March 22, 1991

Docket No. 50-320
Dr. Robert L. Long
Director, Corporate Services/
Director, TMI-2
GPU Nuclear Corporation
P.O. Box 480
Middletown, PA 17057

Dear Dr. Long:

SUBJECT: FUEL VERIFICATION MEASUREMENTS IN TMI UNIT 2

Enclosed is a copy of Pacific Northwest Laboratory's (PNL) final report on the fuel measurements taken at TMI-2 during August 27-30, 1990.

Based on the results of the measurements performed by PNL, it appears that for the locations analyzed by PNL your measurement and analysis methodology generally ensures a conservative fuel estimate. Five cubicles in the Auxiliary and Fuel Handling Buildings and four incore instrument guide tube bundles were measured. In all but one measurement your estimates of the fuel quantities were higher than the PNL central estimates. The single exception was Incore Instrument Guide Tube Bundle 7. Your estimate of fuel remaining in this bundle was within the range estimated by PNL.

The NRC staff and PNL are looking forward to the reactor vessel measurement program scheduled for late this summer. PNL will also provide assistance to the staff in our final radiological survey of the facility prior to entry into Post Defueling Monitoring Storage.

Sincerely,
Original signed by:

Seymour H. Weiss, Director
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
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and Special Projects
Office of Nuclear Reactor Regulation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Docket File 50-320	016
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LPDR	016
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FUEL VERIFICATION MEASUREMENTS IN TMI UNIT 2

S. E. Merwin

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February 28, 1991

SUMMARY

At the request of the U.S. Nuclear Regulatory Commission (NRC) Staff, Pacific Northwest Laboratory (PNL) performed radiological measurements in the Three Mile Island Nuclear Station, Unit 2 (TMI-2) facility to verify the remaining fuel quantities. The goals of the fuel verification measurements were: 1) to provide information regarding the quantity of fuel remaining in the TMI-2 facility during the post-defueling monitored storage period proposed by the licensee and 2) to ensure that the possibility of an inadvertent criticality was precluded for both routine conditions and conditions involving the accidental shifting or movement of fuel. To achieve these goals, the PNL measurement program attempted to determine a precisely measured value for the remaining fuel debris. In contrast, the licensee's measurement program attempted to ensure that the quantity of fuel was not underestimated.

The measurements were performed by PNL on August 27-30, 1990. Measurements were obtained in five cubicles in the Auxiliary and Fuel Handling Buildings (AFHB). In addition, measurements of four incore instrument guide tube (IIGT) bundles were obtained in the reactor building. In all cases, the PNL measurements were conducted using an intrinsic germanium detector to measure the photon flux from ^{154}Eu, a known fuel analog. Fuel quantity was determined from the measured flux values using a computer code and a predetermined ^{154}Eu-to-fuel ratio. A comparison of the fuel quantities estimated independently by PNL and the licensee, GPU Nuclear Corporation (GPU), is provided in Table S.1.

Based on the results of the measurements performed by PNL, it can be concluded that the licensee's analysis methodology generally ensured a conservative fuel estimate. In all but one location, the licensee's estimates of the fuel quantities were higher than the PNL central estimates. The single exception was Incore Instrument Guide Tube Bundle 7; however, the licensee's estimate of fuel remaining in Incore Instrument Guide Tube Bundle 7 was within the range estimated by PNL. For seven of the nine locations, the licensee's estimates were within the PNL minimum and maximum estimates. In the other two locations, the licensee's estimate was significantly higher than the PNL estimate.

TABLE S.1 Estimated Residual Fuel Quantities in Specific TMI Unit-2 Locations

Location	Estimated Fuel Quantity, grams		
	GPU	PNL	
		Central Estimate	Range
Makeup Pump Cubicle	70	5.5	2.6 to 9.3
Waste Transfer Pump Cubicle	<10	1.0	0.47 to 1.7
RCBT 1A[a] Cubicle	310	170	80 to 290
Makeup Tank Cubicle	310	100	30 to 1300
SDS Monitor Tanks[b]	<1000	10	1.2 to 30
IIGT 5[c]	1000	680	320 to 1000
IIGT 7	200	300	140 to 510
IIGT 10	2300	280	130 to 470
IIGT 25	400	230	110 to 390

(a) RCBT = Reactor Coolant Bleed Tank
(b) SDS = Submerged Demineralizer System
(c) IIGT = Incore Instrument Guide Tube

CONTENTS

FIGURES

TABLES

ACRONYMS

AFHB - Auxiliary and Fuel Handling Buildings

GPU - GPU Nuclear Corporation

IIGT - Incore Instrument Guide Tube

MDL - Minimum Detectable Level

NRC - U.S. Nuclear Regulatory Commission

PNL - Pacific Northwest Laboratory

RCBT - Reactor Coolant Bleed Tank

SDS - Submerged Demineralizer System

TMI-2 - Three Mile Island Nuclear Station, Unit 2

1.0 INTRODUCTION

As a result of the March 28, 1979 accident at Three Mile Island Nuclear Station, Unit 2 (TMI-2), fuel debris was transported and relocated within the reactor coolant system, the reactor building and the auxiliary and fuel handling building (AFHB). The subsequent cleanup activities have included the removal of much of the remaining fuel debris from within the reactor vessel as well as that transported ex-vessel.

The licensee (in a letter to the NRC dated December 2, 1986) proposed placing the TMI-2 facility in monitored storage for a period of time following removal of the damaged fuel. On February 22, 1990, the licensee provided the U.S. Nuclear Regulatory Commission (NRC) Staff with the final submittal of their Defueling Completion Report. This report was supplemented by a letter dated April 12, 1990, which contained both the results of the cleanup following the lower head sample program and a revised criticality analysis. The report, as supplemented, provides the licensee's estimate of the quantity of fuel remaining in each location of the TMI-2 facility as well as presenting the results of the licensee's analysis concluding that there is no possibility of an inadvertent criticality resulting from either credible or incredible conditions.

The NRC Staff requested assistance from the Pacific Northwest Laboratory (PNL) to verify the licensee's measurements of the quantity of fuel remaining in the TMI-2 facility. The purpose of the fuel verification measurements was twofold: 1) to provide information regarding the quantity of fuel remaining in the TMI-2 facility during the post-defueling monitored storage period proposed by the licensee and 2) to ensure that the possibility of an inadvertent criticality is precluded for both normal conditions and conditions involving the accidental shifting or movement of fuel. For this reason the PNL measurement program attempted to determine a precisely measured value for the remaining fuel. In contrast, the licensee's measurement program attempted to ensure that the quantity of fuel remaining in the facility was not underestimated. The methods and instrumentation used by PNL differed from that used by the licensee because of the timing of the measurements (PNL's measurements were conducted several years after the licensee's during which

time a substantial decrease occurred in the activity of some key radionuclides). The use of different measurement techniques provided a better test of the licensee's results than would have strict duplication of the licensee's measurement techniques.

This report contains the results of verification measurements performed by PNL including a description of the methods used to choose the sampling locations, the performance of the measurements, and the data analysis methods. The results obtained from these measurements and analysis are compared to those obtained by the licensee, GPU Nuclear Corporation (GPU).

2.0 SAMPLE LOCATIONS

Fuel verification measurements were performed on August 27-30, 1990 in both the auxiliary and fuel handling buildings (AFHB) and the reactor building. The following locations were selected:

Auxiliary and Fuel Handling Buildings

- Makeup Pump Cubicle
- Waste Transfer Pump Cubicle
- Reactor Coolant Bleed Tank (RCBT) 1A Cubicle
- Makeup Tank Cubicle
- Submerged Demineralizer System (SDS) Monitor Tanks

Reactor Building

- Incore Instrument Guide Tube (IIGT) Bundle - 5
- IIGT Bundle - 7
- IIGT Bundle - 10
- IIGT Bundle - 25

The locations were selected to span the range of the licensee's estimated fuel quantities. For example, locations were included where the licensee was unable to measure any fuel and thus was obligated to estimate the quantity of fuel from the minimum detectable limits of the measurement equipment. Locations were also selected where the estimated quantity of fuel was greater than 1 kg. However, rather than randomly selecting locations based solely on the quantity of fuel, it was necessary to select locations based on other criteria, including the general area dose rates, accessibility, and geometry of components. The general area dose rates were an important factor because the performance of the measurement equipment in terms of accuracy and precision was greater in lower dose rate fields. The accessibility of the location depended on both the presence of plant components that could interfere with the placement of equipment and the dose rates to personnel performing the measurements. Also, it was advantageous to select locations where the measurements could be performed as far away from the fuel debris as possible in order to provide the best possible analytical precision. All of these factors were considered in selecting the locations where measurements were obtained.

3.0 MEASUREMENT AND MODELING APPROACH

The optimal approach for quantifying the amount of fuel remaining in the TMI-2 facility would have been to directly measure the photons emitted by the uranium fuel and to correlate the measured flux to the amount of fuel present. However, uranium radionuclides do not emit photons in sufficient quantities and energies to permit accurate measurements. Therefore, gamma photon flux rates from a known fuel analog were measured at one or more locations in each sampling area. (A fuel analog is a radionuclide that had been determined to remain with the fuel during the accident, rather than being leached out or dissolved in the reactor coolant system water.) The facility components and potential source locations and quantities were modeled and a computer code was used to calculate gamma dose rates at specified locations in the cubicle. The source locations and quantities were varied until the calculated dose rates at the measurement locations were similar to the dose rates measured using the detector. Fuel quantities were then determined by converting the fuel analog quantities to fuel quantities based on a documented fuel analog-to-fuel ratio.

Details on the instrumentation, measurement procedures, modeling approach, and error analysis are provided in Sections 3.1 through 3.4.

3.1 INSTRUMENTATION

The instrumentation system that was used to measure residual fuel quantities at TMI-2 was selected based on both its portability and its ability to operate in relatively high dose rate areas (up to 40 mR/h). This ability is attributable to the relatively low efficiency of the detector and the presence of a transistor reset preamplifier. This system consisted of a high purity coaxial germanium detector manufactured by EG&G Ortec, Oak Ridge, TN (Model Number GMX05190-P-Plus-S). The germanium crystal was cooled for operation using liquid nitrogen. The crystal size was 29.2 mm diameter and 29.2 mm long and was located behind a 0.5 mm beryllium window. The detector specifications included a peak-to-Compton ratio of 25.3 and a relative efficiency of 2.2% for 1.33 MeV photons.

The detector had an internally-mounted transistor reset preamplifier, and a recommended operating bias of -1500 volts. The preamplifier was

internally mounted in the detector to avoid many of the problems associated with conventional resistive feedback preamplifiers, and allowed the use of the detector in dose rate fields up to 40 mR/h.

A shield was fabricated for the detector to improve resolution in several locations of the plant where the dose rates were relatively high. The shield protected the detector from some of the scattered radiation without decreasing the detection efficiency for the photons of interest. The shield was made from an alloy of 90% lead, 5% tin, and 5% antimony. The front of the shield contained a removable circular plug (30 mm diameter) that allowed an unobstructed view of the full diameter of the germanium crystal. The shield was enclosed in an aluminum jacket to prevent damage to the soft alloy material. The shield was approximately 1.3 cm thick at the front end and approximately 1.0 cm thick elsewhere. A spring loaded aluminum flapper held the front shield plug in position. The shield was mechanically supported away from the liquid nitrogen dewar to minimize stress on the detector.

The supporting electronics for the detector consisted of commercially available instruments manufactured by Canberra Industries, Inc., Meriden, CT. The high voltage bias supply was Canberra Model 3105 and the amplifier was Canberra Model 2020 Spectroscopy Amplifier with a shaping time of 0.25 microseconds. The multi-channel analyzer was Canberra Model 35 Plus with an external Canberra Model 8075 analog-to-digital converter. The pulse height spectra from the multi-channel analyzer were downloaded to a laptop Gridcase 3 computer for subsequent data analysis.

3.2 MEASUREMENT PROCEDURES

The specific measurement locations and counting times within each area were determined based on the characteristics of the area including the size of the area to be measured, the amount of background radiation present, the strength of the signal from the fuel analog, and the existence of undesirable shielding by system components. In general, measurements were performed as far away from the source as possible to provide the best possible geometry for analysis. When possible, significant shielding by system components was avoided by placing the detector in a position such that a direct line of sight existed to the potential source locations.

In some cases, physical characteristics of the area precluded measurements in certain locations. For example, in the Makeup Pump cubicle, water on the floor prevented personnel access to the majority of the room. In several cubicles, locations where measurements were originally planned had background radiation dose rates that were too high for the detector to function properly, thus precluding measurements in these locations and requiring the selection of other detector locations. In other cases, personnel access to certain areas of the cubicles was impossible because of the physical placement of system components and the lack of sufficient space for personnel and equipment access.

The number of measurements performed depended on the extent and complexity of the potential source geometry. For example, only one measurement was obtained for each Incore Instrument Guide Tube (IIGT) bundle because the geometry was straightforward and there was a large enough area for the detector to be placed far enough from the bundle to minimize the effect of uneven source distributions within the bundle. On the other hand, six measurements were performed in the Reactor Coolant Bleed Tank (RCBT) 1A cubicle because of the large size of the tank and the potential for the source to be distributed unevenly within the tank.

3.3 MEASUREMENT INTERPRETATION

Each measurement resulted in a recorded gamma spectrum. The process of relating the measured spectrum to the quantity of fuel present consisted of three basic steps: 1) converting the measured spectrum to a gamma flux rate for a specific gamma (or gammas) associated with the chosen fuel analog(s), 2) calculating the activity of the fuel analog(s) present in the system components, and 3) converting the fuel analog activity to a fuel quantity based on accepted fuel analog-to-fuel ratios. Each of these steps is discussed separately below.

Determination of Gamma Flux

The gamma flux rate was determined from the measured gamma spectrum by determining the number of counts collected in the relevant photopeak(s) and adjusting for the detector efficiency, the detector area, and the length of

the count time. Although the collected spectra contained numerous photopeaks, it was necessary to limit the analysis to those peaks associated with known fuel analogs. Based on studies by GPU (Calculation Number 4550-4400-87-026), it was determined that ^{144}Ce and ^{154}Eu represented the most reliable fuel analogs. ^{154}Eu was by far the easier of the these two radionuclides to detect; because of its relatively long half-life (8.8 years), it was still present in relatively large amounts. In contrast, the ^{144}Ce had decayed to very low levels due to its relatively short half-life (284 days). Although small amounts of ^{144}Ce were detected, it was not analyzed and the fuel analog ^{154}Eu was selected for the spectral analysis.

The radionuclide ^{154}Eu emits many photons, ranging in energy from 123 keV to 1.597 MeV. For the fuel verification measurements, the high-energy photons were of greatest interest because they were least affected by shielding from the system components and were readily apparent above the ^{137}Cs (background) photopeak. Of the high-energy photons, the 1.274 MeV photon has a much higher abundance (35.5%) than the others, and, therefore, was most prominent on the gamma spectra. Furthermore, its photopeak was not interfered with significantly by background gammas. As a result, the 1.274 MeV photon from ^{154}Eu was the primary photon analyzed.

Calculation of the 1.274 MeV gamma flux rate was accomplished by determining the number of net counts above background in the relevant photopeak on the gamma spectrum and dividing by the detector efficiency, detector area, and the count time. This procedure was performed for each gamma spectrum measured. Detector efficiency was determined by measuring a calibrated ^{154}Eu source and dividing the number of net counts above background in the 1.274 MeV photopeak by the number of 1.274 MeV photons that entered the detector. The calibrated source was far enough from the detector to ensure that the emitted photons approximated a parallel beam at the detector face. Calibrating to a parallel beam of photons eliminates the dependence of the calibration constant on the source-to-detector distance exhibited by a point source located close to a detector. The efficiency value determined by these measurements can be used to reliably convert collected counts under the photopeak to incident flux for all of the geometries encountered in these studies.

Detector efficiency also varies with detector dead time, so additional calibration measurements were performed in several of the cubicles. Efficiencies for each measurement location were determined from a curve that was fit to the calibration data as a function of dead time.

Determination of Fuel Analog Activity

Because of the complexity of the system components and the wide variability in potential source locations and quantities, a computer code was used to estimate the ^{154}Eu (fuel analog) activities present in the system components based on the measured gamma flux rates. For this purpose, the computer code WISE, developed previously under the name SPARC for the Electric Power Research Institute (Reece et al. 1987), was used. The code calculated the dose rates at specified locations from radioactive sources in specified geometries. The calculations were based on point-kernel algorithms. The sources and any shields present could be modeled by several shapes including pipes, planes, cylinders, points, disks, or parallelpipeds, thus allowing realistic modeling of the system components and fuel deposits. The results from the code were verified by performing selected calculations manually.

Each cubicle or measurement location was modeled as accurately as possible using the computer code. Sources of 1.274 MeV gammas were then modeled in various system components in a distribution that appeared reasonable based on the system measured. Dose rates (from unscattered gammas only) were then calculated at each actual measurement location. The source distributions in the components were then varied until the calculated dose rates matched the measured dose rates as closely as possible in each measurement location[1].

In two locations (Waste Transfer Pump cubicle and Makeup Tank cubicle), several different source geometries that were both possible and plausible resulted in excellent matches between the calculated and measured flux rates.

---

[1]This comparison required the conversion of gamma flux rate determined from the measured spectra to dose rate. This was accomplished by multiplying the flux rates by a conversion factor for 1.274 MeV gammas of 2.34×10^{-5} rem h^{-1} cm^2 s calculated using linear interpolation of values published for other energies (Isoulfanidis 1983).

In these cases, the range of calculated fuel quantities associated with the different source geometries were incorporated into the specified fuel estimate.

Estimation of Fuel Quantity

The quantity of fuel in a specific cubicle or component was determined by dividing the calculated ^{154}Eu quantity by a predetermined ^{154}Eu-to-fuel ratio. Based on samples taken by the licensee through 1988, the estimated ^{154}Eu-to-fuel ratio was 33.4 $\mu Ci/g$ (decay corrected to August 29, 1990) (TB 86-41, Rev. 2). The uncertainty in this value, as documented by the licensee, is 86% (Calculation Number 4410-89-L-0097/0356P). However, as described below, a different value for the uncertainty was applied to the PNL measurements.

3.4 ERROR ANALYSIS

There are several sources of potential error associated with the fuel estimates provided in this report. These errors can be classified into three general types: precision, systematic and modeling. Although modeling errors can be considered systematic, they are identified separately in this report because of their relative importance. For each set of measurements, each type of error was estimated and then combined to produce a minimum, central and maximum fuel estimate for each measurement location. Of the three types of errors, precision and modeling errors were the most significant with respect to the fuel estimates.

Precision Error

Precision errors refer to the random variability of a set of observations. For example, the error associated with the number of counts obtained in a specific energy range in a gamma spectrum is a precision error. These errors can be estimated directly from the recorded data and are measured by the standard deviation of the set of observations with respect to the mean value (NCRP 1985).

There were several potential sources of precision error in deriving the fuel estimates. These primarily included 1) the ^{154}Eu-to-fuel ratio, which was estimated by the licensee from measurements of numerous samples, 2) fuel

source counting errors, and 3) calibration source counting errors. The licensee estimated the ^{154}Eu-to-fuel ratio to be 33.4 μCi/g (corrected for decay) with a standard deviation of 86% (Calculation Number 4550-4400-87-026). However, this standard deviation is inappropriate for this analysis because the value was based on individual samples, most of which were obtained from the fuel debris remaining in the reactor vessel following the accident. It is more appropriate for application to the PNL measurements to use the standard deviation of mean ratios for specific areas, cubicles, or components. Based on the mean values determined by the licensee for the reactor vessel, lower head, Once-Through Steam Generator upper tube sheet, and leadscrews, the estimated standard deviation associated with the ^{154}Eu-to-fuel ratio is 15%.

The precision errors associated with source counting are difficult to quantify because repeat measurements were not obtained. However, they are likely to be small because large numbers of counts were obtained in each photopeak analyzed. For this analysis, the standard deviation of both the fuel source and calibration source counting errors is assumed to be 10%.

The proper method for combining individual precision errors to determine total precision error in the fuel estimate is the method of error propagation (Knoll 1979). Stated simply, the total standard deviation is the square root of the sum of the squares of each individual standard deviation. For this analysis, precision errors are stated as 95% confidence limits, which correspond to approximately two standard deviations. Therefore, the total precision error for each fuel estimate is estimated to be 41%.

Systematic Error

Systematic errors cannot be estimated directly. They refer to errors that are repeated in both magnitude and direction. For example, consistently measuring incorrectly the distance from a calibration source to a detector would be a systematic error. These errors were minimized through careful checks and calibration of the equipment and review of the data. (It was assumed that the error associated with the activity of the ^{154}Eu calibration source is very small). A subset of systematic error is modeling error, which is described below.

Modeling Error

Modeling errors refer to the uncertainties associated with the output of the computer code used to analyze the data, and include both the errors inherent in the algorithms in the code and the uncertainties associated with the input to the code. The latter may include uncertainties in the dimensions of system components and the distance between components and detectors.

Determining the modeling errors associated with a specific fuel estimate was not straightforward. Errors may have resulted from geometry uncertainties such as wall thicknesses of pipes and pumps or uncertainties associated with fuel distributions. The magnitude of the potential errors associated with these uncertainties varied widely depending on the specific cubicle or component measured. In some cases, unknown or uncertain parameters were varied in the model to determine the dependence of the outcome on their values. For each case, an estimate of the modeling uncertainty is provided as an upper and lower range around a central estimate. It is assumed that the minimum modeling uncertainty associated with any fuel estimate is 20%.

Total Error

The total error associated with each fuel estimate is a combination of the associated precision and modeling errors. For all estimates in this report, the total error is provided as an upper and lower limit around a best estimate. The upper and lower limits are derived using a two-step process: first, upper and lower limits are estimated based on modeling uncertainty; second, the limits are extended based on the 95% confidence limits associated with the total precision error. Modeling uncertainties are assumed to be at least 20%; higher values were used when it was appropriate. As described previously, precision errors were estimated to be 41% based on error propagation and estimated values for individual precision errors.

4.0 ANALYSIS AND RESULTS

The results of the fuel verification measurements and analysis are presented in this section. Section 4.1 presents a summary of the results. Sections 4.2 and 4.3 present details on the measurements and analysis for the Auxiliary and Fuel Handling Buildings (AFHB) and reactor building, respectively. A discussion of the differences between the methodologies used by GPU and PNL is also provided.

4.1 SUMMARY OF ESTIMATED FUEL QUANTITIES

Table 4.1 lists the fuel quantities (in grams) estimated by GPU and PNL for each of the nine measurement locations. Included with PNL's best estimates are upper and lower limits calculated using the methods described in Section 3.4. Although GPU did provide precision uncertainties associated with several of their estimates, they are not directly comparable to the upper and lower estimates derived by PNL, and, therefore, are not presented in the table.

TABLE 4.1 Estimated Residual Fuel Quantities in Specific TMI Unit-2 Locations

	Estimated Fuel Quantity, Grams		
		PNL	
Location	GPU	Central Estimate	Range
Makeup Pump Cubicle	70	5.5	2.6 to 9.3
Waste Transfer Pump Cubicle	<10	1.0	0.47 to 1.7
RCBT 1A[a] Cubicle	310	170	80 to 290
Makeup Tank Cubicle	310	100	30 to 1300
SDS Monitor Tanks[b]	<1000	10	1.2 to 30
IIGT 5[c]	1000	680	320 to 1000
IIGT 7	200	300	140 to 510
IIGT 10	2300	280	130 to 470
IIGT 25	400	230	110 to 390

(a) RCBT = Reactor Coolant Bleed Tank
(b) SDS = Submerged Demineralizer System
(c) IIGT = Incore Instrument Guide Tube

4.2 AUXILIARY AND FUEL HANDLING BUILDING MEASUREMENTS

Measurements were obtained in five locations in the Auxiliary and Fuel Handling Building (AFHB). The locations are identified on the AFHB diagram in Fig. 4.1. An additional measurement was made in the Reactor Coolant Bleed Tank (RCBT) 1B and 1C cubicle in order to determine the contribution from sources in this cubicle to the background dose rates in a nearby cubicle. However, this measurement was not used to estimate fuel quantities.

4.2.1 Makeup Pump Cubicle

Figure 4.2 illustrates the components and measurement locations in the Makeup Pump cubicle. Measurements were performed in only two locations in this cubicle because water covered most of the floor and prevented access to all areas of the room except near the entrance. However, the two measurements were sufficient to provide a reliable fuel estimate because the geometry of the components in the cubicle was relatively straightforward.

Measurement 1 was obtained at the entrance approximately 10 cm above the floor. Measurement 2 was obtained near a wall approximately 107 cm above the floor. The 1.274 MeV gamma flux rate measured at Location 2 was 35% higher than the flux rate measured at Location 1. A third measurement (not shown in Figure 4.2) was obtained in the same location as the first measurement; however, a 2 inch-thick lead brick was placed at the front face of the detector. This measurement was performed to ensure that the 1.274 MeV photopeak obtained in the first two measurements was associated only with sources present in the Makeup Pump cubicle.

The computer code WISE was used to estimate the ^{154}Eu source distribution and quantity that would produce 1.274 MeV gamma flux rates similar to those measured. For these calculations, the pump was modeled as a cylinder having a wall thickness of 2.54 cm. It was determined using the code that the measurements were consistent with the calculated flux rates if 100% of the source was inside the pump. The calculated and measured gamma ray flux rates for this source geometry are compared in Table 4.2.

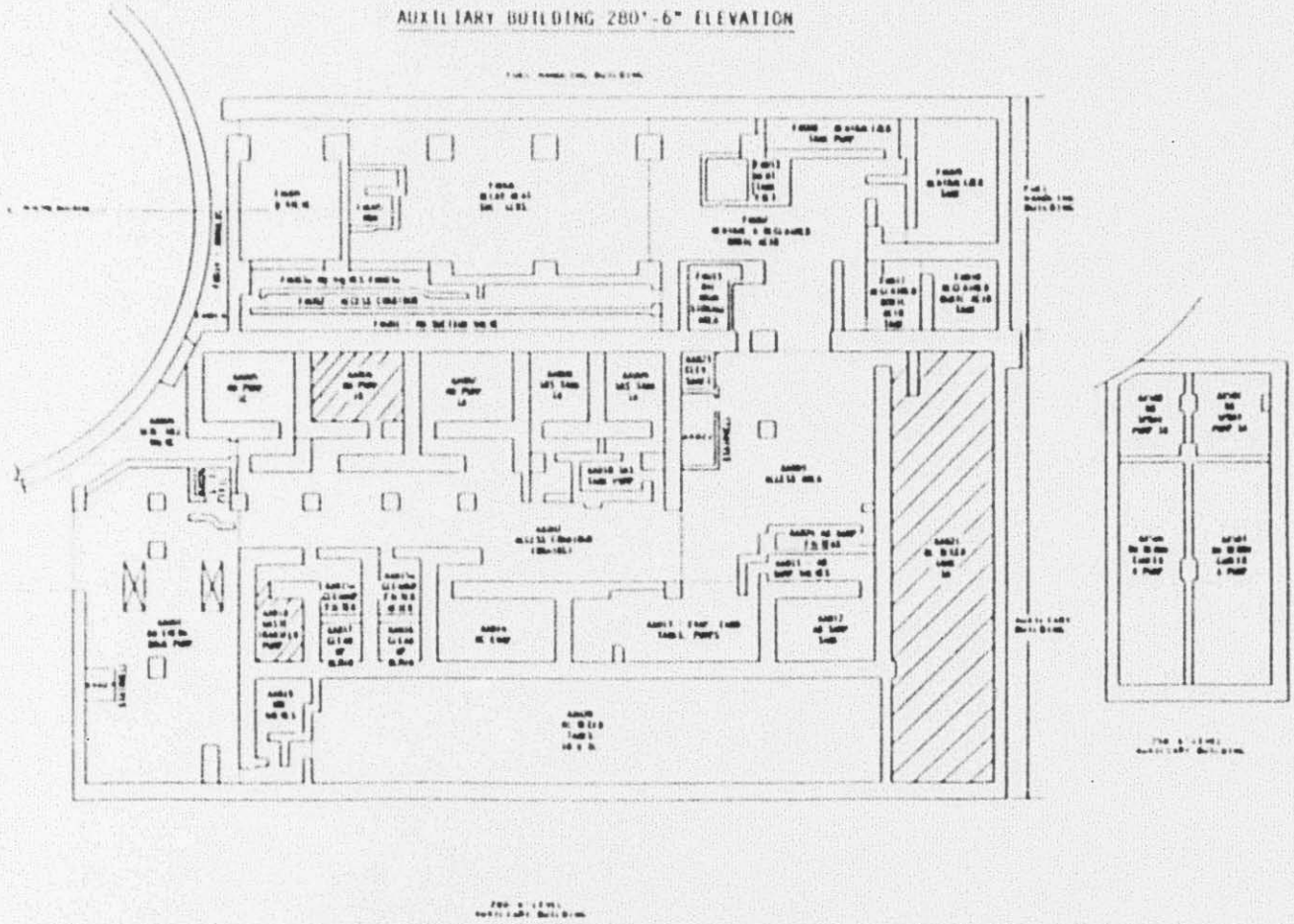


FIGURE 4.1. Measurement Locations in the Auxiliary and Fuel Handling Building

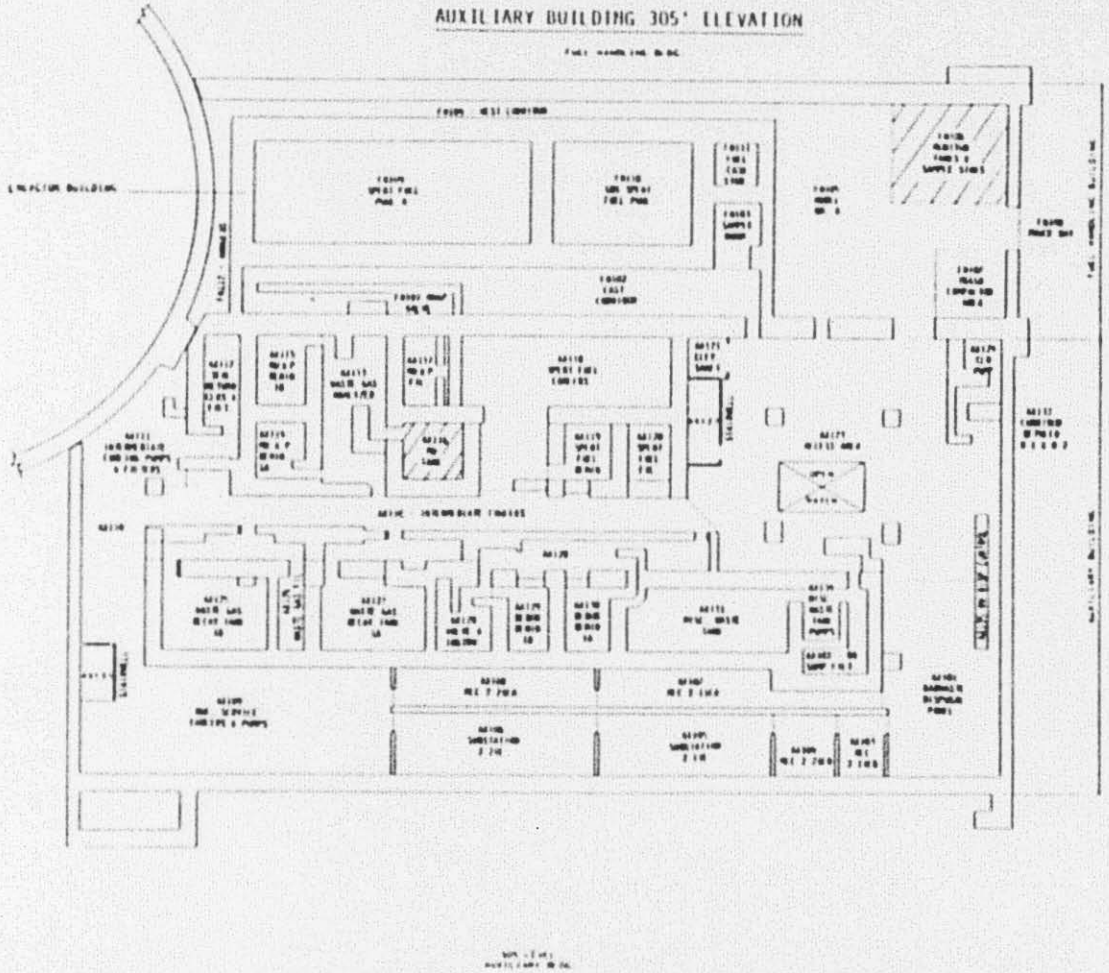


FIGURE 4.1. Measurement Locations in the Auxiliary and Fuel Handling Building (contd)

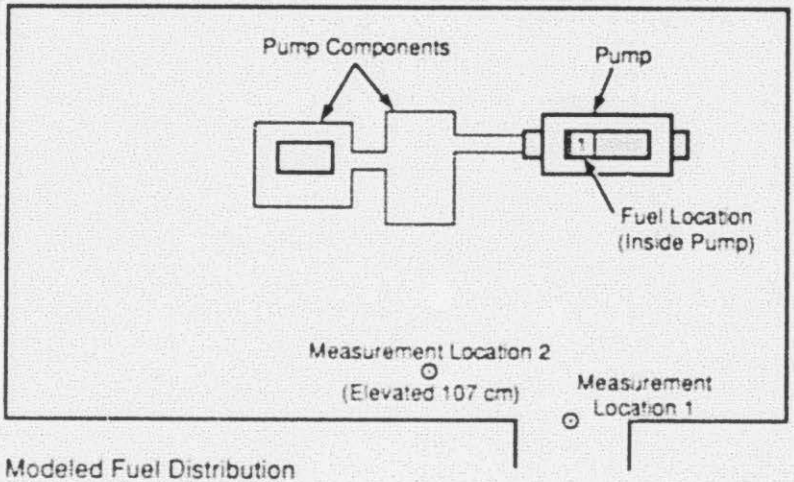


FIGURE 4.2 Measurement Locations in Makeup Pump Cubicle

TABLE 4.2 Calculated and Measured 1.274 MeV Gamma Flux Rates in Makeup Pump Cubicle

Measurement Location in the Makeup Pump Cubicle	Gamma Flux Rate (Relative to Measurement 1)	
	Measured	Calculated
1	1.00	1.00
2	1.35	1.35

Although many other source distributions could have resulted in calculated flux rate ratios identical to those in Table 4.2, the distribution analyzed is realistic based on both the arrangement of the components in the cubicle and the measured flux rates. For example, it is not likely that most of the source was in the surrounding piping (not pictured) because, if this were the case, the ratio of the measured flux rates would have been greater than 1.35. Also, it is not likely that the source was scattered about on the floor. In addition to being realistic, the selected source distribution is unlikely to result in an underestimate of the fuel quantity because the pump wall provides significant shielding relative to that provided by other system components.

Based on the computer calculations, the estimated quantity of fuel in the Makeup Pump cubicle was 5.5 g. Because a modeling uncertainty cannot be quantified based on the single source distribution analyzed and the absence of identified variations in component dimensions, the minimum value of 20% is assumed (see Section 3.4). Using a modeling uncertainty of 20% and the error estimation method described in Section 3.4, the bounds on the fuel estimate are 2.6 g and 9.3 g, respectively.

In contrast to the PNL measurement method, the licensee estimated the quantity of fuel in the Makeup Pump cubicle by measuring the ^{144}Ce photopeak at several locations using a NaI detector. Their estimate of 70 g is significantly greater than PNL's best estimate of 5.5 g and is also greater than the PNL maximum estimate of 9.3 g. This discrepancy is partially attributable to the analysis methodology used by the licensee, whose estimate is based on calculations of the minimum detectable levels (MDL) of fuel. These calculations were necessary because none of the licensee's measurements

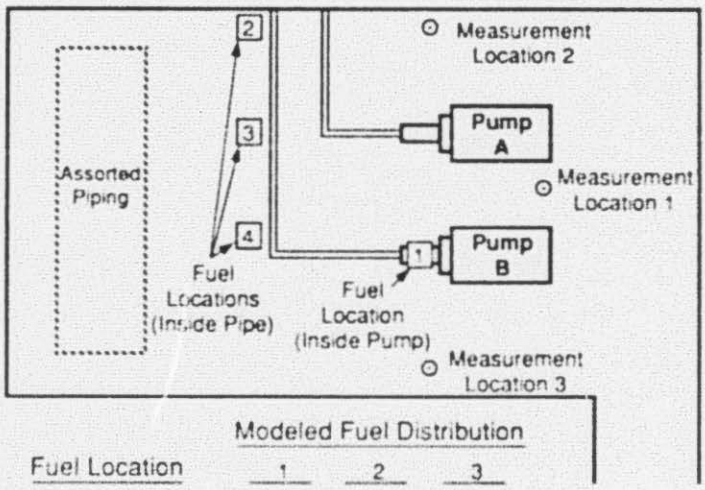
resulted in statistically significant photopeaks. The licensee determined MDL fuel quantities separately for each of nine system components by assuming that the fuel was located only in that component. The individual MDLs were then summed to arrive at a single MDL for the entire cubicle. This methodology overestimated the fuel quantity significantly.

4.2.2 Waste Transfer Pump Cubicle

The waste transfer pump cubicle components and the measurement locations are illustrated in Fig. 4.3. Three measurements were obtained in this cubicle, all approximately 10 cm above the floor. Measurement 1 was obtained between the pump motors. Measurement 2 was obtained near a wall in direct line of sight to one of the pumps (Pump A). Measurement 3 was located symmetrical to Measurement 2, in direct line of sight with the other pump (Pump B).

The measured 1.274 MeV gamma ray flux rates at Locations 1 and 2 were essentially identical and were approximately 40% lower than the measured flux rate at Location 3. A fourth measurement (not shown in Fig. 4.3) was obtained for calibration purposes.

For the computer calculations, each pump was modeled as a cylinder having a wall thickness of 2.54 cm. Pipes were modeled as cylinders having a wall thickness of 0.5 cm. It was determined that the measurements were consistent with the calculated fluxes for many plausible source distributions. Therefore, three different general distributions were analyzed. These distributions assumed that 1) most, 2) some, or 3) none of the fuel was located in Pump B. In each case, the remaining fuel was assumed to be located in the pipes leading from Pump B. Pump A and its components were assumed to contain no fuel because the pump was operating and ion chamber readings of the contact dose rates were much lower than the contact dose rates from Pump B. For each of the three general distributions, the percentages of fuel in the cubicle components were varied until the calculated flux rate ratios for each measurement location were similar to the measured flux rate ratios (see Fig. 4.3 for the resulting amounts). The resulting fuel estimate for each of the three general distributions was then calculated using the method described in Section 3.3.



Modeled Fuel Distribution

Fuel Location	1	2	3
1	71.5%	21.5%	0%
2	7.0%	9.0%	9.9%
3	21.5%	21.5%	21.5%
4	0%	48.0%	68.7%

FIGURE 4.3 Measurement Locations in Waste Transfer Pump Cubicle

Based on the computer calculations, it was determined that the total fuel estimate for the cubicle was relatively independent of the specific source distribution selected, ranging from 0.93 g for Distribution (3) to 1.2 g for Distribution (1). Therefore, the best estimate of the fuel (1.0 g) is based on Distribution (2). The calculated and measured gamma ray flux rates for this source distribution are compared in Table 4.3.

Because the fuel estimate is relatively independent of the fraction of fuel assumed to be in Pump B, bounds on the fuel estimate of 1.0 g were calculated based on a modeling uncertainty of 20% and the methodology described in Section 3.4 (the minimum value of 20% was assumed for the modeling uncertainty because the error based on the potential variations in source distribution was less than 20%). The associated minimum and maximum fuel estimates are 0.47 g and 1.7 g, respectively.

In contrast to the PNL approach, the licensee estimated the quantity of fuel in the Waste Transfer Pump cubicle by measuring the ^{144}Ce photopeak at several locations using a NaI detector. Using this method, the estimated fuel quantity was <10 g. This estimate is consistent with the PNL best estimate of 1.0 g and is also consistent with the PNL minimum and maximum estimates of 0.47 and 1.7 g. However, the PNL and GPU estimates cannot be directly compared because the licensee's estimate was based on calculation of the minimum detectable level (MDL) from three measurements that showed no significant photopeak. In addition, the estimate of <10 g was artificially "rounded up" from the actual calculated MDL of 1.4 g. Furthermore, a fourth measurement performed by the licensee, which did result in a significant photopeak, was ignored. The justification provided for ignoring the

TABLE 4.3 Calculated and Measured 1.274 MeV Gamma Flux Rates in Waste Transfer Pump Cubicle

Measurement Location in the Waste Transfer Pump Cubicle	Gamma Flux Rate (Relative to Measurement 1)	
	Measured	Calculated
1	1.00	1.00
2	0.98	0.98
3	1.58	1.58

measurement was that the resulting maximum fuel estimate of 600 g was "clearly too high" and was attributed to a problem with the instrumentation.

4.2.3 Reactor Coolant Bleed Tank 1A Cubicle

The reactor coolant bleed tank (RCBT) 1A cubicle components and the measurement locations are illustrated in Fig. 4.4. Six measurements were obtained in this cubicle, all approximately 10 cm above the floor. Measurements 1 through 4 were obtained along the wall farthest from the tank and as close to the wall as possible. Measurement 5 was obtained near another wall opposite one end of the tank, also as close to the wall as possible. Measurement 6 (not shown in Fig. 4.4) was a calibration source measurement. Measurement 7 was obtained opposite the other end of the tank. The measured 1.274 MeV gamma ray flux rates were generally highest near the middle of the tank (lengthwise), although the measurement at the tank end near the cubicle entrance (Measurement 7) also indicated a relatively high flux rate. Measurement 8 (not shown Fig. 4.4) was obtained in the same location as Measurement 5, although for that measurement a 2 inch-thick lead brick was placed at the front face of the detector. This measurement was used to ensure that the 1.274 MeV photopeak obtained in the other measurements was associated only with sources present in the Reactor Coolant Bleed Tank 1A cubicle.

For the computer calculations, the tank was modeled as an empty cylinder having a wall thickness of 0.79 cm. The source was modeled as several point sources positioned along the bottom of the tank. Limitations of the code prevented more realistic modeling, such as discrete lumps having varying fuel concentrations. However, the associated errors are minimal because of the relatively large distances from the detector to the bottom of the tank. The point source locations and quantities were varied until the best possible match between the calculated and measured gamma flux rates was obtained (Table 4.4).

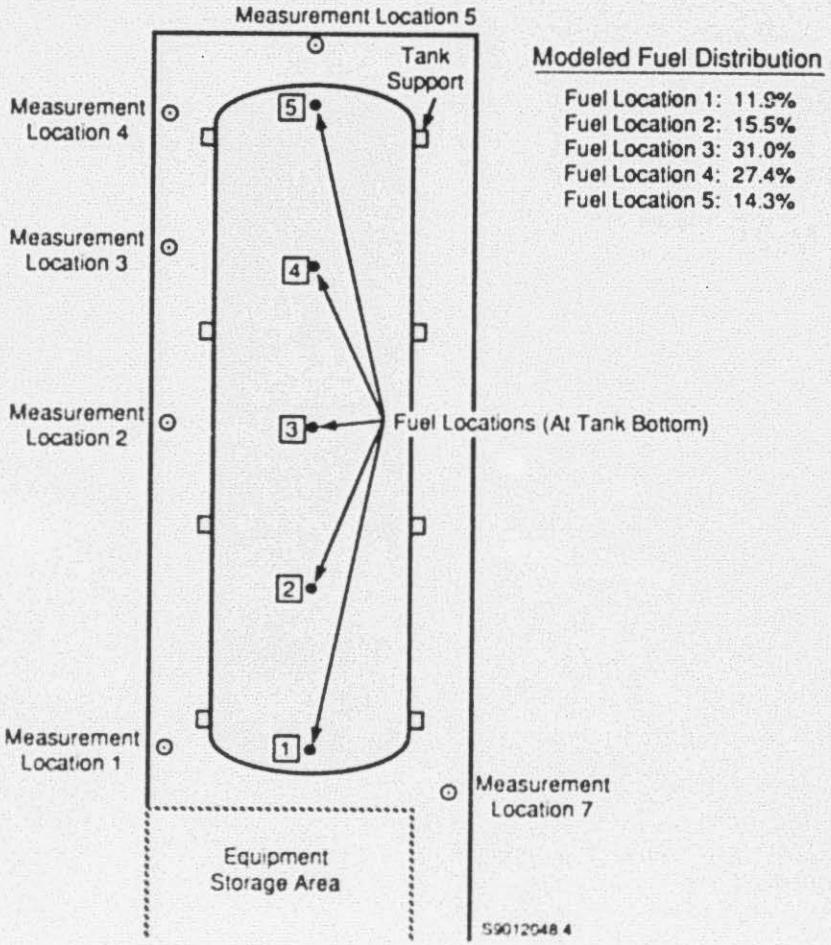


FIGURE 4.4 Measurement Locations in Reactor Coolant Bleed Tank 1A Cubicle

TABLE 4.4 Calculated and Measured 1.274 MeV Gamma Flux Rates in Reactor Coolant Bleed Tank 1A Cubicle

Measurement Location in the RCBT 1A Cubicle	Gamma Flux Rate (Relative to Measurement 1)	
	Measured	Calculated
1	1.00	1.00
2	1.90	1.91
3	2.15	2.15
4	1.19	1.18
5	0.67	0.66
7	1.62	1.65

Although it is unlikely that the modeled source distribution is identical to the actual distribution, the resulting fuel estimate of 170 g is relatively independent of the modeled distribution because the detectors were positioned relatively far from the source. For example, it is not likely that all of the source is at the bottom of the tank; some fraction may be adhered to the tank wall. Also, the source may consist of large fragments rather than discrete points. However, neither of these considerations would result in a significant change in the fuel estimate provided that the calculated flux rates agree with the measured flux rates.

Bounds on the PNL estimate of 170 g were determined based on a modeling uncertainty of 20% and the methodology described in Section 3.4. The associated minimum and maximum estimates are 80 g and 290 g, respectively.

Similar to PNL, the licensee estimated the quantity of fuel in the Reactor Coolant Bleed Tank 1A cubicle by measuring the ^{154}Eu photopeak at several locations using a germanium detector. Using this method, the estimated fuel quantity was 310 g, and the estimated error was 140 g. This estimate is greater than the PNL best estimate of 170 g, although, considering the error bounds for both estimates, the difference is not statistically significant.

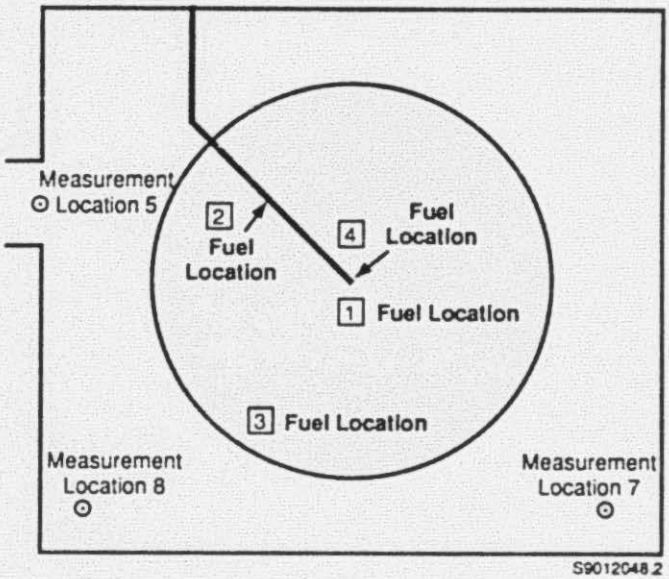
Although the licensee's method for estimating the quantity of fuel in this cubicle was similar, in general, to the PNL method, there were two important exceptions. First, the licensee's measurements were taken directly

underneath and thus very close to the tank. Second, as a result of the detector placement, each detector was assumed to have measured only one discrete tank segment. Consequently, fuel values for individual tank segments were estimated based on single measurements and summed to arrive at a fuel estimate for the entire tank. This measurement and analysis approach is likely to overestimate the fuel quantity and could result in a relatively large analytical error.

4.2.4 Makeup Tank Cubicle

The makeup tank cubicle components and the measurement locations are illustrated in Fig. 4.5. Three fuel measurements were obtained in this cubicle, all approximately 10 cm above the floor. Measurements 1 through 4 were obtained previously during initial measurement preparations and are not discussed in this report. Measurement 5 was obtained near the entrance to the cubicle. Measurement 6 (not shown in Fig. 4.5) was a calibration source measurement. Measurement 7 was obtained in a corner of the room as far away from the tank as possible. Measurement 8 was located in another corner of the room, also as far away from the tank as possible. The measured 1.274 MeV gamma ray flux rates were approximately the same in all three measurement locations.

For the computer calculations, the tank was modeled as an empty cylinder having a wall thickness of 0.88 cm, and the tank bottom was modeled as having a thickness of 1.36 cm. The pipe leading from the bottom of the tank was modeled as an empty cylinder having a wall thickness of 0.55 cm. It was determined that the measurements were consistent with the calculated flux rates for several plausible source distributions. Therefore, three general distributions were analyzed. These distributions assumed that 1) most, 2) some, or 3) none of the fuel was located inside the tank (Fuel Location 1). Based on ion chamber readings, most of the remaining fuel was assumed to be located in the pipe leading from the bottom of the tank (Fuel Locations 2 and 4). A small percentage of the fuel was assumed to be adhered to the tank wall (Fuel Location 3). For each of the three general distributions, the percentages of fuel in the cubicle components were varied until the calculated flux rate ratios for each measurement location were similar to the measured flux rate ratios (see Fig. 4.5 for the resulting amounts). The resulting fuel



Fuel Location	Modeled Fuel Distribution		
	1	2	3
1	99.0%	54.3%	0%
2	1.0%	10.5%	14.8%
3	0%	4.7%	0.8%
4	0%	30.5%	84.3%

FIGURE 4.5 Measurement Locations in Makeup Tank Cubicle

estimate for each of the three general distributions was then calculated using the method described in Section 3.3.

Based on the computer calculations, it was determined that the total fuel estimate for the cubicle was highly dependent on the general source distribution selected, ranging from 51 g for Distribution (3) to 780 g for Distribution (1). This relatively large range is attributable primarily to the large size of the tank with respect to the size of the cubicle. Because the best fit between the calculated and measured gamma flux rates was obtained for source Distribution (2), and because this distribution was the most plausible, the resulting fuel estimate (100 g) is considered the best estimate. The calculated and measured gamma flux rates for this source distribution are compared in Table 4.5.

Because the fuel estimate is highly dependent on the fraction of fuel assumed to be in the tank, bounds on the fuel estimate of 100 g take into account the range of fuel estimates calculated for the different plausible geometries in addition to the precision errors. Using the methodology described in Section 3.4, the associated minimum and maximum estimates are 30 g and 1300 g, respectively.

In contrast to the PNL approach, the licensee estimated the quantity of fuel in the Makeup Tank cubicle by measuring the ^{144}Ce photopeak at several locations using a NaI detector. Using this method, the estimated fuel quantity was 310 g, with an associated error of +/- 31 g. This estimate is higher than the PNL best estimate of 100 g, although the difference is not

TABLE 4.5 Calculated and Measured 1.274 MeV Gamma Flux Rates in Makeup Tank Cubicle

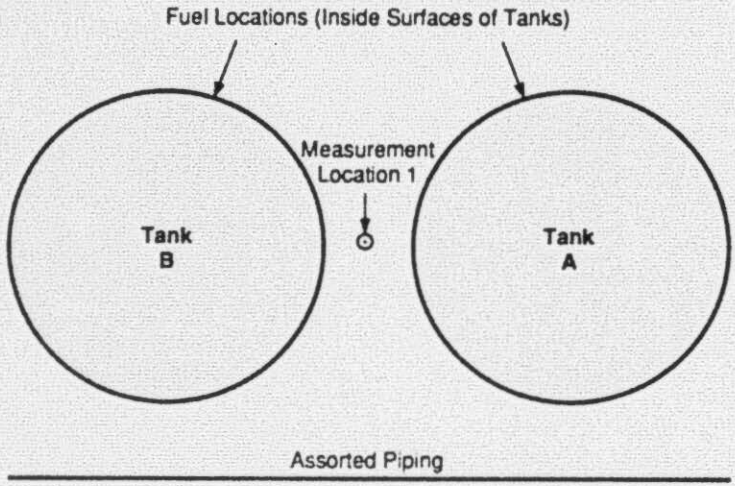
Measurement Location in the Makeup Tank Cubicle	Gamma Flux Rate (Relative to Measurement 5)	
	Measured	Calculated
5	1.00	1.00
7	1.13	1.13
8	1.00	1.00

statistically significant considering the wide range associated with the PNL estimate. Also, the licensee's error estimate was inappropriately small because it did not include consideration of either analytical error or the uncertainty associated with the ^{144}Ce-to-fuel ratio. The analytical error could be especially significant because of the technique used by the licensee to interpret the measurement data.

In their analysis, the licensee assumed that the entire gamma flux rate measured at each location resulted from photons emitted from a single component. For each of the ten gamma spectroscopy measurements, they calculated the amount of fuel that would be present in each identified component if this were the case. Therefore, for each component, there existed ten separate estimates of the fuel quantity in that component. It was known that the actual amount of fuel in each component had to be equal to or less than the minimum of the ten separate estimates (because if more than the minimum amount of fuel was in the component, then the gamma flux rate at one or more of the measurement locations would have been higher than that measured). The licensee's total fuel estimate for the cubicle was obtained by summing the minimum fuel estimate for each component. This approach results in an overestimate of the total fuel quantity because it is unrealistic to assume that the measurement signal at a specific location is attributable entirely to fuel in only one component. The licensee acknowledged that this analysis approach leads to an overestimate. In contrast, the approach used by PNL assumes that the signal at each measurement location could be attributable to fuel in more than one component. This method does not inherently overestimate the fuel quantity.

4.2.5 Submerged Demineralizer System Monitor Tanks

Figure 4.6 illustrates the components and measurement locations near the Submerged Demineralizer System (SDS) Monitor Tanks. Only two measurements were obtained near the tanks. Measurement 1 was obtained in between the two tanks approximately 10 cm above the floor. Measurement 2 was obtained approximately 500 cm away from the tanks and 132 cm above the floor. The 1.274 MeV gamma flux rate measured at Location 1 was approximately five times higher than the flux rate at Location 2.



Modeled Fuel Distribution

Tank A: 50%
Tank B: 50%

FIGURE 4.6 Measurement Locations in Submerged Demineralizer System Monitor Tanks

For the computer calculations, a range in tank wall thicknesses was necessary because the exact thicknesses could not be determined. Therefore, the tanks were modeled as cylinders each having a wall thickness of either 0.5 cm or 2.54 cm. It was assumed that the source was contained entirely inside both tanks in equal amounts. The quantity of water in the tanks was also unknown, so calculations were performed assuming the tank was either filled with water or empty. Also, there existed numerous pipes between the tanks and Measurement Location 2. Because the effective shielding thickness from these pipes was unknown, calculations were performed assuming the attenuation from the pipes was equivalent to the attenuation from either 0 or 1 cm of iron.

It was determined that the measurements were consistent with the calculated fluxes if the source was adhered to the walls of the tanks to a height of approximately 400 cm to 700 cm, the specific height depending on the assumptions regarding tank wall thickness, amount of water in the tank and effective pipe thickness. For each set of assumptions, it was possible to define a source that resulted in a match between the calculated and measured gamma flux rates (Table 4.6).

The estimated fuel quantity ranged from 2.6 g to 18 g depending on the assumptions used. A best estimate of 10 g is adopted, which corresponds to assumptions that 1) the thickness of the tank walls was 2.54 cm, 2) the tanks were empty, and 3) the effective pipe shielding for Measurement Location 2 was 1 cm. These assumptions were thought to be the most realistic. The fuel estimates derived using variations on these assumptions are used to calculate the minimum and maximum fuel estimates. Using the methodology described in Section 3.4, these are 1.2 g and 30 g, respectively.

TABLE 4.6 Calculated and Measured 1.274 MeV Gamma Flux Rates in Submerged Demineralizer System Monitor Tanks

Measurement Location Near the SDS Monitor Tanks Cubicle	Gamma Flux Rate (Relative to Measurement 1)	
	Measured	Calculated
1	1.00	1.00
2	0.21	0.21

The licensee did not estimate the quantity of fuel remaining in the Submerged Demineralizer System Monitor Tanks. They did, however, estimate that a maximum of 1 kg of fuel remained in the Submerged Demineralizer System Monitor Tanks and Spent Fuel Pool "B" combined. This estimate was not based on measurements. The licensee's Defueling Completion Report states that "due to the extensive filtration [of effluent water] it is conservatively estimated that the residual fuel in Spent Fuel Pool 'B' and the monitor tanks is expected to be much less than 1 kg." Therefore, it is not possible to directly compare the GPU and PNL estimates.

4.3 REACTOR BUILDING MEASUREMENTS

Four separate incore instrument guide tube (IIGT) bundles were measured in the reactor building. The measurements were obtained on the 347-foot elevation of the reactor building near the "B" D-ring (Fig. 4.7). The bundles were suspended vertically such that the bottom of each bundle was from zero to a few centimeters above the floor. The detector was located several meters away from the bundles approximately 30 cm above the floor. To perform each measurement, a bundle was lowered into position at least 250 cm from the detector (the actual distance depended on the expected dose rate) and the distance from the bundle to the detector was recorded. The bundle was counted for a period sufficient to produce a significant 1.274 MeV photopeak on the gamma spectrum. In most cases, a calibration source measurement was obtained with the bundle in place to determine the counting efficiency in the same background environment that existed for the bundle measurement. Following the measurement, the bundle was placed back in storage, a new bundle was placed into position, and the process was repeated. Several times during the course of the measurements an area background measurement was obtained when no bundle was present.

Each IIGT bundle measured consisted of either one or two tubes. Dimensions of the tubes, including the shielding surrounding the tubes, were taken from the GPU draft Calculation Number 4800-3211-90-023 describing their own measurement analysis. The inner diameter of each tube was 1.84 cm. Each tube was surrounded by a 4.79 cm thick layer of iron. Details on each of the IIGT bundle measurements are provided below.

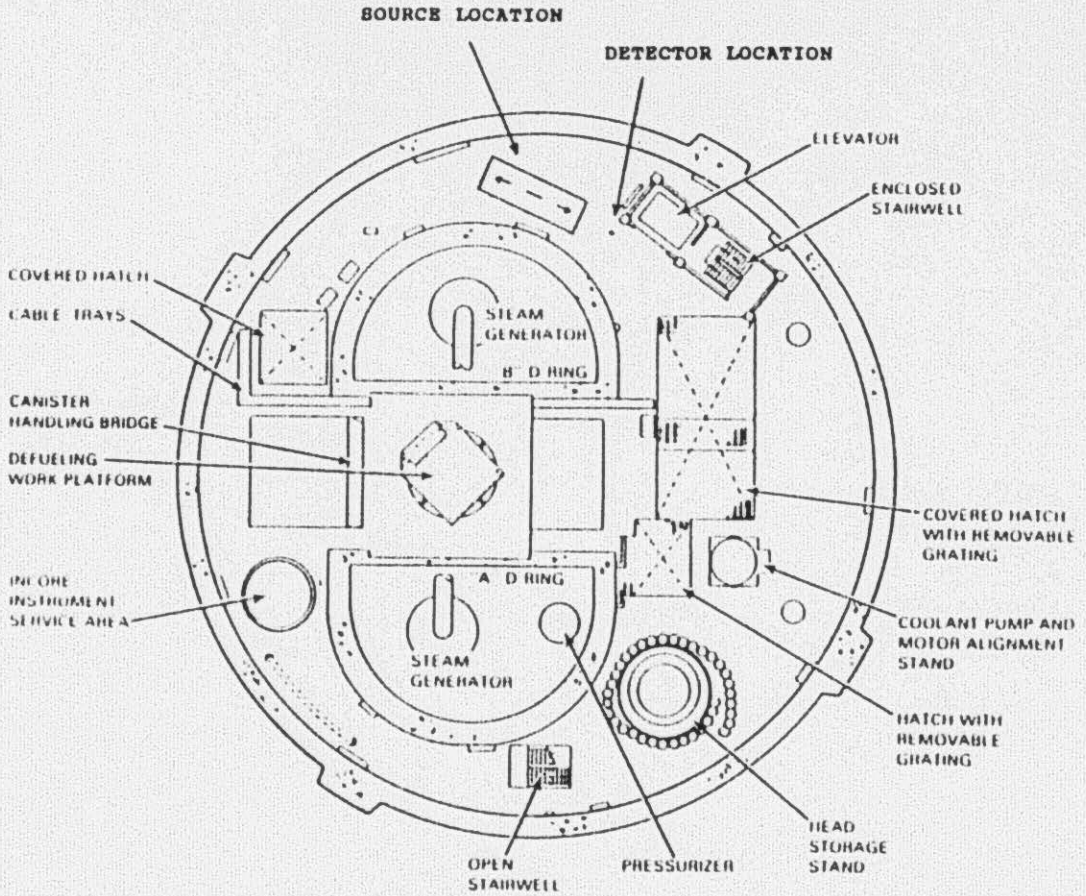


FIGURE 4.7 Location of Incore Instrument Guide Tube Measurements in 347-Foot Elevation of Reactor Building

The licensee's approach to estimating the quantity of fuel in the IIGT bundles was similar, in general, to the PNL approach. Similar to PNL, the licensee made a single measurement of each bundle using a germanium detector. However, the measurement geometries were different in that the licensee rested the bundles horizontally on the floor rather than suspending them vertically above the floor. This often resulted in a difficult modeling geometry. Another difference was that the licensee measured the ^{144}Ce photopeak rather than the ^{154}Eu photopeak. (^{144}Ce was more prominent during the licensee's measurements than it was during the PNL measurements.) Also, the licensee used the Microshield code, when possible, to correlate measured count rate to the quantity of fuel analog in the bundle. The Microshield code is similar to the PNL code in that it is straightforward for simple geometries.

4.3.1 Incore Instrument Guide Tube Bundle 5

Incore Instrument Guide Tube (IIGT) Bundle 5 consisted of a single tube 106 cm in length. The tube was suspended vertically at a distance of approximately 560 cm from the detector.

The WISE code was used to determine the quantity of ^{154}Eu in the tube that would result in a 1.274 MeV gamma flux rate equal to the flux rate measured. For the calculations, the tube was modeled as a cylinder having the dimensions described previously. It was assumed that the tube was completely filled with a uranium compound having a density of 5.0 g/cm^3.

Based on the computer calculations, the estimated fuel quantity in IIGT Bundle 5 was 680 g. Because the dimensions of the tube were relatively well known and the counting geometry was satisfactory, the modeling uncertainty associated with the computer calculation is expected to be small (less than 20%). Using the methodology described in Section 3.4, the bounds on the central estimate of 680 g are 320 g and 1000 g.

The licensee's estimate of the quantity of fuel in IIGT Bundle 5 was 1000 g. This estimate is slightly higher than the PNL estimate, although it is within the PNL error bounds. As discussed previously, the licensee's measurement and analysis method was similar to the PNL method, although the licensee used ^{144}Ce rather than ^{154}Eu as the fuel analog. However, the

licensee's detector was much closer to the IIGT bundle, which results in greater modeling uncertainties.

4.3.2 Incore Instrument Guide Tube Bundle 7

Incore Instrument Guide Tube (IIGT) Bundle 7 consisted of two parallel tubes, one 119 cm in length and the other 107 cm in length. The tubes were suspended vertically from the floor. The shorter tube was approximately 470 cm from the detector and the longer tube was approximately 440 cm from the detector. There was a direct line of sight between each tube and the detector, i.e., one tube did not shield the other.

For the computer calculations, the tubes were modeled as cylinders having the dimensions described previously. It was assumed that the tubes were completely filled with a uranium compound having a density of 5.0 g/cm^3. It was further assumed that both tubes contained equal amounts of fuel per unit volume.

Based on the computer calculations, the estimated fuel quantity in IIGT Bundle 7 was 300 g. Because the dimensions of the tubes were relatively well known and the counting geometry was satisfactory, the modeling uncertainty associated with the computer calculation is expected to be small (less than 20%). Using the methodology described in Section 3.4, the associated minimum and maximum estimates are 140 g and 510 g, respectively.

The licensee's estimate of the quantity of fuel in IIGT Bundle 7 was 200 g. This estimate is slightly lower than the PNL estimate, although it is within the PNL error bounds. As discussed previously, the licensee's measurement and analysis method was similar to the PNL method, although the licensee used ^{144}Ce rather than ^{154}Eu as the fuel analog. However, the licensee's modeling effort was complicated by the fact that one tube rested behind the other for their measurements (the tubes were lying horizontally on the floor). The licensee assumed that, because the tubes were angled slightly, the gammas emanating from the back tube were not impeded in the path to the detector. It cannot be determined from the licensee's data whether this was actually the case. This may account for the licensee's estimate being lower than the PNL estimate, although, as stated previously, the difference is not statistically significant.

4.3.3 Incore Instrument Guide Tube Bundle 10

Incore Instrument Guide Tube (IIGT) Bundle 10 consisted of two parallel tubes, one 114 cm in length and the other 109 cm in length. The tubes were suspended above the floor at an angle 20^{0} offset from the axis perpendicular to the floor. Each tube was approximately 260 cm from the detector. There was a direct line of sight between each tube and the detector.

For the computer calculations, the tubes were modeled as cylinders having the dimensions described previously. It was assumed that the tubes were completely filled with a uranium compound having a density of 5.0 g/cm^{3}. It was further assumed that both tubes contained equal amounts of fuel per unit volume.

Based on the computer calculations, the estimated fuel quantity in IIGT Bundle 10 was 280 g. Because the dimensions of the tubes were relatively well known and the counting geometry was satisfactory, the modeling uncertainty associated with the computer calculation is expected to be small (less than 20%). Using the methodology described in Section 3.4, the associated minimum and maximum estimates are 130 g and 470 g, respectively.

The licensee's estimate of the quantity of fuel in IIGT Bundle 10 was 2300 g. This estimate is significantly higher than the PNL estimate, even considering the bounds on the PNL estimate. The probable reason for the discrepancy is that, due to the positions of the tubes for the licensee's measurements, the Microshield computer code could not be used for the analysis. Instead, the QAD-UE computer code was used. This code is much more difficult to use than Microshield. In other cases where the licensee used this code (none of which are relevant to this report), the code clearly overestimated the amount of fuel present. This may account for some or all of the difference between the licensee's and PNL's estimates.

4.3.4 Incore Instrument Guide Tube Bundle 25

Incore Instrument Guide Tube (IIGT) Bundle 25 consisted of two parallel tubes, each approximately 122 cm in length. The tubes were suspended above the floor at an angle 20^{0} offset from the axis perpendicular to the floor. Each tube was approximately 320 cm from the detector. There was a direct line of sight between each tube and the detector.

For the computer calculations, the tubes were modeled as cylinders having the dimensions described previously. It was assumed that the tubes were completely filled with a uranium compound having a density of 5.0 g/cm^3. It was further assumed that both tubes contained equal amounts of fuel.

Based on the computer calculations, the estimated fuel quantity in IIGT Bundle 25 was 230 g. Because the dimensions of the tubes were relatively well known and the counting geometry was satisfactory, the modeling uncertainty associated with the computer calculation is expected to be small (less than 20%). Using the methodology described in Section 3.4, the associated minimum and maximum estimates are 110 g and 390 g, respectively.

The licensee's estimate of the quantity of fuel in IIGT Bundle 25 was 400 g. This estimate is higher than the PNL estimate, although it is essentially within the PNL error bounds. Similar to the IIGT Bundle 7 modeling effort by the licensee, the IIGT Bundle 25 modeling effort was complicated by the fact that one tube rested behind the other for their measurements (the tubes were lying flat on the floor). In this case, however, the licensee assumed that the back tube was completely blocked by the front tube, which would result in significant attenuation of the photons emitted by fuel present in the back tube. This leads to the potential for analysis errors because 1) the back tube may not have been completely blocked by the front tube, and 2) it is possible that one tube contained significantly more or less fuel than the other. For the PNL analysis, the measurement geometry was such that it did not matter whether or not the tubes contained equal amounts of fuel; the resulting fuel estimate for both tubes together was mostly independent of the source distribution within the tubes. These factors may account for the discrepancy between the licensee's and PNL's estimates, although the difference is not statistically significant.

5.0 CONCLUSIONS

The results of the PNL measurements indicated that the licensee's estimates of fuel quantities were generally conservative. In all but one location, the licensee's estimates of the fuel quantities were higher than the PNL central estimates. The single exception was Incore Instrument Guide Tube Bundle 7; however, the licensee's estimate was within the range estimated by PNL. For seven of the nine locations, the licensee's estimates were within the PNL minimum and maximum estimates. In the other two locations, the licensee's estimates were significantly higher than the PNL estimate. Based on these results it is concluded that for the locations measured, the licensee did not underestimate the fuel quantities; in all cases, their estimates were either within the PNL error bounds or greater than the PNL maximum estimate.

The conservatism associated with the GPU estimates is attributable to the difference between the GPU and PNL measurement and analysis approaches. The PNL measurement program attempted to determine an accurate value for the remaining fuel. In contrast, the licensee attempted to ensure that the quantity of fuel remaining in the facility was not underestimated. In addition, the methods and instrumentation used by PNL differed from that used by the licensee. The primary difference between the approaches is that PNL measured ^{154}Eu using an intrinsic germanium detector whereas the licensee typically measured ^{144}Ce using a NaI detector. In several cases, the PNL method resulted in statistically significant photopeaks, while the licensee's method did not. In these cases the licensee used calculations of the minimum detectable level (MDL) in order to derive their fuel estimates. By definition, MDL calculations result in conservatively high, and often unrealistic, fuel quantity estimates. In one case in which the licensee's estimate was significantly greater than the PNL maximum estimate, the discrepancy was attributable to this approach. In the other case, errors associated with a computer model used by the licensee were the most likely cause of the significant overestimate.

6.0 REFERENCES

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