in-service lifetime or could have otherwise been radiologically impacted during decommissioning. Such areas had the greatest potential to have neutron activation product contamination from reactor operation or to have become contaminated from any fission or activation product containing fluid leaks during operation or decommissioning.

Based on the medium to high-density gamma and alpha-plus-beta scans, 30 locations exhibiting elevated radiation levels, as compared with ambient background, were judgmentally selected for direct measurement and quantification of surface activity. The maximum alpha-plus-beta surface activity measurement was 23,000 dpm/100 cm^2. The activity level was calculated with a detector efficiency that was weighted per the licensee's significant ROC mixture and compared to a modified gross activity DCGL of 69,700 dpm/100 cm^2. This DCGL was also calculated and confirmed using the same fractional activity mixture of the licensee's significant radionuclides of Co-60, Ni-63, and Cs-137 whereby Ni-63 activity was accounted for via a surrogate, fractional relationship with Co-60. The maximum surface activity of 23,000 dpm/100 cm^2 is also less than the individual beta-emitting ROC.

The surface activity results were supplemented with the 10 wet smear samples to assess removable ROC, including HTD, levels and 10 metal samples to assess volumetric ROC contamination. The gross activity LSC results for the screening analyses performed on these samples were unremarkable and, therefore, did not indicate excessive activity that could impact assumptions used in assessing residual contamination levels. The lower and higher energy beta channels did have 100 gross counts per minute as compared to the 3.5 counts per minute of the blank sample, while the gamma region counts were less than the blank. This provides evidence that there is not a gamma component to the above background beta activity that was observed in the beta regions. This should not be the case if Co-60 was the primary ROC rather than the 90% fractional amount of Ni-63 given in the LTP and shown in Table 6.3. The metal shaving samples showed little, if any, gross beta activity distinguishable from the laboratory blanks. Neither smear nor metal shavings gross alpha results showed alpha contamination evidence that would be contrary to the licensee's conclusions as to the insignificant activity fraction to the total ROC activity mixture. Also, because the gross LSC activities in samples were minimal, the planned comparative evaluations between confirmatory and licensee analytical results are unnecessary and likely to not have provided viable results, i.e. above laboratory detection limits, upon which to apply a meaningful DER assessment. The subset of wet smears the NRC selected for further quantitative ROC-specific analyses did not contain removable activity above the maximum recommended removable contamination levels of 10% of the total levels.

In summary, the problem statement was that independent confirmatory surveys were necessary to generate radiological data for NRC's use in their evaluation of the adequacy and accuracy of the NS Savannah FSS results. The confirmatory results address surficial and potential volumetric residual activity and the licensee's assumptions for generating and assessing final status survey data. PSQs 1 and 2 could be qualitatively, but not quantitatively answered, indicating that results would likely be non-detects for PSQ 1 and that the licensee adequately represented the radionuclide mixture and relationship used in calculating FSS data for PSQ 2; both based on the screening LSC analyses and limited quantitative ROC-specific analyses from select wet smear samples. The confirmatory surface scanning and surface activity measurement data address PSQ 3 that although elevated residual surface activity anomalies were identified by scans, the measured surface activity levels were less than the gross activity guideline calculated from the licensee's fractional abundances information.

8. REFERENCES

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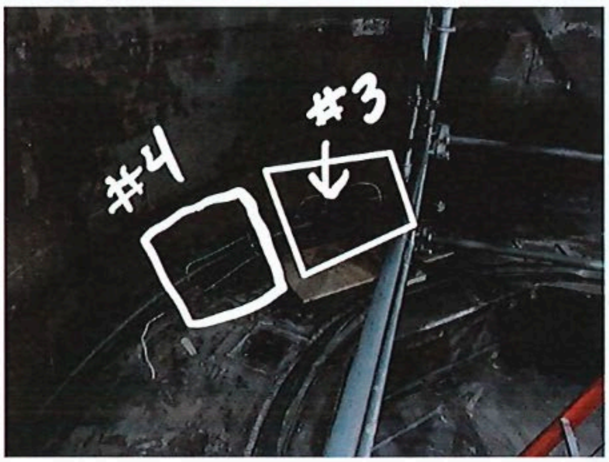
APPENDIX A: FIGURES

AFT locations inside NST. Highest NaI reading along/in lip marked with the arrow.

#1 Gamma = 3301 cpm
#1 Beta = 1105 cpm (detector sitting on lip, some air space)
#1 alpha = 3 cpm

#2 gamma = 1647 cpm
#2 beta = 4180 cpm (on floor just below open hole)
#2 alpha = 0 cpm

ORISE 5381M0002 (shift their #23 from inside the NST)



STBD location inside NST. Highest NaI reading marked with the arrow.

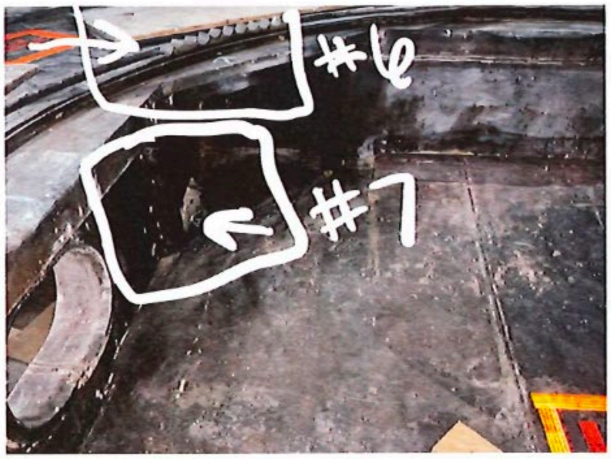
#3 Gamma = 3740 cpm
#3 Beta = 165 cpm
#3 alpha = 2 cpm

ORISE 5381M0001 (shift their #20 from inside the NST)

#4 Gamma = 1629 cpm
#4 Beta = 180 cpm
#4 alpha = 3 cpm

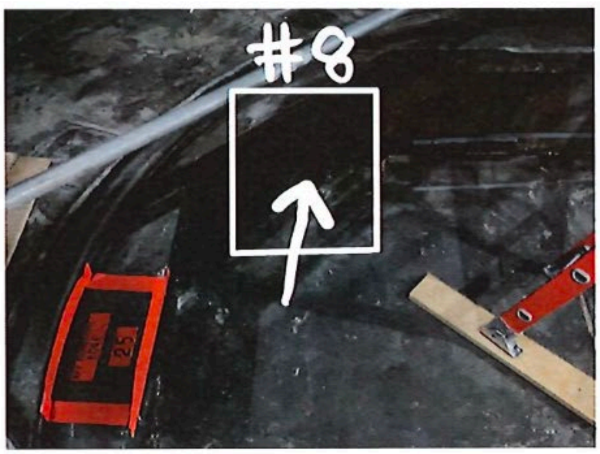
AFT location inside NST. Highest NaI reading marked with the arrow.
#5 Gamma = 1385 cpm
#5 Beta = 686 cpm
#5 alpha = 0 cpm

Figure A.1.



AFT location inside NST. Highest NaI reading marked with the arrow.
#6 Gamma = 1306 cpm
#6 Beta = 1696 cpm
#6 alpha = 0 cpm

#7 Gamma = 1295 cpm
#7 Beta = 1296 cpm
#7 alpha = 3 cpm



FWD location inside NST. Highest NaI reading marked with the arrow.

#8 Gamma = 1809 cpm
#8 Beta = 177 cpm
#8 alpha = 0 cpm

ORISE 5381M0004 (shift their #19 from inside the NST)

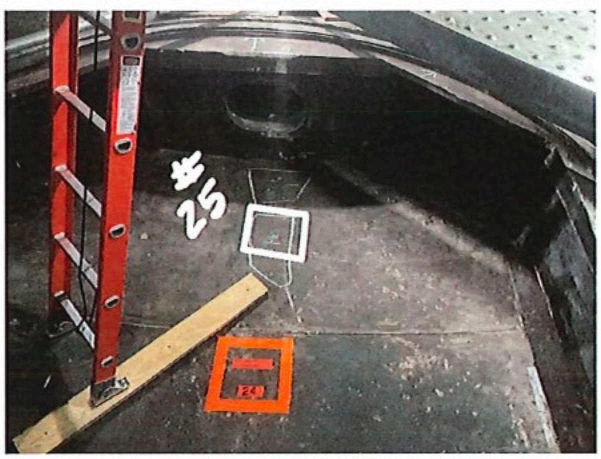
STBD location on exterior of NST. Highest NaI reading is where detector is sitting.

#9 Gamma = 5619 cpm
#9 Beta = 571 cpm
#9 alpha = 1 cpm

Figure A.2.

AFT location on exterior of NST. Highest NaI reading is where detector is sitting.

#10 Gamma = 1620 cpm
#10 Beta = 328 cpm
#10 alpha = 3 cpm

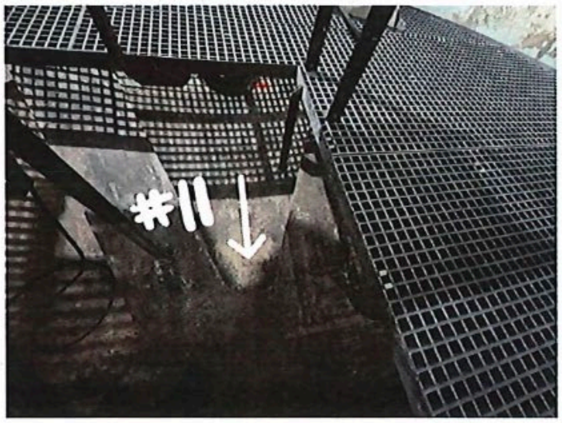


AFT location inside NST on floor. Highest NaI reading marked with the arrow.

#25 Gamma = 1285 cpm
#25 Beta = 1219 cpm
#25 alpha = 1 cpm

ORISE 5381M0003 (shift their #24 from inside the NST)

Figure A.3.

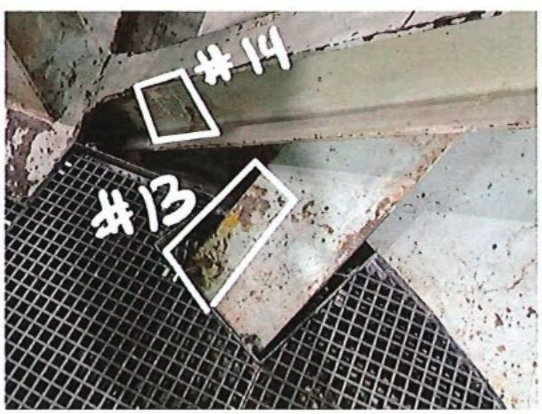


AFT location inside containment, Level 1. Location marked with arrow in photo.

#11 Gamma = 1941 cpm
#11 Beta = 812 cpm
#11 alpha = 4 cpm

AFT, STBD side location inside containment, Level 1. Location marked with box in photo.

#12 Gamma = 7846 cpm
#12 Beta = 2843 cpm
#12 alpha = 1 cpm



AFT, STBD side location inside containment, Level 1. Locations marked with box in photo.

#13 Gamma = 3758 cpm
#13 Beta = 495 cpm
#13 alpha = 3 cpm

#14 Gamma = 3383 cpm
#14 Beta = 3551 cpm
#14 alpha = 3 cpm

Figure A.4.

STBD side location inside containment, Level 1. Location marked with box in photo.

#15 Gamma = 4176 cpm
#15 Beta = 348 cpm
#15 alpha = 0 cpm

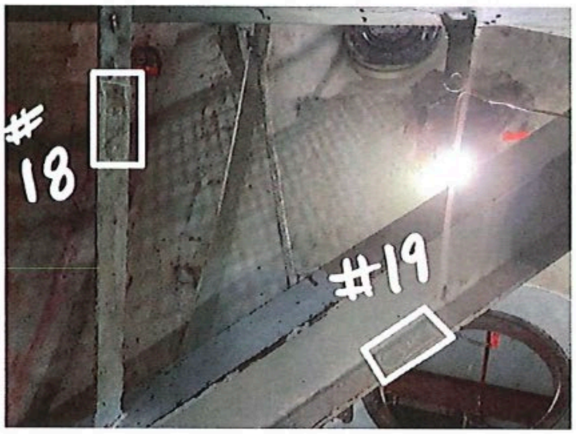
FWD location inside containment, Level 1. Location marked with box in photo. Instrument air lines, penetration 66.

#16 Gamma = 4009 cpm
#16 Beta = 3045 cpm (detector hovering over penetrations, very sharp)
#16 alpha = 2 cpm

AFT, STBD side location inside containment, Level 1. Location marked with box in photo.

#17 Gamma = 1904 cpm
#17 Beta = 2005 cpm
#17 alpha = 1 cpm

Figure A.5.



AFT, STBD side location inside containment, Level 1. Locations marked with box in photo.

#18 Gamma = 1294 cpm
#18 Beta = 1337 cpm
#18 alpha = 6 cpm

#19 Gamma = 1834 cpm
#19 Beta = 2947 cpm
#19 alpha = 6 cpm

AFT location inside containment, Level 1. Location marked with box in photo.

#20 Gamma = 1588 cpm
#20 Beta = 1210 cpm
#20 alpha = 1 cpm

FWD location inside containment, Level 1. Location marked with box in photo.

#21 Gamma = 2824 cpm
#21 Beta = 2503 cpm
#21 alpha = 4 cpm

Figure A.6.

FWD location inside containment, Level 1. Location marked with box in photo.

#22 Gamma = 1055 cpm
#22 Beta = 1618 cpm
#22 alpha = 0 cpm

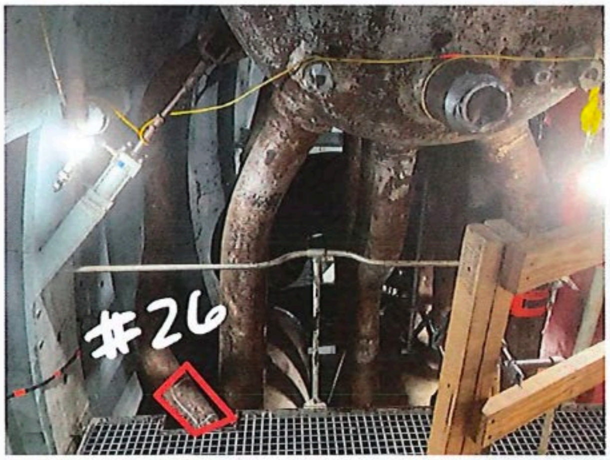
FWD location inside containment, Level 1. Location marked with box in photo.

#23 Gamma = 2218 cpm
#23 Beta = 2144 cpm
#23 alpha = 2 cpm

AFT, port side location on I beam inside containment, Level 1. Location marked with box in photo.

#24 Gamma = 1152 cpm
#24 Beta = 1564 cpm
#24 alpha = 2 cpm

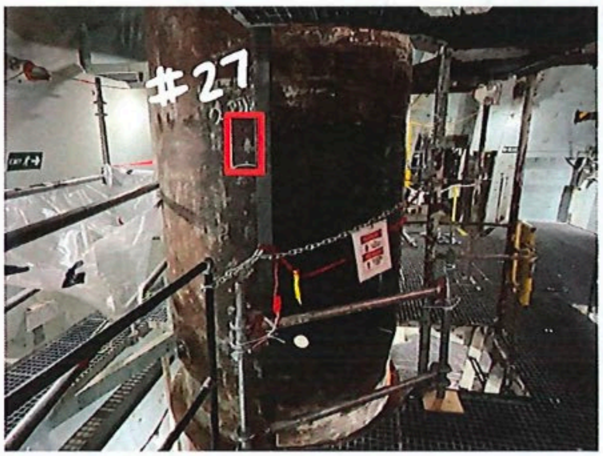
Figure A.7.



Port Steam Generator location on downcomber, port side when looking forward.
Note: the contamination is likely coming from the inside of the tube so the sample needs to go all the way through to include internal surface/ shavings. Site staff indicated they did not want to punch through.

Will be assigned ORISE sample ID 5381M0005 (shift their #4 from the riser steam tube). Marked with red tape.

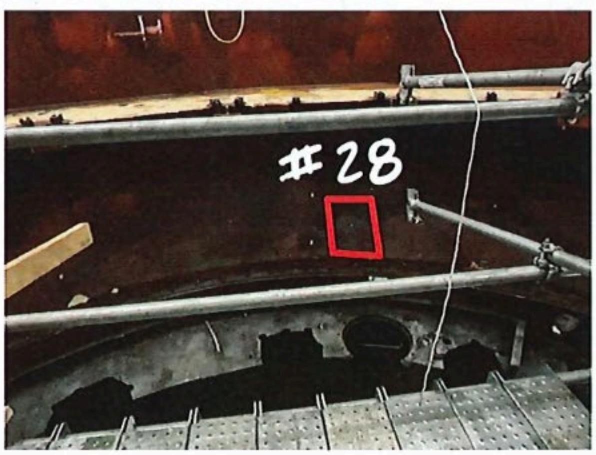
#26 Gamma = 7527 cpm
#26 Beta = 506 cpm
#26 Alpha = 5 cpm



Pressurizer Level 3

Will be assigned ORISE sample ID 5381M0006 (shift their #16 on the pressurizer). Marked with red tape.

#27 Gamma = 1074 cpm
#27 Beta = 1361 cpm
#27 Alpha = 2 cpm

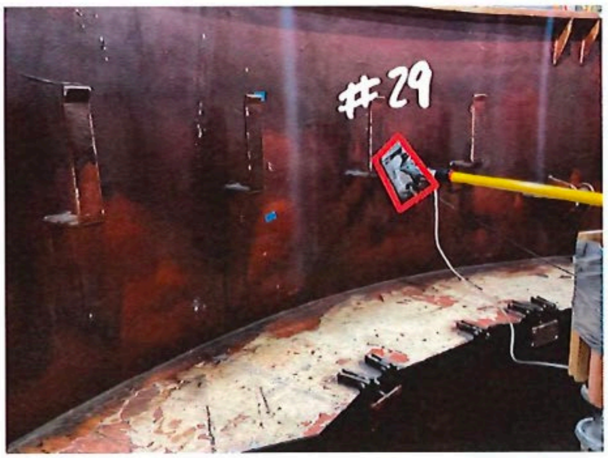


Fuel Transfer Tank Level 3, FWD

Will be assigned ORISE sample ID 5381M0007 (shift their #26 on the FTT).

#28 Gamma = 2238 cpm
#28 Beta = 2615 cpm
#28 Alpha = 2 cpm

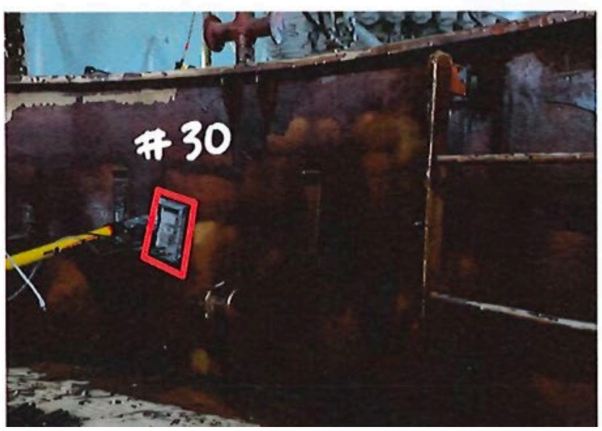
Figure A.8.



Fuel Transfer Tank Level 3, FWD/port side

Will be assigned ORISE sample ID 5381M0008 (shift their #10 on STBD Stem Generator tube section U shape, aft end).

#29 Gamma = 1384 cpm
#29 Beta = 6407 cpm
#29 Alpha = 2 cpm



Fuel Transfer Tank Level 3, aft side

Will be assigned ORISE sample ID 5381M0009 (shift their #7 on the Port Steam Generator Drum closest to core).

#30 Gamma = 1536 cpm
#30 Beta = 5213 cpm
#30 Alpha = 5 cpm

Figure A.9

APPENDIX B: DATA TABLES

Table B.1. Surface Activity Measurements						
Direct Measurement ID	Measurement Location Description	Gamma	Alpha Activity		Alpha-Plus-Beta Activity	
		cpm	cpm	dpm/100 cm²	cpm	dpm/100 cm²
1	AFT Interior NST/Along lip	3,300	3	22	1105	4,000
2	AFT Interior NST	1,600	0	0	4180	15,000
3[a]	**STBD Interior NST**	3,700	2	14	165	600
4	STBD Interior NST	1,600	3	22	180	650
5	**AFT Interior NST**	1,400	0	0	686	2,500
6	**AFT Interior NST**	1,300	0	0	1696	6,100
7	AFT Interior NST	1,300	3	22	1296	4,700
8	FWD Interior NST	1,800	0	0	177	640
9	**STBD Exterior NST**	5,600	1	7	571	2,100
10	AFT Exterior NST	1,600	3	22	328	1,200
11	AFT Interior Containment	1,900	4	29	812	2,900
12	**AFT, STBD Interior Containment**	7,800	1	7	2843	10,000
13	**AFT,STBD Interior Containment**	3,800	3	22	495	1,800
14	AFT,STBD Interior Containment	3,400	3	22	3551	13,000
15	**STBD Interior Containment**	4,200	0	0	348	1,300
16	**FWD Interior Containment/Air Lines**	4,000	2	14	3045	11,000
17	AFT, STBD Interior Containment	1,900	1	7	2005	7,200
18	AFT, STBD Interior Containment	1,300	6	43	1337	4,800
19	AFT, STBD Interior Containment	1,800	6	43	2947	11,000
20	AFT Interior Containment	1,600	1	7	1210	4,400
21	FWD Interior Containment	2,800	4	29	2503	9,000
22	FWD Interior Containment	1,100	0	0	1618	5,800
23	FWD Interior Containment	2,200	2	14	2144	7,700
24	AFT, Port Interior Containment/I beam	1,200	2	14	1564	5,600
25	AFT Interior NST/Floor	1,300	1	7	1219	4,400
26	**Port Steam Generator/Downcomer**[b]	7,500	5	36	506	1,800
27	**Pressurizer Level 3**	1,100	2	14	1361	4,900
28	FWD Fuel Transfer Tank Level 3	2,200	2	14	2615	9,400
29	FWD, Port Fuel Transfer Tank Level 3	1,400	2	14	6407	23,000
30	AFT, Fuel Transfer Tank Level 3	1,500	5	36	5213	19,000

[a]Bold locations have wet smear removable gross activity results, see Table B.3

Table B.2. Liquid Scintillation Counts for Wet Smear Samples				
Direct Measurement ID #/ Smear #	**Counts per Minute (cpm) per Counting Window/Energy (keV) Region**			
	Beta (0-20)	**Beta (0-2000)**	**Alpha (0-2000)**	**Gamma (0-2000)**
12/01	100	100	6.6	230
13/05	20	21	4.7	190
16/09	12	19	16	230
15/13	20	21	6.9	210
9/17	6.0	6.7	4.9	190
5/21	18	20	6.6	200
3/25	8.5	9.3	5.7	210
6/29	14	15	7.1	230
26/33	19	20	7.3	240
27/37	75	77	8.5	240
LSC Cocktail Blank	3.5	4.4	9.5	250

Table B.3. Removable Activity for Wet Smear Samples[a]			
Direct Measurement ID #/ Smear #	**ROC**	**Result and Uncertainty[b] (pCi/100cm^2)[c]**	**Result (dpm/100cm^2)[d]**
12/02	Co-60	26.4 ± 1.6	59
12/02	Cs-137	3.8 ± 0.4	8
12/02	Tc-99	2.9 ± 1.2	6
12/04	Ni-63	2,661 ± 226	5,907
16/10	Cs-137	6.7 ± 1.6	15
16/12	Ni-59	32.8 ± 7.4	73
16/12	Ni-63	82.6 ± 8	183

ROC = radionuclide of concern
[a]Only the analytical results that are considered statistically positive are presented.
[b]Uncertainties are based on total propagated uncertainties at the 95% confidence level. 2 sigma uncertainty is presented.
[c]Analytical results were presented as pCi/smear. The surface area wiped was approximately 100 cm^2, so units have been updated.
[d]pCi results were converted to dpm by multiplying by 2.22.

Table B.4. Liquid Scintillation Counts for Metal Shavings Samples				
Misc. Sample ID/ Direct Measurement ID #	**Counts per Minute (cpm) per Counting Window/Energy (keV) Region**			
	Beta (0-50)	**Beta (50-2000)**	**Alpha (0-2000)**	**Gamma (0-2000)**
5381M0001/3	5.2	0.43	8.3	250
5381M0002/2	3.9	0.32	9.8	260
5381M0003/25	4.0	0.36	9.4	260
5381M0004/8	3.9	0.27	8.9	250
5381M0005[a]/26	4.5	0.32	9.3	260
5381M0006/27	3.5	0.55	7.6	250
5381M0007/28	4.0	0.34	9.4	260
5381M0008/29	3.9	0.42	9.3	260
5381M0009/30	4.8	0.25	8.8	260
5381M0010/ MARAD Location #11	3.9	0.36	9.2	260
LSC Cocktail Blank	3.4	0.45	10.6	280
Boiling Chip Blank	3.3	0.42	10.9	270

[a]The metal sample from the Port Steam Generator Downcomber was collected from the external surface and did not punch through to the interior surface; the elevated radiation levels were believed to be on the interior of the Downcomber.

APPENDIX C: MAJOR INSTRUMENTATION

C.1. SCANNING AND MEASUREMENT INSTRUMENT/ DETECTOR COMBINATIONS

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the author or their employer.

C.1.1 Gamma

Ludlum NaI[Tl] Scintillation Detector Model 44-10, Crystal: 5.1 cm × 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Ludlum Ratemeter-scaler Model 2221

C.1.2 Alpha and Alpha-Plus-Beta

Ludlum Gas Proportional Detector Model 43-68, Physical Probe Area of 126 cm^2
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Ludlum Ratemeter-scaler Model 2221

Ludlum Geiger-Mueller Detector Model 44-9, Physical Probe Area of 20 cm^2
(Ludlum Measurements, Inc., Sweetwater, Texas)
Coupled to: Ludlum Ratemeter-scaler Model 2221

APPENDIX D: SURVEY AND ANALYTICAL PROCEDURES

D.1. PROJECT HEALTH AND SAFETY

The Oak Ridge Institute for Science and Education (ORISE) performed all survey activities in accordance with the Oak Ridge Associated Universities (ORAU) Radiation Protection Manual, the ORAU Radiological and Environmental Survey Procedures Manual, and the ORAU Health and Safety Manual (ORAU 2020b, ORAU 2016, and ORAU 2020). Prior to on-site activities, a Work-Specific Hazard Checklist was completed for the project and discussed with field personnel. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. Should ORISE have identified a hazard not covered in ORAU 2016 or the project's Work-Specific Hazard Checklist for the planned survey and sampling procedures, work would not have been initiated or continued until the hazard was addressed by an appropriate job hazard analysis and hazard controls.

D.2. CALIBRATION AND QUALITY ASSURANCE

Calibration of all field instrumentation was based on standards/sources traceable to the National Institute of Standards and Technology (NIST).

Field survey activities were conducted in accordance with procedures from the following documents:

- ORAU Radiological and Environmental Survey Procedures Manual (ORAU 2016)
- ORAU Environmental Services and Radiation Training Quality Program Manual (ORAU 2024)

The procedures contained in these manuals were developed to meet the requirements of U.S. Department of Energy (DOE) Order 414.1D and U.S. Nuclear Regulatory Commission's (NRC's) *Quality Assurance Manual for the Office of Nuclear Material Safety and Safeguards* and contain measures to assess processes during their performance.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

D.3. SURVEY PROCEDURES

D.3.1 Surface Scans

Scans for elevated gamma and alpha-plus-beta radiation were performed by passing the detectors slowly over the surface. The distance between the detector and surface was maintained at a minimum. The scans were used solely as a qualitative means to identify elevated radiation levels in excess of background based on an increase in the audible signal from the indicting instrument.

Using the scan minimum detectable concentration (MDC) calculation approach outlined in NUREG-1507, the *a priori* confirmatory alpha-plus-beta scan sensitivity was evaluated. The calculation used the following inputs:

- Index of sensitivity = 1.96
- Observation interval = 1 second
- Detector background = 390 counts per minute (cpm)
- Surveyor efficiency = 0.75
- Radionuclide of concern (ROC) fraction = 0.90 of Ni-63, 0.08 Cs-137, and the balance of other beta emitters

Based on the above inputs, the corresponding gross alpha-plus-beta scan MDC was 3,900 dpm/100 cm^2, which is below the gross activity operational DCGL.

Gamma scans were also performed. A quantified *a priori* scan MDC was not considered a critical component of the confirmatory survey planning. For general information, the NaI(Tl-doped) gamma scintillation detectors are sensitive to the primary gamma emitter, Cs-137, assumed in the ROC mixture and were used solely as a qualitative means to identify elevated radiation levels in excess of background. Identification of elevated radiation levels that could exceed the localized background was determined based on an increase in the audible signal from the indicating instrument and then follow up alpha-plus beta direct measurements were performed.

D.3.2 Surface Activity Measurements

Measurements of gross alpha and gross alpha-plus-beta surface activity levels were performed using hand-held gas proportional detectors coupled to portable ratemeter-scalers. Count rates, which were integrated over 1 minute with the detector held in a static position, were converted to activity levels by dividing the count rate by the total static efficiency and correcting for the physical area of the detector plus background. The MDC for alpha-plus-beta static surface activity measurements was calculated using the following equation:

$$MDC = \frac{3 + (4.65\sqrt{B})}{TG\varepsilon_{tot}}$$

Where:

B = background in time interval, T (1 min)

T = count time (min) used for field instruments

ε_{tot} = total efficiency = $\varepsilon_i \times \varepsilon_s$ (instrument efficiency × source efficiency)

G = geometry correction factor (1.26)

The static MDC based on the beta emitters present and relative fractions presented in the LTP is 1,515 dpm/100 cm^2, based on a beta instrument background of 390 cpm.

D.3.3 REMOVABLE ACTIVITY SAMPLING

Removable activity smear samples were collected using numbered filter paper disks. Moderate pressure was applied to the smear and approximately 100 cm^2 of the surface was wiped. Smears for the HTD ROCs of C-14, H-3, Fe-55, and Ni-63, were collected from 10 direct measurement locations. Each smear was first wetted with deionized water before the surface was wiped and then placed in glass vials with deionized water. Locations and other pertinent data were recorded, and all samples were transferred under chain-of-custody to the Radiological and Environmental Sciences Laboratory (RESL) in Idaho Falls, Idaho for analysis.

D.4. RADIOLOGICAL ANALYSIS

D.4.1 GROSS ALPHA AND BETA SCREENING METHOD, LIQUID SCINTILLATION COUNTING

Gross alpha and beta counting is a screening method and therefore the interpretation of the results should only be used to determine if there is activity present above background levels. The samples are mixed with ultima gold LLT liquid scintillation cocktail and counted on a liquid scintillation instrument for 100-min with a matrix-matched blank. There is no established MDA for this procedure, as it is a screen for any nuclide within the sample.

D.4.2 DETECTION LIMITS

Each RESL analytical result is accompanied by its total propagated uncertainty expressed at one standard deviation. All results for which the 95% confidence interval does not contain zero are considered statistically positive (i.e. a 'detect') for the given analyte. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differed from sample to sample and instrument to instrument.