# APPENDIX P
# SERAT REPORT

## Radiological Sampling and Gamma Scans Aboard the N/S SAVANNAH Conducted for the U.S. Maritime Administration April 2005

Keith Welch, Erik Abkemeier, Zachary Edwards
Jefferson Lab, 12000 Jefferson Ave. Newport News, VA 23606

### Background

Thomas Jefferson National Accelerator Facility (Jefferson Lab) has entered into an agreement with the U.S. Maritime Administration (MARAD) to provide support for the Savannah Emergency Response Assessment Team (SERAT) efforts related to N/S SAVANNAH in the event of an incident that might have radiological implications (Reimbursable Agreement #MA-5-A04, to fund Work for Others Project SURA 2004W007). The lab's role is advisory, related to the health physics concerns associated with initial response activities in the event of an emergency.

Commensurate with its role of health physics support for SERAT efforts, Jefferson Lab conducted a series of measurements to confirm the primary nuclides of concern remaining in the reactor systems of the N/S SAVANNAH. This report details the findings of the measurements.

### Acknowledgements

Jefferson Lab was afforded access to the primary plant areas of Savannah in conjunction with work being conducted by WPI. WPI performed radiological and environmental assessments aboard N/S SAVANNAH which required opening all reactor-related spaces. The assessment included a breach of the primary reactor cooling water system, in which samples were taken by WPI. WPI provided various samples to Jefferson Lab for analysis. Jefferson Lab acknowledges the kind assistance of WPI in obtaining data for this report.

## Introduction

The purpose of conducting the measurements described in this report is to obtain a measurement-based estimate of the quantity and distribution of radionuclides in reactor primary systems aboard the N/S SAVANNAH. The specific focus of this assessment is radioactivity in residual liquids and transferable contamination that may be subject to a spill or spread in the event of damage to the ship or flooding of compartments containing reactor systems. In the event of an emergency, knowledge of the reactor plant nuclide inventory is important in facilitating emergency response efforts in which Jefferson Lab may be involved. Previous calculations have been conducted to estimate volumetric nuclide content in the reactor vessel[(1)]. This report does not address volumetric activation (neutron induced activity), but rather the distribution of internal surface contamination and contaminated liquids within piping and components of the primary system. The quantity of radioactivity deposited in the system as contamination is very small compared to the total activity in reactor vessel components.

Internal surface contamination content has been estimated previously, but the present assessment effort provided a rare opportunity to reevaluate radioactivity estimates based on a combination of measurement methods.

## General Approach

Two methods were used to assess the residual radioactivity in the reactor systems. One method was to analyze samples from within the primary system. The second method was by direct scans with portable gamma spectroscopy equipment.

Samples (smears and liquid) taken from the primary system and within the reactor containment were quantitatively analyzed by high-resolution gamma spectroscopy at the Jefferson Lab Radiation Control Group (RCG) radioanalytical lab for presence and amounts of gamma-emitting radionuclides.

The direct gamma scan survey is qualitative in nature, its goal being to gather “snapshots” of the radiation field around various system components to further enhance the understanding of primary system nuclide content.

Equipment and Measurement Techniques

Measurements of sample media were made with a Canberra Industries ultrahigh-purity, coaxial germanium detector (relative efficiency, ~20%) with associated NIM electronics, operated via the Canberra Genie® software package. The system is energy- and efficiency-calibrated for a number of sample geometries annually and receives daily quality assurance checks according to Jefferson Lab RCG procedures. Jefferson Lab also participates in the U.S. Department of Energy Mixed Analyte Performance Evaluation Program (MAPEP) for measurement quality assurance.

Onboard gamma spectra were collected primarily with a Berkeley Nucleonics Corporation Model SAM-935® portable surveillance and measurement system, consisting of a 3 x 3 NaI(Tl) detector coupled to the base unit electronics. The collected spectra can be analyzed with built-in software or uploaded to a PC for analysis using third-party software. A few spectra were also collected with a portable high-resolution germanium detector coupled to a Canberra Inspector® electronics package and analyzed using the Genie® software. This system proved to be difficult to manage in the shipboard environment due to its bulkiness and required a lengthy stabilization period each time the detector was shut down for movement and subsequently restarted.

Energy calibration of the SAM-935 is initially conducted by the factory using a multinuclide source. The calibration coefficients are stored in the firmware of the instrument. Field adjustment/drift correction of the energy calibration is done with an automated calibration routine using a small Cs-137 check source. This routine can be conducted repeatedly at the user's discretion. In addition, to enhance the accuracy the field measurements, some spectra were collected with reference sources present. The reference sources provided gamma rays of known energy, which can be used for a posteriori energy calibration corrections.

Nuclide identification from the SAM spectra was conducted using on-board analysis routines. Some of the spectra were also analyzed using a third-party program, PGT Quantum® gamma analysis software. This was done to conduct manual energy calibration corrections that allowed better photopeak identification when a peak could not be confidently identified by the SAM. Quantum also contains a superior nuclide library.

Energy calibration of the portable high-resolution system was initially conducted at Jefferson Lab, with manual fine adjustments made in the field using reference peaks from small sources and known nuclides in the sampled spectrum.

Scope and Limitations

The direct survey is limited to those nuclides which decay with gamma emissions between approximately 30 keV and 3 MeV. Locations for measurement were chosen with the intent to monitor a reasonable cross-section of systems that contain radioactivity. Consideration had to be given to ambient radiation intensity such that the monitoring system could acquire spectra without encountering detector saturation problems (ambient radiation fields above about 1 mR/hour cause significant detector dead-time), as well as the physical constraints of manipulating the detector and associated equipment within the spaces aboard the ship and protecting the equipment from potential radioactive contamination. Several locations within and outside the primary containment were monitored. Since these measurements were made in a "general area" radiation field involving complex source geometries, quantitative results regarding the concentration of radioactive material are not possible. However, gamma energy peaks provide qualitative verification of the presence and distribution of the most predominant gamma-emitting nuclides.

A limitation inherent in all the area scans is that the spectra include contributions from all sources in the vicinity of the item being monitored. One cannot determine conclusively that the activity indicated is attributable exclusively to the item of interest. Another limitation in assessing the contents of components is the self-shielding of the radiation by the components themselves.

Analysis of samples from the primary system provides the best opportunity to determine what nuclides might be present in the event of a spill from the system. The gamma analysis system used for sample counting has a functional energy range of about 5–2000 keV. Detector response extends below 5 keV (making detection of Fe-55 possible in principle), but sample configuration and self-shielding probably prevent detection of photons below about 5–7 keV.

One goal of the WPI assessment team was to investigate the existence and quantity of water in the primary system beyond the reactor vessel. Steam generator hot-leg access was performed for this purpose. It was discovered that a significant quantity of water

was present in the generators and lower hot-leg piping. Smear and water samples were obtained from inside the steam generators. An estimate of the total contamination inventory is made based on samples from the starboard steam generator. Also analyzed were smear samples from the primary containment enclosure that showed positive results during gross alpha/beta counting.

General Findings

Co-60 was expected to be the most widespread nuclide in the primary system due to the radiological decay characteristics of the isotopes involved. This expectation was confirmed in the measurements taken. All the area monitoring spectra taken around primary systems indicated Co-60 activity. Most monitored locations also indicated the presence of Cs-137 (this may have some practical implications, as is discussed below). A photopeak present in some of the spectra at approximately 75 keV is attributed to lead fluorescence X-rays ($K_\alpha$—72.8 keV, $K_\beta$—74.9 keV), as significant quantities of lead shielding are present around the reactor vessel and in other monitored areas.

The WPI assessment team found very little surface contamination external to primary system piping and components. A few smear samples from reactor spaces showed a combination of Co-60 and Cs-137. In one case, only Cs-137 was present. This is reasonable given the low activity in that area and the ratio of Cs-137 to Co-60 on the other smears (see detailed findings). It might also be surmised that the presence of the contamination is due to past spills of system coolant or ion exchange media, rather than the dry release of crud from piping internals. This deduction is discussed further below.

Detailed Findings

Samples from inside the primary system showed the following characteristics. The steam generator water sample contained Cs-137 almost exclusively (Cs-137 concentration was about 1000 times greater than that of Co-60), but contamination on interior surfaces of the steam generator was found to contain only Co-60. This is undoubtedly a result of the chemical form of the contaminants. Co-60 is usually found as an insoluble oxide and tends to deposit on surfaces of reactor systems (forming the common “crud” deposits found in all reactors), whereas Cs-137 is present as a very soluble oxide or hydroxide.

A spill of the coolant would be likely to spread both Co-60 and Cs-137, as the Co-60 is easily removable and would be flushed from surfaces by any significant movement of the water (hence the speculation above that contamination on surfaces in the reactor compartment may be the result of past liquid (or ion exchange media) spills). A spill to the environment (i.e., into the James River) would probably behave similarly with respect to the distribution of these nuclides. The Cs-137 would likely remain dissolved in the river water, whereas insoluble components would eventually find their way into sediment.

The tables below summarize the area monitoring and sample analysis results. The area scans performed with the SAM 935 contain exposure rate estimates associated with the identified nuclides. This is a calculation made by the SAM using an algorithm that converts counts in a photopeak to an energy-corrected exposure rate. The exposure rate indication provides a reasonably accurate relative intensity measurement.

**Table 1. Area Monitoring Results**

| Scan ref. # | Location | Component or system | Nuclides and exposure rate (μR/h) | | Cs-137/ Co-60 exp. rate ratio | Notes |
|---|---|---|---|---|---|---|
| M1 | Hold Deck, Port passageway | 4” piping below deck level | Cs-137 | 0.95 | 0.03 | |
| | | | Co-60 | 30 | | |
| M2 | Hold Deck Port passageway | Aft end of passage, effluent piping under deck | Cs-137 | 0.08 | 0.05 | |
| | | | Co-60 | 1.6 | | |
| M3 | Hold Deck Port passageway between port charge pumps aft | Small-diameter pipe behind cage chg. pmp. buffer seal system | Cs-137 | 1.4 | 0.24 | |
| | | | Co-60 | 5.9 | | |
| M4 | Port Stabilizer Room | 6” piping from charging pump buffer seal system | Cs-137 | 12.1 | 0.56 | 1 |
| | | | Co-60 | 21.6 | | |
| M5 | Primary Containment upper level | Primary coolant line interface to reactor vessel forward | Co-60 | 581 | N/A | |
| M6 | Primary Containment upper level | Primary coolant line interface to reactor vessel aft | Co-60 | 564 | N/A | |

| Scan ref. # | Location | Component or system | Nuclides and exposure rate (μR/h) | | Cs-137/ Co-60 exp. rate ratio | Notes |
|---|---|---|---|---|---|---|
| M7 | Primary Containment upper level | Upper pressurizer head, port | Co-60 | 183 | N/A | |
| M8 | Primary Containment upper level | Forward upper regen./nonregen. heat exchanger | Cs-137 | 6 | 0.025 | |
| | | | Co-60 | 242 | | |
| | | | Co-60 | 137 | | |
| M9 | Primary Containment 2nd level | Crossover line from upper to lower regen./nonregen. heat exchanger | Cs-137 | 8.5 | 0.062 | |
| M10 | Primary Containment 3rd level | Main pressurizer leg to primary coolant line, just under pressurizer. | Cs-137 | 6.8 | 0.044 | |
| | | | Co-60 | 156 | | |
| M11 | Primary Containment 4th level | Check valve adjacent to forward primary coolant line near vessel | Cs-137 | 19 | 0.053 | |
| | | | Co-60 | 360 | | |
| M12 | Primary Containment 4th level | Reactor vessel (shield tank wall) forward, just starboard of center | Cs-137 | 11 | 0.023 | |
| | | | Co-60 | 479 | | |
| M13 | Primary Containment 1st level | Rx ventilation plenum duct, starboard | Cs-137 | 18 | 0.21 | 2 |
| | | | Co-60 | 84.5 | | |
| M14 | Cold Chem Lab upper level | Rx ventilation duct | Cs-137 | 0.043 | 0.17 | 2 |
| | | | Co-60 | 0.25 | | |
| M15 | Cold Chem Lab lower level | Primary sample sink, sample bulb inside sink hood | Cs-137 | 3.6 | 0.015 | |
| | | | Co-60 | 242 | | |
| M16 | Port Charge Pump Room | Between pumps at aft bulkhead | Cs-137 | -- | 0.096 | 3 |
| | | | Co-60 | -- | | |

1. Exposure rates estimated.
2. Measurement on ventilation ducts.
3. Measured with high-resolution Ge detector. Ratio taken from peak area data.

**Table 2. Sample Analysis Results**

| Ref # | Location/Component | Sample Type | Nuclides | Activity |
|---|---|---|---|---|
| S1 | Starboard steam generator tube sheet | Smear | Co-60 | 144,300 dpm/ 100 $cm^2$ |
| S2 | Starboard steam generator interior (average)[1] | Smear | Co-60 | 22,000 dpm/ 100 $cm^2$ |
| W1 | Starboard steam generator water | 100 mL water | Cs-137 | 1.04E-3 µCi/ml |
| | | | Co-60 | 1.45E-6 µCi/ml |
| S3 | Reactor 3rd level forward at pressurizer (highest)[2] | Smear | Cs-137 | 1200 dpm/100 $cm^2$ |
| | | | Co-60 | 250 dpm/100 $cm^2$ |
| S4 | Reactor 1st level forward Rx head (average)[3] | Smear | Cs-137 | 350 dpm/100 $cm^2$ |

[1] Average of four smears, excludes tube sheet.

[2] Composite count of six smears, all activity attributed to one smear.

[3] Composite count of five smears, activity averaged over the total.

Calculation of Total Contamination Inventory

The total contamination inventory for the primary system was estimated based on the sample data. The contamination inventory is broken into two parts: internal surface contamination and contamination entrained in residual coolant.

Surface Contamination

The surface contamination estimate begins with an assessment of the steam generator contamination content. Published industry data(2) indicate that in pressurized-water reactors (PWRs), the majority of coolant-borne corrosion/fission products that are not removed by the chemical volume and control system (CVCS) are deposited in the steam generators. For a *reference PWR**, the generators contain about 85% of the total deposited activation product inventory. The balance of the activity is distributed in various other components based on relative surface area and deposition characteristics of the system/component.

Steam generator activity content was estimated based on the highest contamination level found in the starboard generator. Assumptions for the calculation are as follows:

- The only nuclide of concern for surface contamination is Co-60.
- Smears were taken over a 100-$cm^2$ area.
- The removal factor for smears is assumed to be 0.1

Steam generator dimensional estimates:

- Tube diameter: 0.5 inches (1.27 cm)
- Average tube length: 30 feet (900 cm)
- Number of tubes: 2000
- Shell interior diameter: 100 cm
- Total plenum length: 100 cm

*The *Reference PWR* in the literature was the Trojan Nuclear Plant. Distribution of radioactivity in three other PWRs was evaluated and reported in Ref. 2. The percentage of radioactivity deposited in steam generators was similar in each case.

Tube surface area: $2\pi(0.635)(900)(2000) = 7.18E6\ cm^2$.

Total tube sheet area: $2[\pi(50)^2 - \pi(0.635)^2(2000)] = 1.06E4\ cm^2$.

Plenum area: $2\pi(50)(100) = 3.14E4\ cm^2$.

Internal surface area of one steam generator: $7.18E6 + 1.06E4 + 3.14E4 = 7.2E6\ cm^2$.

Total activity in one generator in curies is calculated as follows:

$$\frac{144{,}300\ dpm \times 7.2E6\ cm^2}{0.1 \times 100\ cm^2 \times 2.22E12} = 0.0468\ Ci,\ \text{or}\ 93.6\ mCi\ \text{for both steam generators.}$$

Adjusting for reactor/steam generator surface area ratios and unit layout (2-loop vs. 4-loop), activity distribution assignments were made based on the reference PWR. Associated activity levels were calculated and are summarized in the following table.

**Table 3. Total Surface Contamination Inventory**

| System | Activity distribution (%) | Total activity (Ci) |
|---|---|---|
| Reactor vessel and internals | 5 | 0.0054* |
| Steam generators | 87 | 0.0936 |
| RCS[1] piping | 3 | 0.0032 |
| Non-RCS piping | 2.3 | 0.0025 |
| Pressurizer | 0.2 | 0.0002 |
| Other | 2.5 | 0.0027 |
| Totals | 100 | 0.108 |

* Excludes volumetrically distributed activation products in the reactor vessel

[1] RCS = Reactor Cooling System (main cooling loops)

Contamination in Residual Coolant

Using visual indications from the steam generator coolant content, the estimated volume of water in the primary system is calculated below, with the associated total radioactivity.

Volume of generator primary side: $\pi(0.635)^2(900)(2000) + \pi(50)^2(100) = 3.1E6$ cc (mL).

In addition, a portion of the RCS hot and cold legs run horizontally into and out of the generator. The total length of this piping is estimated to be about 26 feet (780 cm) for each loop. The piping diameter is estimated at 18 inches (45 cm).

Volume of horizontal piping: $\pi(22.5)^2(780) = 1.2E6$ mL.

Total volume of contiguous horizontal coolant envelope (1 loop): 1.2E6 + 3.1E6 = 4.6E6 mL.

The water level in the starboard generator was observed to be about halfway up the generator tube sheet; the port generator was reported to be about one-third full. For this estimate, both will be considered half full.

Total water volume in horizontal legs: 4.6E6 (2) = 4.6E6 mL (~1200 gal).



It has been estimated by others that about 1100 gallons of water resides in the lower reactor head. We estimate another 200 gallons is distributed around the balance of the

reactor systems (this is based partly on the observation discussed below regarding location of liquid via the presence of Cs-137). This brings the total volume to 2500 gallons ($9.5E^6$ mL). Assuming the activity in the water is uniform through the plant and represented by the activity in the steam generator, the total activity is:

Cs-137—(1.04E-3 µCi/mL)(9.5E6 mL) = 9840 µCi.
Co-60—(1.45E-6 µCi/mL)(9.5E6 mL) = 14 µCi.

Additional Observations and Some Speculation

The observed distribution of Co-60 and Cs-137 might serve as an indicator of the presence of liquid within various systems and components. If the same physical separation of nuclides found in the steam generator is assumed to exist throughout the system, one could use the presence of Cs-137 in an area scan of primary piping as an indicator of liquid in the component in question. If only Co-60 is present, it may be an indication that the piping or component is internally dry or contains little liquid.

The results of the area scans taken qualitatively support this idea. For instance, no Cs-137 was seen in scans of the upper main coolant lines at their interface to the Reactor Vessel. By comparison, all the scans of the lower-level reactor compartment (containing the primary side of the steam generators and other low-point piping) show Cs-137. Although not conclusive, these data are is consistent with the hypothesis that dry piping contains little or no Cs-137 contamination. The ratio of Cs-137 to Co-60 activity was found to be highest near piping outside the primary containment in the lowest levels of the ship (e.g., piping in the Hold level, Stabilizer Room lower level, and Charge Pump Room). Table 1 includes these ratios for information purposes.

Several gamma scans were taken on reactor ventilation ductwork, both inside and outside the primary containment. In these scans, the ratio of Cs-137 to Co-60 is considerably higher than in primary piping. (It is difficult to state this conclusively since the source of the radiation in any given scan cannot be isolated to a particular component, but this limitation is inherent to all the scans.) Based on the characteristics of the contaminants, we surmise the following process. Soluble Cs-137 was preferentially released to the atmosphere (compared to Co-60) during plant operations via “weeping” of small primary system leaks. The dissolved cesium contamination was released as an aqueous vapor and distributed through the ventilation system, some of it

being deposited within the system. No samples from within the ventilation system were analyzed to confirm the nuclide ratio.

Conclusions

Scans and samples confirmed that the primary nuclide deposited on surfaces in reactor systems aboard N/S SAVANNAH is Co-60. It is estimated that the total inventory of Co-60 in surface deposits is approximately 100 mCi. This figure is in reasonable agreement with previous estimates[(1)]. Cs-137 is the predominant nuclide present in residual water within the primary system. We estimate the presence of about 2500 gallons of water total within the primary system. The total waterborne Cs-137 content in the reactor system is estimated at about 10 mCi.

We believe that this represents the bulk of the potentially mobile nuclide inventory. This result supports the conclusion that even a worst-case incident aboard SAVANNAH, resulting in the loss of all the transferable contamination to the environment, would have no significant impact on the environment or on dose to the public. The conclusion is based on the results of the characterization program that indicates that the radiological consequences of a breech to the primary system would be insignificant. It is not a problem due to dilution. The water in the primary system needs to be disposed of before decommissioning.

References

1. "Nuclear Ship Savannah Reactor Vessel, Internals and Neutron Shield Tank Characterization and Classification Assessment," April 2004, prepared by R. J. Stouky, J. W. Bowen, R. Ranellone, for U.S. DOT (attached as Appendix 4).

2. "Potential Recycling of Scrap Metal From Nuclear Facilities," Appendix A, September 2001, R. Anigstein, et al., for U.S. EPA. Available at http://www.epa.gov/radiation/docs/cleanmetals/tsd/scrap_tsd_041802_apa1.pdf.

3. Materials from meeting presentation "NS Savannah Decommissioning Plans" for NRC, September 24, 2003.