50-320

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 4, 1994

Mr. T. Gary Broughton
Vice President
GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057-0191

Dear Mr. Broughton:

SUBJECT: POST-DEFUELING SURVEY REPORT REVIEWS

The Nuclear Regulatory Commission staff has performed a review of your Defueling Completion Report and Post-Defueling Monitored Storage Safety Analysis Report. The measurements you made and your subsequent analyses were generally well done and conservative. In several cases we believe that the uncertainty of the measurement is greater than what you had estimated. However, in no case have we found a reason to change the estimate of the fuel quantity remaining at TMI-2.

The staff in conducting the review used the information contained in 10 Post-Defueling Survey Reports which were developed by our consultant, Battelle Pacific Northwest Laboratories. The following Post-Defueling Survey Report reviews are attached for your information.

A & B Steam Generators (OTSGs)
Auxiliary and Fuel Handling Buildings
Letdown Coolers
Plenum Structure
Pressurizer
Reactor Building Basement
Reactor Building Miscellaneous Components
Reactor Coolant System
Reactor Vessel
Reactor Vessel Head.

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A copy of these Post-Defueling Survey Report reviews will be placed in the docket.

Sincerely,

Lee H. Thonus, Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Docket No. 50-320

Attachment:
As stated

cc: See next page

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Sincerely,

Original signed by

Lee H. Thonus, Project Manager
Non-Power Reactors and Decommissioning
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T. G. Broughton
GPU Nuclear Corporation Unit No. 2

Three Mile Island Nuclear Station
Docket No. 50-320

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June 28, 1994

Mr. Lee Thonus
U.S. Nuclear Regulatory Commission
Three Mile Island
Middletown, PA 17057

Dear Mr. Thonus:

PNL completed reviews of ten Post-Defueling Survey Reports (PDSRs) submitted by the TMI-2 licensee. We submitted draft copies of these reviews to you and Dr. Masnik previously for your review. Since your reviews of these writeups did not result in any major changes, we are now prepared to submit our reviews in final form. Enclosed please find the following ten PDSR reviews:

A & B Steam Generators (OTSGs)
Auxiliary and Fuel Handling Buildings
Letdown Coolers
Plenum Structure
Pressurizer
Reactor Building Basement
Reactor Building Miscellaneous Components
Reactor Coolant System
Reactor Vessel
Reactor Vessel Head

Several general conclusions have become apparent from our study of the licensee's PDSRs. We feel that the measurements and analyses were generally well done. The working environment for these studies was extremely harsh, and in many cases it dictated a limit on the type of measurement that could be performed or the thoroughness of a given measurement. The licensee was able to work within these limitations to arrive at a credible estimate of the fuel remaining in the TMI-2 plant. We found the analyses to be thorough and well-documented. For most of the efforts we agreed with the methods chosen by the licensee; in some cases we have had minor disagreements with the technique or method of analysis, and in several instances we disagreed with the uncertainty attached to a fuel estimate. In no case, however, did our review find a reason to change the estimate of the quantity of fuel remaining in or removed from the TMI-2 plant. We concur with the licensee's contention that ALARA considerations make it unwise to perform any additional study, and we agree that the fuel remaining in the individual locations are unlikely to move out of those locations.

We have enjoyed working on these reviews and we hope that you will feel free to call us at the above number if you have any questions or comments.

Sincerely,

Robert I. Scherpelz
Technical Group Leader
Radiation Measurements and Assessments
HEALTH PROTECTION DEPARTMENT

Rebekah Harty
Senior Research Scientist
Instrumentation and Dosimetry Development
HEALTH PROTECTION DEPARTMENT

RIS/blh

Enclosures

cc: Dr. Michael Masnik

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE 'A' and 'B' ONCE-THROUGH STEAM GENERATORS

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1989) discusses the process for estimating the amount of UO_2 remaining in the two Once-Through Steam Generators (OTSGs). The quantity of fuel estimated to be present in these components is 55.5 kg with an uncertainty of 15%.

For this study, several different measurement techniques were used. For the OTSG lower heads and associated J-legs, small Geiger-Muller (GM) tubes were enclosed in polyethylene tubing to measure the gamma exposure rates. A computer code was then used to model the fuel deposition and estimate the quantity that would cause the recorded signal. Gross gamma measurements with a small GM tube were also used to quantify fuel remaining in the tube bundles. Computer modeling was again used to determine the fuel.

For the OTSG upper tube sheets, copper foils were exposed, and neutrons emitted by the fuel created radioactive copper in the foils. The emitted radioactivity could then be used to determine the neutron flux at the copper foil, and the fluxes could be used in conjunction with computer modeling to determine the fuel mass in the region.

MEASUREMENT METHODOLOGY

Lower OTSG Heads

For quantifying the fuel remaining in the OTSG lower head regions, detector strings were made by placing small GM detectors inside a polyethylene tube. The tube was inserted into the lower head and exposure rates were measured at a number of positions in the lower head. Exposure rates were measured at 10 locations at one-foot intervals over a 12-foot distance, and the measured rates ranged from a low of 0.8 R/h to 17.5 R/h in the 'A' OTSG. In the 'B' OTSG, the exposure rates were measured at four locations separated by one-foot intervals, and these measurements ranged from 7.8 to 17.0 R/h. During the measurements, video cameras were used to verify the positioning of the detectors.

J-Legs

After measuring exposure rates in the lower head regions, the measurement strings were pushed an additional fifteen feet to measure exposure rates in the associated J-legs (the large pipes that carry water from the bottom of the OTSGs into the Reactor Coolant pumps; two J-legs are connected to each OTSG's lower head). The measured exposure rates ranged up to 76 R/h in the 'A' legs and up to 700 R/h in the 'B' legs.

Tube Bundles

A measurement string containing GM tubes was assembled to measure gammas emitted by clumps of fuel debris lodged in the tube region. The measurement string consisted of six small GM detectors spaced at one-foot intervals inside a polyethylene tube. The string was deployed in 52 different OTSG tubes, carefully chosen to ensure the detection of any fuel blockages that may be present in the 9.5-foot-diameter tube bundle. For each OTSG tube, measurements of gamma exposure rates were taken at five-foot intervals over the entire 56-foot tube length. The measured exposure rates could then be translated into fuel quantities using computer modeling.

OTSG Upper Tube Sheets

Copper foils were exposed in the upper tube sheets of the 'A' and 'B' OTSGs to measure the neutron fluxes and thus determine the fuel remaining in the upper tube sheet regions. Four foils were exposed at different positions above the 'B' upper tube sheet and two were exposed in 'A'. An additional foil was exposed elsewhere in the Reactor Building (RB) to give a background reading. In order to properly account for the effect of neutron scattering and neutron loss, an AmBe source was placed in the 'A' upper tube sheet region to provide neutrons for calibration purposes during the exposure of one of the 'A' foils.

After the foils were exposed in the OTSGs, they were placed in a low-level coincidence counting system to measure the β^- particles emitted by the radioactive copper atoms created by neutron activation. The measured radioactivity was used to determine the neutron fluxes exposing the foils.

Comparison of the first foil exposed in 'A' to the background foil exposed in the RB showed no net neutron signal, indicating that the quantity of deposited fuel was too low for this type of measurement. Gamma exposure rates measured above the 'A' and 'B' tube sheets were thus used to arrive at the estimate of fuel remaining in the 'A' OTSG upper tube sheet.

ANALYSIS METHODOLOGY

Lower OTSG Heads and J-Legs

The measured exposure rates were used to estimate the quantity of fuel remaining in these regions by computer modeling with the Microshield computer code. The computer calculations modeled the fuel debris deposits based on visual inspections of the debris. The modeling estimated the exposure rates that would result from a given quantity of fuel deposited in the regions, and the actual quantities of fuel calculated by scaling according to the measured dose rates. The fuel masses considered to be the estimate of record were:

'A' OTSG Lower Head	0.29 kg UO_2;
'B' OTSG Lower Head	0.46 kg UO_2;
'A' J-legs:	0.67 kg UO_2; and
'B' J-legs:	5.79 kg UO_2.

Copper foils had also been used with these GM measurements, and the quantities estimated by these means were compared to the estimates of record. The foils in the lower head regions did not measure significant neutron fluxes, but the foils in the J-legs measured fluxes that resulted in fuel estimates of 5.36 kg for the 'A' J-legs and 5.14 kg for the 'B' J-legs. The agreement between the gamma measurements and the copper foil measurements were good for the 'B' J-legs, but not for the 'A' J-legs.

Tube Bundles

The study of fuel in the tube bundles initially assumed that the fuel deposits consisted of 1-inch long clumps that plugged a tube. The Microshield code was used to model gamma emissions from these blockages and use the results of this modeling to plan the placement of the detector strings. The modeling concluded that each string could cover an eight-inch effective horizontal radius, so this effective radius led to the choice of 52 tubes for the measurements.

For analysis of the measured exposure rates, Microshield calculations were performed in which a one-inch long debris plug was placed at a position displaced 5.5 inches vertically and eight inches horizontally from the detector. This position resulted in a conservative estimate. The model assumed that the debris plug was similar in density and fuel concentration to debris material in the 'B' OTSG upper tube sheet. The measurement data were then entered into a spreadsheet for analysis and vertical exposure rate profiles were plotted for the tube bundle regions. These profiles showed that no significant radiation sources from fuel blockages were detected.

Next these exposure rates were adjusted for background radiation signals. The primary components of the background signal were:

1) cosmic rays;
2) contamination outside the OTSG;
3) contamination on the outside of the polyethylene tube holding the GM detectors;
4) ^{137}Cs activity on the inside surface of the OTSG tubes; and
5) ^{137}Cs contamination in the water in the lower part of the tubes.

When the background signals were subtracted, the measurement data showed no significant positive readings for fuel in the tube bundles. Thus a lower-limit-of-detection (LLD) analysis was performed on the measurement data, and this analysis resulted in estimates of 1.7 kg UO_2 in the 'A' OTSG tube bundle, and 9.1 kg UO_2 in the 'B' tube bundle. The LLD analysis gave a higher fuel mass for 'B' because of the higher gamma

background levels in 'B'. The uncertainties associated with these estimates were both ±48%.

Upper Tube Sheets

Before the copper foil measurements were made, two other methods had been used for estimating fuel remaining in these regions: video inspection and gross gamma counting. The video inspection gave an estimate of 28.9 kg fuel, and the gross gamma measurements gave an estimate of 53.9 kg for the sum of both OTSGs. The copper foil was seen as a means of increasing the accuracy, and its results were meant to be used for the estimate of record.

The four foils exposed in the 'B' OTSGs were counted in the coincidence system, and since two of the foils exhibited low activation levels, only two foils were actually used to determine neutron fluxes. The foil exposed in the RB gave a background rate that was subtracted from the 'B' foils' signals. One foil exposed in 'A' was activated by neutrons emitted from an AmBe source, and this was also the calibration for the 'B' foils. The calibration was necessary to know what quantity of fuel would produce a given flux at the foil. This analysis led to an estimate of 36 kg ±6.5 kg (18%) of fuel in the 'B' OTSG upper tube sheet.

The foil exposed in the 'A' upper tube sheet (with no AmBe source) gave a zero reading after the background signal was subtracted. Thus the results of the gross gamma measurements were used to estimate the fuel in 'A'. The exposure rate measured at one foot above the 'B' upper tube sheet was 126 R/h, and at two feet above the 'A' tube sheet it was 3 R/h. This 'A' reading was normalized to a one-foot position, based on Microshield calculations, to 5 R/h, and the ratio of 5/126 was multiplied by 36 kg to arrive at a fuel estimate of 1.4 kg UO_2 for the 'A' OTSG upper tube sheet. The uncertainty assigned to this measurement is 18%, which is identical to the uncertainty assigned to the 'B' estimate.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

Lower OTSG Heads and J-Legs

The methods of gross gamma measurements and computer modeling used to estimate the fuel in these regions is similar to methods used in a number of other PDSRs, and it appears to be properly applied here. It is unfortunate that the measurement locations in the 'B' regions had fewer data points measured than in the 'A' locations (only four in the 'B' lower head versus 10 in the 'A' lower head; only five in the 'B' J-leg versus 11 in the other three J-legs), because 'B' contained more fuel; the best quality data should be taken where there is the highest likelihood of fuel. However, the number of data points were sufficient to arrive at the estimate of record, and the uncertainty bounds of 47 to 49% are sufficiently large.

Tube Bundles

It is unfortunate that so many individual measurements were made in the tubes and that the subsequent data analysis showed that the signals were not significantly above background. The measurements showed, however, that there are no significant fuel debris blockages in the tube region. The measurements also provided the data used to determine minimum detectable levels that were used for the estimates of record. These estimates appear to be properly derived and have sufficiently large uncertainty bounds.

Upper Tube Sheets

The method of neutron activation for measuring fuel deposits has one advantage over gamma measurements; since there are very few neutron-emitting background sources, any detected neutrons have a good likelihood of coming from the fuel. The disadvantage of neutron measurement is that the neutron emission rate from a kilogram of fuel is very low, so the sensitivity of this technique is poor. In this study, the 1.4 kg of UO_2 concluded to be in the 'A' upper tube sheet region did not produce enough activation in the copper foil to give a signal above background, so the foils exposed there gave no estimate of fuel. In the 'B' region, 36 kg UO_2 were identified, and two foils gave readings sufficiently above background to arrive at this estimate.

One problem with the foil activation study is the use of a small number of measurements. Four foils were exposed in the 'B' upper tube sheet region, but only two gave meaningful data. Since the tube sheet is over nine feet in diameter, the analysis must rely heavily on an assumption of the even distribution of fuel in the tube sheet. The 36 kg value is probably not too far off the mark, and confirming evidence is given by the visual inspection which gave a value a bit lower. However, the stated 18% uncertainty bound is too small to account for the assumption made in the fuel distribution.

The study used an AmBe source as a known source of neutrons to account for the neutron scattering and neutron loss effects of neutrons emitted by the fuel. However, the geometry of the calibration is a point source and the geometry of the fuel distribution is an area source, so the calibration uncertainty must be higher. Also, the distribution of energies of neutrons emitted by the AmBe source is higher than the energies of neutrons emitted by the fuel, so activation of the foils will not be identical for the calibration case versus the fuel measurement case. Again, a higher uncertainty band should be stated to account for this.

The measurement of fuel in the 'A' upper tube sheet was done by gross gamma measurements, using analogy to the 'B' measurements. The analogy relies on the assumption that fuel distribution in the 'A' tube sheet is identical to the distribution assumed for the 'B' tube sheet and that background effects are identical. These assumptions should lead to larger uncertainties.

SUMMARY AND CONCLUSIONS

The following table summarizes the estimate of record for UO_2 remaining in the two OTSGs.

	'A' OTSG	'B' OTSG
Upper Tube Sheet	1.4 kg ±21%	36 kg ±18%
Tube Bundle	1.7 kg ±48%	9.1 kg ±48%
Lower Head	0.29 kg ±48%	0.46 kg ±48%
RCP-1 J-Leg	1.4 kg ±48%	1.5 kg ±47%
RCP-2 J-Leg	1.4 kg ±48%	4.3 kg ±49%
TOTAL	4.1 kg ±22%	51.4 kg ±16%

The techniques chosen for fuel estimation were generally appropriate, given the difficult environment and the necessity for ALARA considerations. The biggest shortcoming that we see is the small uncertainty assigned to the neutron activation foil measurements in the 'B' upper tube sheets, and the way that this small uncertainty (assigned to the largest component of the fuel estimate) propagated through the entire analysis. The foil measurements relied on exposure of only two foils, requiring assumptions about fuel distribution, and used a calibration system with a higher neutron energy distribution and a point calibration geometry. Thus we feel that the uncertainty for the upper tube sheet estimates should be on the order of 50%, and the uncertainties for the entire estimates of record should be about the same size.

The values chosen for the estimates of record, on the other hand, appear to be sound. Where available, other measurement methods tended to confirm the values chosen, lending credibility to the analyses.

REFERENCES

GPU Nuclear. March 14, 1990. TMI-2 Post-Defueling Survey Report for the 'A' and 'B' Once-through Steam Generators. GPU Communication #4410-90-L-0019/0356.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE AUXILIARY AND FUEL HANDLING BUILDINGS

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU 1991) for the Auxiliary and Fuel Handling Buildings (AFHB) discusses the procedures used to estimate the amount of UO_2 remaining in the Reactor Vessel Head Assembly. The AFHBs are divided into 131 individual areas. The Auxiliary Building contains the tanks, pumps, piping, and equipment for processing and storing water for the Reactor and primary cooling system and for the treatment of radioactive wastes. The Fuel Handling Building contains the fuel handling and storage equipment and a limited number of tanks, filters, pumps and coolers.

To perform the analysis of the fuel content remaining in the AFHB, the AFHB was considered by systems rather than by individual areas. There are 26 systems in the AFHB. An engineering analysis was performed to determine the areas where the flow paths were during the accident and the areas where the total quantity of UO_2 in the system was believed to be insignificant. Decisions to measure the fuel remaining in each specific area was based on the analysis of the system. Four different types of measurements were made on 38 of the areas as discussed in the next section.

MEASUREMENT METHODOLOGY

The selection for analysis of those areas of the AFHB that could contain fuel deposits was based on the Rogovin Report (Rogovin 1980). Measurements were made on 38 of the areas, using the techniques described above. The remaining areas were analyzed using an engineering analysis as discussed further in the next section.

The following four different measurement methods were used:

1) gamma scintillation using a sodium iodide detector (NaI(Tl));
2) gamma scintillation using a high-purity germanium (HPGe) spectrometer;
3) gross gamma survey; and
4) individual sample analysis.

Gamma Spectroscopy - NaI(Tl)

Twenty-two areas were surveyed using a NaI(Tl) detector with a thallium activated scintillation crystal and gamma spectroscopy. This device measured fuel by detecting the 2.19 MeV gamma ray of ^{144}Ce, which has been shown to be an analog for the UO_2 fuel (Babel 1988). The problems of a high background gamma dose were resolved by using the smallest detector that could see the 2.19 MeV gamma (1/2-inch diameter by 3/4-

inch long crystal) and the maximum practical amount of shielding (35 to 80 pounds of tungsten shielding) assembled in various configurations.

High-Purity Germanium Spectrometer

Because of the decay of ^{144}Ce (half-life of 284 days), the NaI(Tl) detector was unable to distinguish its small peak from the background, so it was necessary to use a HPGe detector with much better resolution. Six areas were measured using the HPGe detector which was housed in a two-inch thick cylindrical lead shield. The HPGe detector used both ^{144}Ce and ^{154}Eu as analogs for the residual fuel, and were calibrated using standard and ^{144}Ce and ^{154}Eu sources.

Sample Analysis

The analyses for four areas were based on individual samples of resin media that were analyzed by a radiochemical laboratory at Oak Ridge National Laboratory or at a IMI laboratory.

Gross Gamma Surveys

Gross gamma directional surveys were used in six areas to determine the quantity of fuel by using a portable gamma survey instrument with a directional probe. A correlation between gross gamma-ray output and the quantity of fuel present was based on measurements from a sample of reactor core debris. The field measurements that were made were compared with a computer model of the exposure rates for a given volume of fuel under the appropriate geometry. The results were used to determine the quantity of fuel present in any area or component.

ANALYSIS METHODOLOGY

The analysis of each of the systems for fuel content was performed before, or in conjunction with, measurements of specific individual areas. There are 26 systems in the AFHB. An engineering analysis was performed to determine the areas where the flow paths were during the accident and the areas where the total quantity of UO_2 in the system was believed to be insignificant. Eight plant systems (69 individual areas) were evaluated for fuel content:

1) Makeup and Purification System (MU&P);
2) Liquid Radwaste Disposal System (WDL);
3) Solid Radwaste Disposal System (WDS);
4) Waste Disposal Gas System (WDG);
5) Spent Fuel Cooling System (SFC);
6) Nuclear Sampling System (SNS);
7) Submerged Demineralizer System (SDS); and
8) Defueling Water Cleanup System (DWCS).

The remaining systems were judged as either not requiring any special nuclear materials (SNM) assessment (43 individual areas) - based on an analysis

showing that there was no means to transport the fuel into these areas, (i.e., no piping connected to any of the processing systems that support operation of the Reactor Coolant System), or were analyzed as areas that were assessed to be free of fuel, but could contain quantities of fuel if their normal function had been misused (19 areas).

The determination that certain areas did not require an SNM assessment was based on the Rogovin report's analysis of the accident (Rogovin 1980). These 43 areas contain non-water processing equipment such as electrical switchgear, unit substations, HVAC blower and ducting, or they may have no equipment and just serve as an access corridor to service other areas. Because there was no means for transporting fuel into these areas, (in other words, no piping connected to any processing systems that supported operation of the Reactor Coolant System) there was no expectation that fuel would be present.

During the review of twelve of the systems, it was concluded that 19 areas did not contain fuel. Some of these areas are part of the SNS, SDS, Standby Pressure Control System, Miscellaneous Decay Heat Removal System, temporary cleaning facility, or the modified fuel handling bridge - which were constructed after the accident and installed as part of the TMI-2 Cleanup Program. Others were part of the six systems, Chemical Addition, DH Closed Cooling Water, Health Physics, Interim Closed Cooling Water, Nuclear Service Closed Cooling Water, and the Reactor Building Emergency Cooling Systems - that were originally installed when TMI-2 was constructed. The 19 areas were analyzed based on their vulnerability during the TMI-2 accident and their service history during the TMI-2 Cleanup Program. The determination was made that it is improbable that any remaining fuel not already analyzed and accounted for elsewhere exists in any of those areas.

The MU&P contains 18 areas - 12 of which were measured using either a NaI(Tl) or HPGe detector. The ^{144}Ce isotope was used as a tracer for the fuel. If the ^{144}Ce was not found, ^{154}Eu was used. In four cases neither ^{144}Ce or ^{154}Eu were identified, and a minimum detectable level (MDL) calculation was performed. In one case, (makeup pump 1C), a tungsten-shielded directional probe was used to obtain a gross gamma count. Gamma-ray spectrometry was initially used for the two areas AX114 and AX115 (MU&P demineralizer 1A and 1B respectively). However, the number of records was based on resin/fuel samples taken a year earlier. The analysis of the remaining areas was based on the service record of those areas and the fuel deposits measured in similar areas. The estimate of record for the MU&P is 2.81 kg UO_2 with an uncertainty of ± 27%. The MDL values were 0.60 kg. Uncertainties for specific areas ranged from ±25% to ±100%. The largest quantity of fuel (1.06 kg) was in the MU&P demineralizer 1A.

The WDL contains 29 areas - 20 of which were assayed for fuel. A NaI(Tl) detector was used for the assays of 17 of the 29 areas, and again the spectra for ^{144}Ce or ^{154}Eu were used to identify the presence of fuel. If a peak was not identifiable (as occurred in seven areas), a MDL calculation was performed by determining the gross counts in the region of interest for the Ce or Eu, and converting this number to a MDL value which was divided by the measured detector efficiency, calculated photon fluence per kg of fuel, and the count time. Gross gamma exposure rate measurements were performed on the Auxiliary

Building sump filters (AX024), and direct samples were obtained for the deborating demineralizers. Nine areas were not measured, instead an analysis of the fuel deposits was made based on measured fuel deposits in areas with similar flow origin and water processing history. The estimate of record for the WDL was 4.13 kg UO_2, with an uncertainty of ± 71%. The MDL values were 0.11 kg. Uncertainties for specific areas ranged from ±46% to ±104%. The largest quantity of fuel (3.5 kg) was located in the Reactor Coolant bleed holdup tanks B and C.

The function of the WDS prior to the accident was to store and transfer bead type resins, concentrated liquid wastes and reclaimed boric acid. This system is composed of two independent subsystems, the resin waste subsystem and the concentrated liquid waste subsystem. Two areas of the WDS were measured directly for fuel content. The first area - concentrated waste tank (AX218), was measured using an HPGe detector. The second area - concentrate liquid waste pump room (AX124), was measured using the tungsten-shielded directional probe to obtain a gross gamma survey. The remaining five areas were appraised based on their service history and fuel deposits measured in similar areas which were exposed to similar service conditions. The estimate of record for the solid radwaste disposal system was 0.01 kg UO_2 with an uncertainty of ±57% on the measured values. The MDL values were set to 0.0.

The remaining five systems include the DWCS, WDG, SNS, SFC, and the SDS. Specific areas of the SDS, and SNS systems were discussed previously. Only three areas from these five systems were formally assayed for fuel. The three areas include the model room (FH105), monitor tanks and sample sink (FH106), and the spent fuel pool A (FH109). All three areas were assayed for gross gamma counts using a tungsten-shielded directional probe. The remaining 14 areas were analyzed based on their service history and their similarity to other areas that were measured. The estimate of record for these systems was 3.80 kg UO_2. This was based on the fuel located in spent fuel pool A. All other areas contained fuel quantities less than 0.005 kg. The uncertainty on the fuel value in spent fuel pool A was +35%, -92%.

Approximately 10% of the process piping was assumed to be embedded in concrete walls and floors and was not included in the analysis of those areas of the AFHB where fuel was thought to be. The contribution for each of these systems was determined as summing up to 0.21 kg based on 10% of the fuel in the process piping that was measured.

REVIEW OF MEASUREMENT AND ANALYSIS METHODOLOGY

The methodology used to determine the quantity of fuel remaining in the AFHB is conventional and appropriately applied. A number of areas were not directly measured, but rather the estimate of fuel was based on their similarity to other areas or equipment that was measured. Although a thorough analysis of the fuel would have dictated that every area be measured, for ALARA purposes this would not have been appropriate, and thus the linking of similar areas was a good idea.

The method of calculating the error and adding the errors from separate measurements appears to be valid.

Appendix E (which discusses the AFHB areas that were analyzed to not have fuel) does not explain the absence of fuel in the FH007 access-area north, FH107 trash compactor, and the FH108 truck bay. However, the possibility of fuel appears very unlikely. The explanations for the remaining 16 areas appear to be appropriate, and the conclusion that none of these 19 areas contain fuel is not questioned.

SUMMARY AND CONCLUSIONS

The analysis methodology was found to be appropriate, in that a systems approach to the AFHB was developed, where the flow of fuel during the accident, and the movement of fuel during the cleanup process was used to determine which areas needed further analysis and measurement and which areas could be assumed to not contain fuel any longer. The measurement techniques were also found to be appropriately used. Thus, the fuel estimate of 11.5 kg UO_2 with a range from 4.2 to 15.8 kg UO_2 was found to be acceptable. A summary of the distribution of the fuel by area is shown below:

Spent Fuel Pool A (FH109)	3.8 kg
RC Bleed Holdup Tanks B&C (AX020)	3.5 kg
MU&P Demineralizer 1A (AX114)	1.1 kg
MU&P Demineralizer 1B (AX115)	0.1 kg
MU Suction Valves (FH001)	0.5 kg
Seal Return Coolers (AX112)	0.3 kg
Makeup Tank (AX116)	0.3 kg
RC Bleed Holdup Tank A (AX021)	0.3 kg
MU&P Valves (FH101)	0.3 kg
Makeup Pump 1A (AX007)	0.2 kg
Other Areas of AFHB (78)	1.1 kg
AFHB TOTAL	11.5 kg

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REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE LETDOWN COOLERS

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1989) discusses the process for estimating the amount of UO_2 remaining in the letdown coolers and associated piping in the Letdown Cooler Room. The quantity of fuel estimated to be present at this location is less than 3.7 kg with an uncertainty of 53%.

In this study, a collimated NaI detector was used to count 2.19 MeV photons from ^{144}Ce. In conjunction with the measurements, computer calculations were used to model the piping and coolers and the detector geometries. The modeling effort was designed to determine the quantity of fuel, distributed in likely locations in the Letdown Cooler Room, that would produce the measured ^{144}Ce signal. During the actual measurements, no counts from ^{144}Ce were recorded, so the fuel quantity was determined using a minimum detectable level (MDL) analysis.

MEASUREMENT METHODOLOGY

The room containing the letdown coolers has a very high background dose rate (10 R/h or greater), preventing personnel entry. Thus, the fuel measurement had to rely on remote measurements. The measurements involved placing a shielded NaI detector into an access port in the Letdown Cooler Room wall. The shielding around the detector had an opening which provided the detector with a well-defined field of view of equipment in the room.

The collimated detector was placed in two slightly different positions to obtain two different fields of view of the room. At each position, a very long count was performed (43 hours for the first position, 67 hours for the second). The recorded spectrum from each count could be analyzed to determine the number of events in the 2.19-MeV peak.

ANALYSIS METHODOLOGY

The gamma measurements relied on determining the quantity of ^{144}Ce as an analog for the fuel that was redistributed during the accident. The isotope ^{144}Pr, a progeny radionuclide of ^{144}Ce, emits a gamma with energy of 2.19 MeV. Thus, a count of the 2.19 MeV gammas can be converted to a mass of ^{144}Ce, and this mass can be converted to a mass of UO_2 using a measured Ce-to-fuel ratio.

Analysis of the recorded gamma spectra showed no events attributable to the 2.19 MeV photon emitted by ^{144}Ce. Counts were recorded in the region of interest corresponding to the peak, but they were part of a background

continuum of events and did not form a statistically identifiable peak. Thus, no direct evidence of the presence of ^{144}Ce was produced by the gamma measurements.

Since the radiation background in the Letdown Cooler Room was high, it is possible that gammas emitted by ^{144}Ce in the room could have been lost in the background and not detected by the NaI detector. The licensee took the conservative approach of calculating the maximum quantity of ^{144}Ce that could be present in the room but masked by the background signal at the detector. MDL analysis was used to find the estimate of record.

The MDL analysis started by calculating the highest gamma flux at the detector that could give a signal that would be lost in the background. The analysis then applied the computer codes ISOSHLD-II and QAD-QL to model the room with an assumed fuel distribution. The first calculation used an improbable fuel configuration chosen to give a low MDL, and determined the quantity of fuel that would produce the MDL gamma flux at the detector. This quantity was 1.9 kg. A different fuel distribution, chosen to give a high MDL quantity, was then modeled, and gave a MDL fuel quantity of 25.6 kg.

The 1.9 kg and 25.6 kg represented boundaries for the MDL fuel quantity, since it represented extreme (and improbable) fuel distributions. The fuel quantity of record was found by modeling the most likely distribution of fuel in the room, with a uniform distribution of fuel in the cooler inlet distribution header, and in the bottom quadrant of the first spirals of the tubes in the cooler. With this distribution, an MDL quantity of 3.7 kg was determined.

An error estimate of ±53% (one σ) was chosen for the fuel quantity. This uncertainty value is based on uncertainties associated with the calibration and measurement counting, physical measurements, calibrations, and the standard source.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

The methods chosen for analysis of the quantity of fuel in the Letdown Cooler Room is basically appropriate. It would seem beneficial to have duplicated the measurement with a HPGe detector, which would give better resolution and probably lower MDL values than the NaI detector, but as the report stated, space restrictions in the penetration precluded use of the HPGe. The measurement conditions were very difficult with the high gamma backgrounds, and the experimenters did a fine job in dealing with these conditions. When the gamma measurements failed to find identifiable peaks for the fuel analog, an MDL analysis was performed. Using this MDL value for the estimate of record is a prudent and conservative approach.

The stated uncertainty of ±53% (one σ) does not adequately address two important factors in the analysis: the fact that an MDL value is a maximum; and the fact that assumptions were made about the distribution of fuel in the room. Since the gamma measurements showed no evidence of fuel in the room, a value of 0 kg should be a likely possibility, but as stated, the value of 0 kg

lies 1.9σ away from the most probable value, so there is less than a 3% probability that there is no fuel in the room. A logical analysis of the translocation of fuel would indicate that the presence of fuel in the room would be possible, but the measurements give no evidence for the presence. Thus, a value of zero should be included in the error bound.

The assumptions about fuel distribution that are necessary to arrive at the quantity of fuel are not reflected in the ±53% uncertainty. The maximum MDL modeled was 25.6 kg, which is almost 6σ above the best estimate. The report stated that this was an improbable distribution, but no attempt was made to find a "probable" maximum configuration.

Rather than stating the errors as a Gaussian distribution, it would have been better to identify a likely range of fuel values, with the best estimate of 3.7 kg and an uncertainty ranging from 0 kg up to the value matching the maximum credible configuration.

SUMMARY AND CONCLUSIONS

The licensee's best estimate of the quantity of fuel remaining in the Letdown Cooler Room is 3.7 kg UO_2, with an uncertainty of ±53%, ranging from 1.7 kg to 5.7 kg UO_2.

The use of gamma measurements for determining the fuel quantity is appropriate, although the high gamma background was sufficient to mask any peak that may have come from the 2.19-MeV gammas that would have been emitted by ^{144}Ce. In the absence of recorded data, it is conservative to perform a minimum detectable level analysis and use this value as the estimate of record. The stated error did not appear to be an appropriate method of presenting the uncertainties, because it did not adequately reflect the fact that a MDL is a maximum value, and a value of zero is a definite possibility. The error distribution also did not adequately reflect the necessary assumptions about fuel distributions that occurred in the modeling effort. However, the estimate of fuel loading in the Letdown Cooler Room appears basically sound.

REFERENCES

GPU Nuclear. September 29, 1989. TMI-2 Technical Bulletin: Letdown Cooler Room SNM Accountability Summary. SNM-88-07, Rev. 1.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE UPPER PLENUM ASSEMBLY

INTRODUCTION

This review of the licensee's PDSR (GPU Nuclear, 1988) describes the process for estimating the amount of UO_2 remaining in the upper plenum structure of the TMI-2 reactor. The remaining amount of fuel in this structure was estimated to be 2.1 kg with an uncertainty of -45% to +140%.

This estimate was based on the analysis of contamination found on two leadscrew samples and a sample from a control rod assembly support tube. Contamination levels found on these samples were applied to surfaces in the rest of the plenum region to derive an estimate of the total fuel content in the whole structure. A video examination of approximately 50% of the plenum internals gave no evidence of lodged debris.

MEASUREMENT METHODOLOGY

The estimate of the fuel loading in the plenum is based on an analysis of contamination on two leadscrews and one control rod guide tube section. The leadscrews were removed from the reactor before head and plenum were removed from the reactor vessel. Surface deposits and films were dissolved from various locations on the leadscrews and measured using delayed neutron emission analysis to determine the amount of ^{235}U. The amount of UO_2 in each of these deposits was then determined using the average core enrichment value of 2.57% ^{235}U. The derived values of UO_2 contamination, in units of μg per cm^2, were used as representative contamination values for the entire surface area of the upper plenum. Different values were derived for surface films (residing on both horizontal and vertical surfaces), and for sediment deposits (residing only on horizontal surfaces).

A control rod drive mechanism guide tube (also known as a leadscrew support tube) was analyzed to determine contamination values. This analysis was based on concentrations of ^{137}Cs and ^{134}Cs on the inside and outside of the tube. The results of this analysis were assumed to be representative of the fuel distribution on the guide tube and were used to account for the differences in contamination level between internal and external control rod guide tube surfaces. The analysis found an activity ratio of 2:1 for outside surfaces versus inside surfaces. This activity ratio was assumed to be due to coolant flow rate, with the higher flow outside the tube causing higher surface contamination. Thus the leadscrew fuel contamination values were doubled for surfaces outside the control rod guide tubes where there was a higher flow.

Four types of surfaces were present in the plenum: horizontal surfaces exposed to low flow, horizontal surfaces exposed to high flow; vertical surfaces exposed to low flow and vertical surfaces exposed to high flow.

Video inspection of a large fraction of the upper plenum surfaces indicated that there was no evidence of lodged debris or granular particles on the surfaces, so the method of depositing contamination on the surfaces was assumed to be identical to the method of deposition on the leadscrews. Thus the fuel loading values based on the leadscrew analyses were then applied to the total surface areas in each of the four categories to arrive at a total UO_2 inventory for the upper plenum structure.

ANALYSIS METHODOLOGY

As described in the previous section, the following fuel loadings were calculated:

1) UO_2 film on low-flow surfaces = 216 $\mu g/cm^2$ = 337 g
   based on same fuel loading as leadscrews;

2) UO_2 film on high-flow surfaces = 432 $\mu g/cm^2$ = 1166 g
   based on twice fuel loading on leadscrews;

3) UO_2 sediment on low-flow surfaces = 2520 $\mu g/cm^2$ = 164 g
   based on average UO_2 loading on smooth section of leadscrew, subtracted from UO_2 loading on threaded section divided by 0.2 to adjust for 20% of thread that is horizontal; and

4) UO_2 sediment on high-flow surfaces = 5040 $\mu g/cm^2$ = 432 g
   based on average UO_2 loading on non-threaded section of leadscrew.

These contamination levels were used to find a total quantity of 1.5 kg UO_2 in surface films and 0.6 kg UO_2 in sediments, for a total of 2.1 kg UO_2 in the upper plenum.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

The measurement and analysis methodology appears appropriate. This analytical method assumes that the four fuel loading values were representative of all upper plenum surface contamination, and that these four loadings were uniformly distributed over the four types of surfaces. However, the fuel loadings would have varied:

1) if the axial profiles of UO_2 concentrations on the leadscrew surfaces varied;

2) if the two leadscrews studied were not representative of the upper plenum concentrations;

3) if the difference in measured UO_2 concentrations on the leadscrews was due to something other than the threaded versus non-threaded regions;

4) if the surface concentrations were not uniform if (i.e., "blotches" of higher concentrations or lower concentrations were present);

5) if the flow patterns did not justify deposition based only on a simplistic model of high-versus-low flow; and

6) if the deposition of Cesium on the control rod drive mechanism guide tube did not reflect the deposition of ^{235}U.

These six items may represent additional factors that could have increased the stated uncertainty for the UO_2 estimate. The stated uncertainty associated with the 2.1 kg fuel inventory (-45%, +140%) is based on the uncertainty in the delayed-neutron analysis of ^{235}U on the leadscrew samples and on postulated distribution extremes for the plenum contamination.

The accepted value changed from 3.2 kg (Revision 1 of SNM-87-07, 3/8/88) to 2.1 kg in the final version (Revision 2) of the PUSR. In the original analysis, the licensee had used a model with a surface loading of 400 $\mu g/cm^2$, and a high flow region with 800 $\mu g/cm^2$, but they did not consider sedimentation. For Revision 2, they fine-tuned the surface loading values to 216 and 432 $\mu g/cm^2$, but added sedimentation. The effect of these changes was to lower the best estimate of fuel from 3.2 kg to 2.1 kg UO_2. As they made this change, however, they re-evaluated the error bounds, changing them from 1.1 kg to 4.7 kg (Revision 1) to 1.2 kg to 5.0 kg (Revision 2). Thus, even though the best estimate decreased, the lower and upper bounds were increased only slightly.

SUMMARY AND CONCLUSIONS

The licensee's best estimate of the quantity of fuel remaining in the Upper Plenum is 2.1 kg UO_2, with an uncertainty ranging from 1.2 kg to 5.0 kg UO_2.

The use of measured uranium concentrations on leadscrews should be a good indication of contamination on other surfaces; ^{235}U is the best isotope to study. Additional measurements of other leadscrews or of other parts of the plenum are unwarranted due to the difficulty and personnel exposure involved in the additional work and the small amount of fuel present in these areas, despite the concern regarding the representativeness of the leadscrews studied. It may be appropriate to increase the error bounds, and perhaps to choose a higher value within the bounds, to account for the possibility that the two leadscrews studied had lower-than-representative concentrations on them. However, the estimate of fuel loading in the upper plenum appears basically sound.

REFERENCES

GPU Nuclear. October 18, 1988. TMI-2 Technical Bulletin: Upper Plenum Assembly S.M Accountability Summary. SNM-87-07, Rev. 2.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE PRESSURIZER

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1989) discusses the process for estimating the amount of UO_2 remaining in the pressurizer. The quantity of fuel estimated to be present at this location is 0.3 kg with an uncertainty of 52%.

In this study, the volume of debris remaining in the bottom of the pressurizer was determined by examining videotapes. The density of the debris, determined from previous studies, was applied to the volume to determine the mass of debris. The concentration of UO_2 in debris was determined by counting a debris sample that had been retrieved from the pressurizer. This sample was counted using passive gamma measurements of the ^{144}Ce and ^{154}Eu contained in the sample and active neutron measurement of the ^{235}U. A concentration based on these measurements was applied to the debris mass to arrive at an estimate of the remaining UO_2. Since fuel is likely to exist in films coating the interior surfaces of the pressurizer, an estimate of this fuel quantity was also made, but it was insignificant compared to the mass of fuel in the debris.

MEASUREMENT METHODOLOGY

The amount of residual UO_2 in the pressurizer was estimated by determining the mass of fuel debris in the bottom of the pressurizer and applying a fuel concentration factor determined by measurements performed on a sample of debris retrieved from the pressurizer. The volume of residual solids was determined by visual inspection using a remote video camera. The mass of debris was found by applying a density value to the estimated volume.

A 100-gram sample of fuel debris had been retrieved from the pressurizer, and this sample was analyzed using both gamma spectroscopy and active neutron interrogation. In the gamma spectroscopy measurements, gammas emitted by ^{144}Ce and ^{154}Eu were detected and used to determine the quantity of these two isotopes present in the sample. Since these two isotopes had both been identified as analogues of fuel, the quantity of fuel could be determined from the measured quantities of ^{144}Ce and ^{154}Eu.

In the active neutron interrogation, a thermal neutron source was used to irradiate the fuel sample. The neutrons would cause fission to occur in the ^{235}U contained in the fuel sample, and fast neutrons emitted by the fission reaction were counted by a ^{4}He detector. The neutron detection system was run in 3 configurations: with no uranium present, to obtain a background count rate (primarily from the source neutrons); with a UO_2 standard in place to give a count rate from a known source of ^{235}U; and with the debris sample in place for measurement of the U contained in the sample.

ANALYSIS METHODOLOGY

The volume of debris remaining in the bottom of the pressurizer was determined by having five engineers independently examine the videotapes of the pressurizer's interior. Each one arrived at a volume. The value of record (908 cm^3) was the mean of these five estimates, and a standard deviation of ±35% was derived from the range of the five estimates. A bulk density of 4.4 g/cm^3, previously determined for TMI-2 core debris, was applied to the volume.

Gamma spectroscopy was performed on the 100-gram sample of debris that had been retrieved from the pressurizer. This measurement found 420 μCi of ^{144}Ce and 284 μCi of ^{154}Eu in the sample. Previous studies had determined that these isotopes stayed with the fuel as it was transported during the accident and recovery, and concentration ratios for these nuclides in fuel had been determined. These concentration ratios were applied to the results of the gamma measurements to give values of .0051 kg UO_2 (in the debris sample) based on the ^{144}Ce measurement, and .0070 kg UO_2 based on the ^{154}Eu measurement.

In the active neutron interrogation measurement, the fuel sample was exposed to thermal neutrons emitted by a moderated Sb-Be source. The thermal neutrons induced fission in the fissile isotopes of the fuel, which caused the emission of fast neutrons. After determining the emission rate from a standard source of ^{235}U (in a UO_3 sample), the quantity of UO_2 in the debris sample was estimated to be 0.0097 kg.

The three different values for UO_2 in the debris sample were "averaged" (using a weighting scheme) to arrive at a value of 0.0067 kg UO_2 in the sample. The quantity in the sample was then scaled up, using the ratio of the pressurizer debris mass to the sample mass, to obtain a value of 0.267 kg UO_2 in the pressurizer.

The presence of fuel in thin films on the interior surface of the pressurizer and internal components was taken into account. Measurements had been done to estimate the quantity of fuel in films adhering to the interior of the pressurizer's manway cover. This concentration per unit area was then applied to the total surface area of the interior of the pressurizer and its components. The resulting mass of fuel in surface films was estimated to be in the range of 1 gram. This quantity was insignificant compared to the quantity of fuel in the debris, so it was essentially ignored.

The estimate of record was reported to one significant digit as 0.3 kg UO_2. An overall uncertainty of ±52% was assigned to this quantity. The uncertainty was based on the counting uncertainties and the volume-determination uncertainties.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

The analysis technique was appropriate for this situation. Determining the volume of debris by inspecting videotapes, and using the estimates of five independent observers is a good approach. It appears that the uncertainty analysis did not include any uncertainty associated with the debris density - it seems likely that there would be some variability in densities that would influence the overall uncertainty.

Using gamma spectroscopy to count the gamma-emitting fuel analogues, then comparing these values to active neutron counting, is a good approach to determining the quantity of fuel in the retrieved samples. The fact that the three error bars overlap, lends credence to the analysis. The neutron measurements would seem to be more reliable than the gamma measurements, since these are directly measuring uranium as opposed to analogues to the fuel. The neutron measurements, however, used UO_2 for its standard source. The UO_2 has only one fissile isotope - ^{235}U, while the debris sample would contain other fissile isotopes such as ^{239}Pu. This would cause some uncertainty in comparing the standard to the unknown, but this uncertainty is probably small compared to the other uncertainties in the measurements.

SUMMARY AND CONCLUSIONS

The licensee's best estimate of the quantity of fuel remaining in the pressurizer is 0.3 kg UO_2, with an uncertainty of ±52%. The pressurizer was chosen for special nuclear material (SNM) study because of the likelihood of the deposition of fuel in the system. The analysis proved that the quantity of fuel was very small compared to the rest of the TMI-2 inventory. The method for determining the quantity of remaining fuel is basically sound.

REFERENCES

GPU Nuclear. September 29, 1989. TMI-2 Technical Bulletin: Pressurizer SNM Accountability Summary. SNM-88-05, Rev. 2.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT
for
THE REACTOR BUILDING BASEMENT

INTRODUCTION

This document reviews the TMI-2 licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1989) describing the measurements and analytical methods used to determine the quantity of fuel remaining in the Reactor Building (RB) basement. The estimated quantity of UO_2 is 1.3 kg ± 54%, almost all of which was located on the basement floor. The measurement methodologies include gamma spectroscopy, and analyses of sediments, water and concrete samples. The major portion of the measured fuel is in the reactor coolant drain tank (RCDT) discharge area. The amount of UO_2 in this area was determined by applying a known ^{144}Ce/fuel ratio to the amount of ^{144}Ce determined from the gamma spectroscopy measurement.

MEASUREMENT METHODOLOGY

Three gamma spectroscopy scans were performed in different parts of the RB basement. Two of these measurements were not used in the evaluation, however, due to the lack of a ^{144}Ce peak and a high minimum detectable level (MDL) value caused by high water levels. The third gamma spectroscopy measurement was performed using a NaI detector in the RCDT discharge area, and this measurement was used for the estimate of record.

The UO_2 contained in the sediment was measured by analyzing samples of the sediment. Radiochemical analysis was used to estimate the UO_2 contents. Although many samples of sediment were obtained, only three of the samples contained enough mass to be used. One of these three samples, taken in the RCDT, was not included since it was not representative of the reactor basement floor.

To quantify the UO_2 content of the water on the floor of the reactor basement, 17 water samples were taken in 1988. The amount of UO_2 in the water samples was determined by boiling off the liquid, and subjecting the remaining solids to alpha counting using a scintillation probe.

Gamma spectroscopy measurements of concrete cores removed from the basement walls were performed but did not show the presence of ^{144}Ce. The UO_2 content in the walls was therefore assumed based on estimated parameters for the diffusion of water into the concrete.

ANALYSIS METHODOLOGY

The analysis of the gamma spectrometry measurement resulted in an estimate of 1.2 kg of UO_2 for the RCDT discharge area, with an uncertainty of ±58%. This amount was determined by quantifying the amount of ^{144}Ce based on the 2.2 MeV photopeak and applying a known ^{144}Ce/fuel ratio.

The quantity of UO_2 in the remainder of the RB Basement sediments (in areas outside the RCDT discharge area) was estimated by determining the uranium concentrations in the sediments (3.7×10^{-6} g UO_2 per gram of sediment sample) and multiplying by the sediment mass. This analysis gave an estimate of 0.05 kg ±80%. The total quantity of fuel in the sediment was therefore estimated to be 1.3 kg ±54%.

The alpha counting of the basement water samples showed very low concentrations of fuel in the water, and the analysis gave an estimate of less than 0.3 g of fuel dissolved in the total volume of water. This small quantity did not affect the estimate of 1.3 kg for the RB Basement fuel inventory.

Gamma spectrometry of concrete cores taken from the basement walls did not show the presence of ^{144}Ce, the gamma-emitting analog for fuel. Thus the quantity of fuel contained in the walls was estimated from the concentration of fuel in the water (which was very small) and assumptions about the absorption of water into the concrete. This analysis resulted in an estimate of 0.003 g of UO_2 in the walls. This figure did not affect the value of 1.3 kg.

Corrections to these measurements were necessary because some of the sediment was removed after the measurements were made. One correction was based on a measured amount of UO_2 (0.2 kg) removed during the desludging operation. The removed quantity of fuel was based on video inspections (which were the basis for an estimate of the mass of sludge removed), and measured concentrations of the fuel in sediment. This correction lowered the mass of UO_2 by 0.2 kg.

The licensee also accounted for the reactor fuel that may have been deposited in the drain system as a result of the decontamination of reactor defueling tools at the 347 foot elevation in the post decontamination facility. The amount of fuel deposited in the drain was estimated from the number of tools decontaminated, an apparently measured surface activity, and an assumed average surface area. The resulting fuel was assumed to have been washed into the sump. This value, 0.2 kg, was added to bring the final estimate of record up to 1.3 kg UO_2.

REVIEW OF MEASUREMENT AND ANALYSIS METHODOLOGY

The measurement methods used in this study should have provided a reasonable estimate of the amount of UO_2 remaining in the RB basement. If the study had any inadequacies, they resulted from the difficulty of working in a very high radiation area.

One cause for concern arises from the estimate of fuel in the sediment. The general sediment concentration was found to be 3.7 μg of UO_2 per gram of sediment. This value is very close to the average concentration of uranium in soil, and indeed, much of the sediment is assumed to come from river water. Thus the measured uranium could be natural (environmental) uranium rather than fuel. A bigger concern, however, is whether this concentration is representative of the entire basement. The estimate is based on two samples, both taken from the "Impingement Area". These measured concentrations for these two samples had a high standard deviation, 81%. Thus it is hard to justify the assumption that 3.7 $\mu g/g$ is representative of the entire basement. It would be preferable to select samples from areas all over the basement to get a better estimate of the fuel in the sludge. Of course, the radiation environment in the basement is so severe that a limited sampling scheme was necessary, and the mass of fuel was so small that it did not merit a more detailed study.

Another concern arises concerning the gamma spectrum measurement of the RCDT discharge area. The mass of fuel determined by this measurement constituted the bulk of the fuel in the basement, yet the determination was based on only one measurement. The concentration of fuel in this sediment was much greater than in sediments in other areas. The analysis assumes that all the high-concentration sediments were enclosed in the NaI detector's viewing area; however, if any high-concentration sediments were missed by the spectral measurement, fuel would have been missed.

The gamma spectroscopy measurements of the RCDT discharge area resulted in a quantity of 1.2 kg ±58% of UO_2 estimated to be present in the RB basement. Although the reported amount included an associated standard deviation of 58%, this standard deviation was based only on counting errors and the error associated with the ^{144}Ce/fuel ratio (neither of these errors were stated separately). Possible errors associated with geometry or calibration were not mentioned. Because details regarding the counting geometry were not provided, the magnitude of the potential error associated with the uncertain measurement geometry can only be conjectured. However, considering the layout of the area that was scanned and the high local dose rates, this error could have been significant.

Omitting an uncertainty estimate for the quantity of fuel attributed to tool decontamination is significant, since this contributor accounts for 15% of the fuel estimate and the analysis included large assumptions. Some tool decontamination activity occurred after this estimate was made, and this activity should have been included in the estimate.

The methods of measurement and analysis for determining the amount of fuel in the RB basement water and the RB basement walls seems reasonable. In both cases, the amount of UO_2 determined to be present was negligible.

SUMMARY AND CONCLUSIONS

All measurement methods used to determine the amount of UO_2 remaining in the RB basement appeared to be reasonable. However, the analysis methods failed to account for several potentially important sources of error. The most prominent of these was the apparent failure to assess the geometric uncertainty associated with the gamma spectroscopy measurement of the RCDT discharge area. This measurement accounted for over 80% of the final estimate of 1.3 kg UO_2 ±54% in the RB basement. As a result, it is possible that the uncertainty associated with the measurement should be greater than that indicated by the stated error.

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GPU Nuclear. September 29, 1989. TMI-2 Technical Bulletin: Reactor Building Basement SNM Accountability Summary. SNM-89-06, Rev. 0.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR REACTOR BUILDING MISCELLANEOUS COMPONENTS

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1991) discusses the process for estimating the amount of UO_2 remaining in miscellaneous components of the Reactor Building. The quantity of fuel estimated to be present in these components totals 64.0 kg. Adding the uncertainty estimate, which is asymmetrical for the evaluation, gives a range of 31.6 kg to 85.4 kg.

The components included in this study were evaluated using several different techniques, including:

- measuring gross gamma exposure rates;
- gamma spectroscopy;
- neutron interrogation; and
- direct sampling and analysis.

The choice of analytical method used for each sample depended on the nature of the component, the accessibility and the background radiation field.

The components addressed in this survey include:

1) Reactor Coolant drain tank;
2) fuel transfer canal (FTC)/transfer tubes
   - fuel that dropped from canisters transferred through the canal
   - fuel remaining in the fuel transfer tubes
   - loose fuel on the Upper Plenum assembly
   - rubber hose stored in the FTC
   - fuel washed into the FTC during reactor vessel draindowns;
3) core flood system;
4) upper endfittings;
5) Tool Decontamination Facility (TDF);
6) drain line from TDF;
7) defueling water cleanup system;
8) temporary RV filter system;
9) incore guide tubes; and
10) defueling tools.

MEASUREMENT METHODOLOGY

Reactor Coolant Drain Tank

The mass of UO_2 remaining in the Reactor Coolant drain tank was estimated based on video inspections of the interior. A small camera was lowered into the tank and operated at one end. The images were

evaluated to determine the depth of the deposits on the bottom, and this value was used to calculate a volume of deposited debris. Samples of the debris were also collected, and the samples were analyzed to determine the density of the debris and the uranium composition. These values were then used with the volume to calculate the mass of UO_2.

Fuel Transfer Canal/Transfer Tubes

This survey estimated the fuel remaining in the FTC resulting from five different sources:

- fuel that dropped from canisters transferred through the canal;
- fuel remaining in the fuel transfer tubes;
- loose fuel on the Upper Plenum assembly;
- rubber hose stored in the FTC; and
- fuel washed into the FTC during Reactor Vessel (RV) draindowns.

1) Measurements of fuel debris in the transfer canal were made after 301 canisters had been transferred. Gamma spectroscopy was used to estimate the quantity of ^{137}Cs in the FTC, and this value was then translated into a fuel quantity using a ^{137}Cs-to-fuel ratio that had been determined previously. An additional 41 canisters were transferred through the canal after these measurements had been made, so the fuel debris contributed by these transfers was found by ratioing from the measured quantity.

2) The quantity of fuel in the fuel transfer tubes was determined using gross gamma measurements. A Geiger-Muller (GM) tube was used to measure the gamma exposure rate, and these measurements were used to determine the fuel in the tubes using the assumption that the fuel was deposited only in the lower 60° of arc.

3) The upper plenum assembly is stored in the FTC. Video images were used to estimate the small quantity of loose fuel that resides on the assembly.

4) Approximately 700 feet of rubber hose is stored in the FTC. The inner surface of these hoses have been exposed to fuel. The fuel contained inside the hoses was estimated by cutting short lengths of the hoses and determining the fuel concentration per unit area. This concentration was then applied to the entire inner area to get a deposited fuel mass.

5) At the time that the PDSR was written, additional draindown of the RV was scheduled to occur in the future. It was certain that additional fuel would be washed into the FTC during these operations, but of course it was not possible to make any measurements. The quantity of fuel that would be deposited in the FTC was therefore estimated using engineering judgement.

Core Flood System

This PDSR addresses fuel deposited in the A side flood tank, B side tank, in the associated piping, and in pieces of the lower core support assembly (LCSA) stored in the A core flood tank. Video inspections were performed on the LCSA pieces to ensure that no fuel was adhering to the surfaces. Gamma measurements were made on the LCSA pieces, tanks and pipes to estimate fuel quantities. The gamma measurements were all gross exposure rate measurements rather than gamma spectral measurements. Measurements on the piping were made with HP-220A probes inside a hemispherical tungsten collimator to minimize background.

Upper Endfittings

Six storage containers holding a total of 17 upper endfittings were analyzed for fuel content using a neutron interrogation system. The detection system used a SbBe photoneutron source to irradiate a storage container with neutrons. The neutrons would cause fission in the ^{235}U atoms contained in the fuel, and the emitted neutrons were then counted by ^{4}He detectors as an indicator of the quantity of fuel present.

For each storage container measured, the detectors were first operated with no SbBe source in place to get a background reading. A second measurement was then made with neutron irradiation. A third count was then made with a known quantity of natural uranium in the container to produce an increase in the count rate. The increase would be due to the known quantity of the spike, and this rate would then be used to find the calibration factor.

Neutron interrogation measurements were performed on five of the six storage containers.

Tool Decontamination Facility

The quantity of fuel remaining in the three rooms of the TDF was estimated using gamma exposure rate measurements in each of the rooms. The contamination was assumed to be spread on the floor of each room, with no contamination on the walls or ceilings. Gamma exposure rates were measured at a number of points in the room to determine the uniformity of the gamma field and to provide the data for modeling the fuel contamination.

Drain Line from the Tool Decontamination Facility

Fuel that had been deposited in the drain line that extended from the TDF down to the basement was quantified using a miniature ("peanut") GM tube. The GM tube, shielded with a cadmium cover to improve its energy response, was sealed inside a long polyethylene tube and passed through the pipes, measuring the gamma exposure rate at intervals through the pipe. The exposure rates were very high: the general background dose rate was 5 R/h, and the GM tube had readings in the pipe ranging up to 44 R/h. The detector was used in a 33-foot stretch of the horizontal

drain line that runs beneath the floor of the 282-foot level, but not through the horizontal pipe that runs under the 347-foot level.

Defueling Water Cleanup System

Video inspections were used to estimate the quantity of fuel remaining in two circulating water pumps that had been used to remove fuel fines from the reactor coolant system water. The inspections identified fuel in several areas of the pumps, including the horizontal surfaces of the pump transition areas, and two other areas. The video images were inspected to estimate the fuel quantities there.

The defueling water cleanup system (DWCS) manifold, consisting of piping, valves and other components, were assessed using gross gamma measurements. A detector with a hemispherical tungsten collimator was positioned on top of the A D-Ring, pointed at the manifold. This detector read 5.48 mR/h, including a background of 5.05 mR/h. The detector was then moved to the top of the B D-Ring, and pointed at the manifold to give a reading of 6.59 mR/h, with a background of 2.37 mR/h. These detector locations were approximately 24 feet from the manifold.

The third component of the DWCS analysis consisted of 6550 feet of hoses. One-foot segments of these hoses were counted using gamma spectrometry. A second method used gross gamma measurements to arrive at an independent fuel mass estimate.

Temporary Reactor Vessel Filter System

The estimate of fuel remaining in the two filters was estimated by taking a sample of diatomaceous earth from one of the filters, and performing an isotopic analysis in the laboratory. The fuel content of the sample was then applied to the entire contents of both filters.

Incore Guide Tubes

Gamma spectrometry using a lead-shielded high-purity germanium (HPGe) detector was used to measure the gammas emitted by ^{144}Ce contained in incore instrument guide tubes (IIGTs) that were contained in 13 different plates that were cut from the flow distributor assembly. The ^{144}Ce was used as an analog for the fuel contained in the IIGTs.

Defueling Tools

A total of 117 defueling tools were evaluated for their fuel content. The study first used gamma spectrometry with a shielded HPGe detector to count the gammas emitted by ^{144}Ce on five tools. Data from this measurement was used to estimate the fuel quantity on these five tools. A gross gamma exposure rate measurement was also performed on these five tools to arrive at a calibration factor. Gross gamma measurements were then performed on 69 additional tools to arrive at their fuel quantity using the calibration factor derived previously.

ANALYSIS METHODOLOGY

Reactor Coolant Drain Tank

The video inspection of the interior of the tank gave an estimate of 0.16 cm for the thickness of the debris deposited on the bottom of the tank. This estimate was based on inspections performed at only the west end of the tank. The debris was assumed to cover one-eighth of the interior surface of the tank, which would have a surface area of 52,000 cm^2. This depth, assumed to be uniform over the deposition area, multiplied by the area gave a volume of 8320 cm^3. The mass of the debris was then calculated using the density of 6.2 g/cm^3, determined from analysis of the debris sample, and an assumed water content of 50% in the debris volume. The debris mass was thus estimated to be 26 kg.

The total quantity of UO_2 in the debris was found by applying a factor of 3.7 mgU/g. This factor was derived from the results of neutron activation analysis studies performed on the debris sample that was retrieved from the tank. Applying this factor to the 26 kg of debris mass gave a total mass of 0.1 kg UO_2 in the tank.

The uncertainty in the fuel mass was assumed to come from two components: the uncertainty in the laboratory analyses of the debris sample (assumed to be 20%), and an uncertainty in the debris volume, which was much greater. The overall uncertainty in the fuel mass is 54%.

Fuel Transfer Canal/Transfer Tubes

1) A measurement of the FTC was performed in 1990, after 301 canisters had been transferred through the canal. The activity of ^{137}Cs was estimated, based on gamma measurements, and a ^{137}Cs-to-fuel activity ratio of 1800 $\mu Ci/g$ was applied to get a fuel quantity of 12.2 kg. The ^{137}Cs-to-fuel activity ratio has since been revised to 1355 $\mu Ci/g$, which caused a revision in the uncertainty value to +24%, -100%. After the measurements, 41 additional canisters were moved through the canal, presumably depositing additional fuel during the transfer. Since the first 301 canister movements deposited a presumed 12.2 kg, the average deposition per canister would be .041 kg. The 41 additional canister movements would deposit 41 x .041 kg = 1.7 kg UO_2. The total fuel deposited in the FTC by movement of canisters is therefore estimated as 13.9 kg, with an uncertainty of +24%, -100%.

2) It was assumed that the fuel is deposited in the fuel transfer tubes was located in the lower 60° of arc. This produces a fuel estimate of 0.5 kg with an uncertainty of +34%, -92%.

3) The Upper Reactor Plenum assembly is currently stored in the fuel transfer canal. This PDSR addresses only loose debris that may reside on the assembly and may fall into the FTC, which was estimated by video inspection at less than 0.1 kg. Additional fuel is fixed to the surfaces of the plenum, but these deposits were addressed in a separate PDSR.

4) The laboratory analyses of the rubber hose sections currently stored in the FTC, determined a fuel mass of .015 grams deposited on an inner surface of 48E cm^2. Applying this concentration to the entire 37,700 cm^2 of inner hose surface gives a total fuel estimate of 0.01 kg, with an uncertainty of 100%.

5) The additional fuel that would be added to the FTC during draindown was estimated based on estimates of the quantities of fuel in other areas and estimates of the fuel quantities that would wash into the FTC. These estimates were made for each part of the draining and refilling cycles as stated below.

- During the first draindown cycle, done to accommodate the RV neutron measurements, it was estimated that most of the fuel would wash in from the lower head region. There was an estimated 8.1 kg UO_2 in the lower head region, and the jet pump would be put in a position to collect about 13% of this, so the study attributed a 1.1-kg transfer of fuel to the FTC.

- The study assumed that the Reactor would be refilled during the neutron measurement program, and then drained again. The RV would be refilled with water from the "B" makeup line, and this procedure would wash loose fuel out of the "1B" cold leg and nozzle. As the water fills the RV, there will be additional sloshing, and more fuel could be relocated to the lower head. The second draining would then relocate 1.9 kg UO_2 to the FTC.

- The water in the steam generators (OTSGs) would also be drained to the RV, and the study estimated that this would contribute another 1 kg fuel to the RV. As the OTSGs are being drained, water from the RV will be periodically drained to the FTC, and this process could contribute 0.1 kg UO_2 to the FTC.

- The final draindown of the RV will transfer about 1.4 kg fuel to the FTC, so the total quantity of fuel transferred from the RV to the FTC would be (1.1 + 1.9 + 0.1 + 1.4) = 4.5 kg UO_2, with an uncertainty of ±78%.

The final estimate of record of fuel in the FTC is a sum of the five components listed above, totaling 18.9 kg UO_2, with an asymmetrical uncertainty of +37%, -95%. Thus a range from 2.4 kg to 26.2 kg bounds the estimate of record.

Core Flood System

The results of the gamma measurements were used to estimate fuel quantities by computer modeling, using a source geometry for fuel deposits inside a pipe. Most of the piping on the A side was inaccessible, so measurements were made on the B side, and the resulting linear fuel estimate was applied to the length of piping on the A side to arrive at a total fuel estimate. The estimate of fuel deposits inside the piping were 0.89 kg for the A side and 1.67 kg for the B side.

Besides deposits, the insides of the pipes and tanks were also assumed to be coated with a film similar to the fuel film inside the Reactor Coolant System (RCS). The average mass of fuel per unit area inside the RCS was multiplied by the inside surface area of the core flood system to give a total fuel mass due to films. The mass of fuel in films on the A side was estimated at 0.008 kg, and the mass of films in the B side is 0.010 kg.

The mass of fuel fixed to the LCSA pieces in the A side core flood tank was determined by the analysis of gamma and alpha measurements performed on the pieces. The estimate of fuel on the LCSA pieces is 2.3 kg.

The estimate of record for the UO_2 in the core flood system is the sum of the components mentioned in the preceding paragraphs, 4.9 kg. The uncertainty estimate is ±77%.

Upper Endfittings

Five endfitting storage containers, all of which are stored on the 347-foot level, were measured using the neutron interrogation technique. The measurement showed that these containers, holding 14 endfittings, contained 4.85 kg of fuel. The sixth container is stored in the deep end of the FIC, where it was inaccessible for neutron interrogation. The average fuel mass per endfitting, 0.346 kg, that was determined in the five other measurements, was applied to the three endfittings in the sixth storage container. The estimate of record for all six containers was 5.89 kg UO_2. The uncertainty was estimated at ±85%, with the major components being system efficiency and measurement error.

Some of the measurements experienced count rates that were low compared to the background signal, and Chavenet's criterion had to be applied to determine which counts above background were significant. This technique is a useful statistical technique that is sometimes applied in low count rate applications.

It should be noted that one storage container resides in the FTC, but the contained UO_2 is included in the endfitting total, not in the FIC estimate.

Tool Decontamination Facility

Measured dose rates in the first room of the TDF were fairly uniform throughout the room, so the room was modeled as a uniform area source with dimensions matching the floor dimensions. The computer code Microshield was then used to estimate the quantity of fuel spread over the area source that would give the measured dose rate, and the resulting calculation gave less than 0.1 g of UO_2 in the room (the measured dose rate was essentially equal to the background dose rate). The second room was also modeled as a uniform area source, and its higher measured dose rate gave a fuel estimate of 5 g UO_2.

The third room had variable exposure rates, so Microshield calculations were performed separately for six different dose points throughout the room. The results of the six values were combined to give a room estimate of 0.11 kg UO_2.

The sum of the estimated masses in the three rooms, rounded off, gave an estimate of record of 0.11 kg UO_2. The uncertainty assigned to this estimate was ±100%.

Drain Line from Tool Decontamination Facility

The measured exposure-rate data from the piping under the 282-foot level were smoothed using a spline fit, and corrected for background (5.2 R/h) to obtain a net exposure rate for each one-foot segment of the pipe. The Microshield code was run to model a sludge deposit on the inside bottom of the pipe containing ^{137}Cs, producing a dose rate at the detector position. The ^{137}Cs-to-fuel fraction was then used with the Microshield results to give an exposure-rate-to-fuel mass factor of 0.21 R/h per g UO_2, and this factor was then used to convert the fitted next exposure rate value for each pipe segment to a mass of fuel for the segment. The masses were then summed to give a value of 2475 g UO_2 in the measured segment of the drain line.

Measurements were not taken in the 16-foot stretch of pipe running under the 347-foot level, but engineering analysis of the flow and sedimentation characteristics in the drain piping concluded that the results of the measurements of the first 16-foot stretch in the 282-foot pipe could be applied to the 347-foot piping. Since 1944 g fuel were measured in the first 16 feet of the 282-foot pipe, 1944 g could be used as an upper estimate of the fuel in the 347-foot run.

The sum of the two values gave an estimate of record of 4.4 kg UO_2, with an uncertainty of 87%.

Defueling Water Cleanup System

The mass of fuel deposited inside the pumps was estimated by viewing video images to get an estimate of the fuel debris volume. The volume could then be multiplied by the debris density (4.7 g/cm3, a value found in other samples collected outside the RV) and the ratio of UO_2 in the debris (0.72). Video inspections were made in only one pump, with the assumption that fuel deposition was identical in both pumps. 0.86 kg fuel was estimated to reside on the horizontal surfaces of both pump transition areas. 0.78 kg per pump was estimated in the two other areas containing fuel. Thus the total quantity of fuel estimated in the pumps was 2.4 kg, and the corresponding uncertainty was 100%.

For measuring the manifold using gross gamma measurements, Microshield calculations were performed that allowed the conversion of exposure rate to mass of fuel. The modeling used simple geometric approximations of lines and points for the radioactive sources and simple slabs for the shields. The most probable shielding and geometry model gave an estimated fuel quantity of 1.1 kg UO_2, with a 50% uncertainty.

For estimating the fuel on the DWCS hoses, gamma spectral measurements were made on one-foot segments. Using 2-inch diameter hoses, a one foot segment gave an estimated mass of 0.015 g. This hose segment corresponded to 486 cm^2 of inside surface area. The total collection of hoses, with diameters ranging from .5 to 4 inches, had a calculated total inner surface area of 614,986 in^2 (about 3,968,000 cm^2). Thus, assuming that the fuel concentration measured inside the measured hoses was representative of all hoses, a mass of 0.122 kg was estimated inside the hoses.

A second estimate was made of the fuel in the hoses by performing a gross gamma exposure rate measurement and using a gamma shielding code to convert the reading into mass of fuel. This estimate was 0.295 kg. For the estimate of record, the PDSR averaged these two values, to arrive at 0.21 kg. After this estimate was made, sections of hose were moved to the FTC for storage, and the quantity of fuel contained in the moved hoses were included in the total with the FTC, as discussed above. This quantity, 0.012 kg, was subtracted from the 0.21 kg, to give a final estimate of 0.20 kg UO2 ±100%.

The final estimate of record for the DWCS is the sum of the three components (2.4 + 1.1 + 0.2), or 3.7 kg. The uncertainty was derived from a combination of the three other uncertainties, ±67%.

Temporary Reactor Vessel Filter System

The analysis assumed that the fuel loading measured on the analyzed samples of diatomaceous earth were applicable to both filters. It applied these values to the entire filter contents and estimated 3.80 kg of UO_2 in one filter and 0.59 kg in the other, for a total of 4.4 kg UO_2. The uncertainty assigned to this estimate of record is ±90%.

Incore Guide Tubes

A ^{144}Ce-to-fuel ration of 152.5 μCi/g on 8/1/87 was decayed to each measurement date to use as the basis for the estimation of fuel in the IIGTs. The gamma shielding codes Microshield and QAD were used to model the measurement setups, assuming that the source region (the inside of each IIGT) was uniformly filled with fuel. The combination of measurement data and modeling gave a total fuel mass of 23.8 kg in the IIGTs that were assessed, with 1.42 kg fuel found as the highest quantity in any one guide tube. After the measurements, two of the IIGTs were cut out of the surrounding plates and shipped offsite, so fuel had to be deducted from the total estimate to account for these two. The maximum value of 1.42 kg was assigned to each of these tubes, so 2.8 kg was subtracted from the 23.8 kg figure to give a final estimate of 21 kg UO_2, with an uncertainty of ±54%.

Defueling Tools

The gamma spectrometry performed on the first five tools determined ^{144}Ce concentrations on the tools with the help of Microshield modeling. The ^{144}Ce-to-fuel ratio was then used to arrive at an estimate of 61 grams UO_2 ±60% for the five measured tools.

After the fuel concentration was determined for the first five tools, a gross gamma exposure rate measurement was performed on these five, and the measured exposure rate was combined by the fuel quantity to get a factor of 0.094 R/h per g UO_2. Gross gamma exposure rate measurements were then performed on 69 additional tools, and the factor of .094 was applied to the measured exposure rate to get a total fuel mass of 318 g UO_2 ±70% for these 69 measured tools.

Due to radiological restrictions, 43 additional tools could not be measured, so the average fuel mass per tool, 5.1 g, derived from the 74 measured tools, was applied to the unmeasured tools. This gave a fuel estimate of 220 g ±85% for the final group of 43.

The estimate of record for the 117 defueling tools is the sum of the three groups above, or 0.6 kg UO_2. The associated uncertainty is ±75%.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

Reactor Coolant Drain Tank

The estimate of fuel remaining in the tank relied on some gross assumptions. The debris thickness was estimated based on video inspections taken on a small part of the tank, and the conditions were assumed to be identical over the entire tank interior. The mass of

debris and the fuel content both relied on factors derived from the laboratory analysis of debris samples, and it requires an assumption that the samples were collected from a sufficiently large number of locations to ensure that the samples represented debris in the entire tank. The derived mass was small, however (0.1 kg fuel) and the assigned uncertainties were large enough to ensure that any resulting inaccuracies would be insignificant.

Fuel Transfer Canal/Transfer Tubes

The estimate of fuel mass in the FTC was based on several things, including gamma measurements, video inspection, and engineering judgement. The measurement of fuel mass dropped from transferred canisters used ^{137}Cs as a tracer for fuel, and it used a Minimum Detectable Level argument. Since the measurements were made before all the canisters were transferred through the FTC, they had to assume that similar quantities would be dropped by the additional canisters and ratio their estimate upward accordingly. A major part of the estimate for this fuel quantity was based on informed judgement, making estimates of how much fuel would relocate from one reactor component to another as water is drained in and out of the RV. One is tempted to use the term "guessing" to describe the method of estimation, but in fact the guessing was guided by knowledge of how much fuel currently resided in each component, of what the water velocities would be, and what routes relocated fuel might follow. While the methods of estimation were definitely inferior to actual measurement, the estimates had to be made before measurements were possible, and the study used the best methods available. Appropriately large uncertainty bounds were assigned.

One matter for concern is that the study estimated relocation of fuel out of components such as the cold leg and the OTSGs. The estimate of fuel relocating out of one component could lead to double counting. The quantities affected are small.

Core Flood System

The techniques used in measuring these components are similar to those used in other fuel assessment efforts. Some assumptions had to be made about the uniformity of fuel distribution inside the pipes, and the POSR report states that the fuel film concentration in the core flood system is probably lower than the value used, which was determined inside the RCS. The uncertainty bound of 77% is a reasonable means of covering the estimates.

Upper Endfittings

The neutron interrogation technique is a useful technique for measuring uranium, particularly because of the usual absence of interfering sources of background neutrons. In this case the signal neutrons gave very low count rates, which produced a large error bound around the measured values.

The application of the average mass per fitting for the five measured containers to the sixth, unmeasured container should have resulted in a large uncertainty. The mass per fitting varied from 0.06 kg to 0.78 kg in the first five casks, giving a standard deviation of 98% in the five values of fuel mass per fitting. Thus there should be a larger error bound on the mass of fuel in the sixth cask.

Tool Decontamination Facility

The general method of measuring a dose rate, then modeling the deposited quantity of fuel that produced the dose rate, is a standard technique that has been successfully applied in a number of PDSRs. The application to this analysis appears to be correctly done.

Drain Line from the Tool Decontamination Facility

This estimate was also based on gamma measurements and Microshield modeling. The measurements were difficult because of the very high background dose rates and the variability of the detector readings as the detector progressed down the pipe. There was some speculation about the possibility that the detector position was not accurately known for each measurement position (the polyethylene tube may have been "snaking" rather than laying flat), but this uncertainty should have averaged out, especially with the spline fitting.

The analysis had to assume that there was no additional fuel in the unmeasured segments of piping under the 282-foot level. This assumption was based on assumptions about the movement of the fuel in the pipe, and it was confirmed by the fact that measured exposure rates were high at the beginning of the pipe but dropped off to background rates as the GM tube moved down the pipe. The analysis also had to assume that the deposition of fuel in the piping under the 347-foot level matched the deposition in the 282-foot pipe. There is no evidence to the contrary, but of course there is no confirming measurement. The low quantities of fuel involved do not justify a confirmatory measurement, however.

Defueling Water Cleanup System

The estimates of fuel inside the pumps, found using video inspection, used a technique that is used in other parts of the plant. The estimate of debris volume is probably good. The application of a fuel debris density value and fuel concentration fraction, however, are based on samples that did not come from inside the pump. The licensee had to assume that these values were indeed applicable to the debris that was seen inside the pump. This analysis also assumed that the two pumps were identical in terms of UO_2 contamination, since inspections were only performed inside one pump.

The gross gamma measurements of the manifold were performed with the detector at a distance of about 24 feet from the manifold. Thus the collimator, with a 90° field of view, accepted radiation from a much larger area than just that of the manifold. The analysis assumed that the radiation, above background, that was detected by the probe was emitted only by the manifold and not by other radiation sources seen by the detector. Another weakness of the study is the simple geometry used in the modeling of detected radiation. The large error bounds in the estimate should account for these assumptions.

The measurement of fuel in the hoses used two different approaches. The first approach, using gamma spectrometry on short hose samples, relied on the assumption that the measured segments were representative of all the hoses. The second approach relied on a somewhat cruder measurement system. The two estimated values were 0.122 kg and 0.295 kg, which justified the 100% uncertainty.

Temporary Reactor Vessel Filter System

The major assumption in this analysis was the applicability of the measured sample of diatomaceous earth to the entire contents of the filter. This was not an unreasonable assumption, so the analysis should be valid.

Incore Guide Tubes

The use of gamma spectrometry to measure the contained ^{144}Ce, and the subsequent modeling with Microshield and QAD, is a technique that has been used in other studies and seems to have been properly applied here. The error bound of 54% is sufficiently large to cover the measurement and analysis uncertainty.

Defueling Tools

The scheme developed for these measurements was a good one. The use of gamma spectrometry should give a better estimate of the fuel concentration on the contaminated tools than gross gamma measurements, so they wisely evaluated the first five tools using this technique. These evaluated tools could then be used as a calibration for the gross gamma measurements, which could then be applied to the other tools. The application of the results of these measurements to the unmeasured tools required an assumption that there was no unusually high contamination on the unmeasured ones, but such an assumption is not unreasonable.

SUMMARY AND CONCLUSIONS

The listing of components and associated masses of UO_2 is given below:

Incore Guide Tubes (in "A" D-Ring)	21.0 kg
Fuel Transfer Canal/Transfer Tubes	18.9 kg
Upper Endfittings	5.9 kg
Core Flood System	4.9 kg
Drain Line from Tool Decon Facility	4.4 kg
Temporary RV Filter System	4.4 kg
Defueling Water Cleanup System	3.7 kg
Other Components	0.8 kg
TOTAL	64.0 kg UO_2

Several different techniques were used in these analyses, including gamma spectrometry, gross gamma measurements, neutron interrogation, video inspection, and laboratory analyses. The techniques were appropriately applied - good judgement was shown in selecting the analysis technique that was appropriate for each situation.

In some cases, measurements could not be taken because of physical constraints such as inaccessibility or severe radiological conditions. In these cases, the results of similar measurements were applied to give fuel estimates. In the evaluation of fuel in the Fuel Transfer Canal, the estimate had to be made before all the fuel was transferred into the canal, and this is probably the most speculative estimation in this POSR. An appropriately large uncertainty was assigned to the estimate to reflect the speculation, however.

The estimate of 64.0 kg UO_2 is assigned an uncertainty of ±26.9 kg. The fuel estimates determined in this study account for 37% of the fuel that was found outside the RV.

REFERENCES

GPU Nuclear. June 18, 1991. IMI-2 Post-Defueling Survey Report for the Reactor Building Miscellaneous Components.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE REACTOR COOLANT SYSTEM

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1991) discusses the process for estimating the amount of UO_2 remaining in the Reactor Coolant System (RCS). The quantity of fuel estimated to be present in the RCS is 25.8 kg, with an uncertainty of ±43%.

The major components of the RCS were analyzed using the following techniques:

- video inspection;
- gamma spectrometry;
- gross gamma exposure measurements;
- laboratory analyses; and
- engineering analyses.

The components of the RCS that are addressed in this PDSR include the following:

1) Reactor Coolant pumps;
2) core flood lines;
3) 4 cold legs;
4) 2 hot legs;
5) decay heat drop line;
6) pressurizer lines; and
7) surface films.

A number of major components of the RCS were addressed in other PDSRs, so they are not included in this report. These include the core flood tanks, cold leg and hot leg nozzles, steam generators, pressurizer, RC drain tank, letdown coolers and the Reactor Vessel.

MEASUREMENT METHODOLOGY

Reactor Coolant Pumps

The quantity of fuel remaining in the Reactor Coolant pumps was estimated by the inspection of video recordings taken inside the pump casings. The video inspections used were made in only one pump, RC-P-2A, to determine the volume of sediment in the pump. The quantity of fuel in this pump was then estimated by correlating the debris volume with the mass and volume of a fuel sample from in the associated cold leg to obtain the fuel mass in the pump. Fuel masses in the other pumps were then found by ratioing the fuel estimates in their associated cold legs to the fuel found in pump RC-P-2A.

Core Flood Lines

The fuel remaining in the core flood lines was estimated using a GM tube to measure gross gamma exposure rates inside the lines. The GM detectors were positioned inside the pipe, about two cm above the bottom of the pipe. It was used to make dose rate measurements in one-foot increments over an eight-foot length of the pipe on the A side, and was then placed inside the B pipe to again make measurements over an eight-foot length at one-foot intervals. These measured dose rates could then be used in conjunction with Microshield modeling to estimate the fuel masses.

4 Cold Legs

The estimate of fuel in the cold legs was performed by making gross gamma measurements inside the cold leg piping with a GM tube. The GM tube was inserted inside the pipe and used to measure dose rates at one-foot intervals over the 25-foot length of each cold leg. Video inspections were also made inside the pipes to assess the sediment layer on the bottom of the pipes. These images were used to develop the modeling of the source region in the Microshield code.

2 Hot Legs

The estimation of fuel remaining in the hot legs used techniques that were very similar to the study of fuel in the cold legs. A GM tube was inserted into the hot leg piping and used to make dose rate measurements at a number of intervals inside the pipe. These readings were used with computer modeling to estimate the fuel quantity in the pipes.

Decay Heat Drop Line

Fuel remaining in the decay heat drop line was measured using a GM tube inside the pipe, as in other studies in this PDSR. Gross gamma exposure rates were measured over a three-foot section of piping, and used with computer modeling to determine the quantity of fuel in the pipe. This modeling effort used samples of debris that were extracted from the pipe being measured to ensure a more accurate modeling of the fuel deposits.

Pressurizer Lines

A portable directional gamma survey meter (HP-220A) was used on the outside of the surge line piping to detect the presence of fuel in the piping. For the north-south surge line section, the measured dose rates were not elevated above background, so it was assumed that no residual fuel was present in this line. The eastward section of the surge line gave positive readings, so it was studied for fuel content. The HP-220A detector was exposed to the pipe at a number of discrete locations and the dose rates were recorded. Two-inch thick lead bricks were used to provide additional shielding for the meter to lower the background from gammas emitted from the pressurizer.

Surface Films

Three independent measurement methods were used to analyze the surface films that adhere to the internal surfaces of the RCS. Radiochemistry was performed on scrape samples to determine the uranium content, direct alpha counting was performed to measure alphas emitted by the fuel, and gamma spectrometry was used to measure the ^{144}Ce and ^{154}Eu analogs for fuel. Of these methods, the radiochemistry was considered the most reliable. The analysis of the scrape samples led to an average film thickness and fuel concentration that could be applied to the entire inner surface of the RCS to give a fuel estimate.

ANALYSIS METHODOLOGY

Reactor Coolant Pumps

The video inspection of pump RC-P-2A gave an estimate for the debris depth of 0.25-inch. Applying this depth to the area of the sediments found on the pump's internal surfaces gave a debris volume of 11.7 liters. The analysis assumed that the debris inside this pump would have the same density and uranium concentration as the debris found in the associated cold leg piping. The volume of debris found in Cold Leg 2A was 21.6 liters, and the associated fuel mass was 3.9 kg UO_2. Multiplying the ratio of 11.7/21.6 by 3.9 kg gave an estimate of 2.1 kg UO_2 for the inside of pump RC-P-2A.

The fuel contents of the other three pumps were found by assuming that the fuel content of the corresponding cold leg was also directly proportional to the mass of fuel in the pump. Thus fuel masses for the remaining pumps were found by multiplying the associated cold leg's fuel mass by (2.1/3.9), which is the ratio of the fuel in pump RC-P-2A to the mass of fuel in cold leg 2A. The fuel masses assigned to the four pumps are listed below.

RC-P-1A	1.8 kg UO_2
RC-P-1B	1.0 kg UO_2
RC-P-2A	2.1 kg UO_2
RC-P-2B	1.3 kg UO_2
TOTAL FUEL	6.2 kg UO_2

The uncertainty associated with this mass estimate is ±55%.

Core Flood Lines

A one-foot segment of fuel debris on the inside of a pipe was modeled in Microshield to arrive at the exposure rate that would result from the ^{137}Cs in a standard amount of fuel in the debris. The density and composition of the debris were chosen to match samples retrieved from the RV lower head. These calculated values were then used in conjunc-

tion with the dose rates measured by the GM tubes to estimate the fuel remaining in the pipes. The fuel quantities estimated in the core flood lines were 0.6 kg for the A side (CF-A), and 0.4 kg for the CF-B line. The total fuel was 1.0 kg UO_2, with an associated uncertainty of ±55%.

4 Cold Legs

The measured dose rates inside the cold legs were used in conjunction with Microshield calculations to arrive at the fuel estimate. In the computer model the source region was assumed to cover the bottom 120° of the inside of the piping. Calculated dose rates resulting from standard amounts of fuel were then compared to measured dose rates to arrive at fuel quantity estimates. The study concluded that the fuel contents were as listed below.

Cold Leg 1A	3.3 kg
Cold Leg 1B	1.8 kg
Cold Leg 2A	3.9 kg
Cold Leg 2B	2.4 kg
TOTAL FUEL	11.4 kg

Note that these estimates were also used to estimate the quantity of fuel in the associated reactor pumps, as described above.

Following the measurements, water was moved through the 1B cold leg in order to drain the RV, perform neutron measurements in the RV, and drain the once-through steam generators (OTSGs). It was assumed that this water flow would drain all the fuel-bearing sediment out of the 1B cold leg. Thus the fuel estimate for the 1B cold leg was reduced to 0, so the estimate of record for the cold legs is thus 1.8 kg less than shown above. The estimate of record for the cold legs is 9.6 kg UO_2, with an uncertainty of 55%.

2 Hot Legs

Microshield modeling was used along with the readings inside the hot leg piping to estimate the fuel remaining in these pipes. The techniques were very similar to the methods used for estimating the fuel in the cold leg. The study determined that 0.9 kg fuel remained in the A hot leg, while 1.8 kg remains in the B hot leg. Thus the estimate of record is 2.7 kg UO_2, with an uncertainty of ±55%.

Decay Heat Drop Line

The gamma dose rate measurements were taken in a three-foot-long section of the 12-inch-diameter pipe. Microshield was then used to model the fuel deposits that would cause these dose rates. For this study,

samples of debris taken from the decay heat drop line were analyzed and found to contain 8% UO_2 by weight. This fraction was used in the modeling to properly handle the gamma emission and attenuation characteristics of the fuel debris. Although measurements were taken over only three feet of the piping, the results were applied to an additional 15-foot length of piping that had not been measured but contained fuel debris. Thus the estimate of record for this 18-foot length of piping is 1.5 kg UO_2, with an uncertainty of ±55%.

Pressurizer Lines

Engineering analysis, including a review of the water flows through piping sections during the accident and during subsequent cleanup efforts, determined that no fuel should remain in the pressurizer spray line, or the safety and relief lines connected to the top of the pressurizer. Similarly, inspections of the drain lines to the Reactor Coolant drain tank showed no significant fuel deposits. Thus the only lines connected to the pressurizer that were surveyed for fuel deposits were the surge lines. The north-south lines showed no gamma readings above background, so it was assumed that no fuel deposits were there.

The eastward segment of the pressurizer surge lines were checked with a HP-220A directional survey meter at a number of discrete locations on the outside of the pipe. Using an assumed uniform deposition pattern inside the pipe, the total fuel content was estimated to be 0.2 kg UO_2. The study did not explicitly assess the uncertainty of this measurement, so the PDSR assigned an uncertainty value of ±60%.

Surface Films

Surface films were collected from a number of locations inside the RCS, including the inspection port cover plates in the pressurizer and in the OTSG manway. The nature of the film samples varied, both in color and thickness, throughout the RCS. Based on these samples, an average film thickness of 0.0019-inch was assigned to the RCS inner surfaces. The radiochemical analyses of the films led the study to assign an average fuel content of 18 $\mu g/cm^2$ of UO_2. This concentration was then applied to the entire area of the inner surfaces of the RCS to give a total fuel estimate of 4.6 kg UO_2. The uncertainty for this estimate was ±60%.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

Reactor Coolant Pumps

The estimates of fuel in these units were based on minimal measurements and a large amount of ratioing. Video inspections were evaluated to find the volume of the debris for only one pump - fuel contents of the other pumps were found strictly by analogy.

The estimate assumes that the nature of the debris inside the pump (density and uranium content) is identical to the debris in the cold legs. It also assumes that the quantity of debris inside each pump can be found by measuring the debris in the associated cold leg. The engineering estimate for this analysis thus depended on some major assumptions.

Core Flood Lines

This analysis relied on gross gamma measurements coupled with computer modeling. The critical parts of the modeling were the choice of source geometry and the assumed cesium-to-fuel ratio. These methods have been used successfully however, and appeared to be properly applied here.

4 Cold Legs

The assessment of fuel in these areas were again performed using gamma exposure rate measurements and Microshield modeling. This assessment was enhanced by the use of video inspection to make sure that the modeling of the fuel deposits were accurate. This estimate relied on a major assumption when it subtracted the fuel from the 1B cold leg due to anticipated flushing during the draining procedures. No measurements were made to check up on this assumption. However, mass subtracted from this region was assigned to other regions, such as the fuel transfer canal, so the entire plant's special nuclear material accountability should be unaffected by this assumption.

2 Hot Legs

The assessment of fuel in this piping was performed using the gross gamma measurements and Microshield modeling, as in other studies. The techniques appeared to be properly applied and the uncertainty bound of 55% seems appropriate.

Decay Heat Drop Line

Again, gross gamma exposure rate measurements were made inside the piping and used with Microshield calculations to arrive at an estimate. Since measurements were made in only a three-foot section of pipe but applied to an 18-foot length, the analysis relied on the assumption that the deposition was uniform in the entire 18 feet. This study had the advantage of using a debris sample from the measured pipe itself for the computer model. The fuel estimate and the associated uncertainty appear to be appropriately chosen.

Pressurizer Lines

The dose rate measurements performed on these lines were done with a survey instrument held on the outside of the pipe. This technique would be less accurate than measurements performed inside the pipe, since gamma attenuation by the pipe wall is important, and background radiation from other parts of the plant are more significant. Since the

estimated quantity of the fuel is small, however, the additional uncertainty should not be important.

Surface Films

The radiochemical analysis of scrape samples is an appropriate method of determining the isotopic content of the films. The films showed a great deal of variability in thickness and color from place to place in the RCS, so it would seem to be a significant assumption to assign an average fuel concentration per unit area to the entire inner surface of the RCS. The number of samples taken also seems small compared to the large surface area that the concentration is assumed to cover. It is conceivable that some surfaces have no surface film, and others perhaps have a higher concentration than assumed. The uncertainty interval of 60% should be large enough to cover these uncertainties, however.

SUMMARY AND CONCLUSIONS

The components analyzed in this PDSR and the associated fuel masses and uncertainties are given below.

Reactor Coolant Pumps	6.2 kg UO_2 ±55%
Core Flood Lines	1.0 kg UO_2 ±55%
4 Cold Legs	9.6 kg UO_2 ±55%
2 Hot Legs	2.7 kg UO_2 ±55%
Decay Heat Drop Line	1.5 kg UO_2 ±55%
Pressurizer Lines	0.2 kg UO_2 ±60%
Surface Films	4.6 kg UO_2 ±60%

Based on these estimates, the estimate of record for these RCS components is 25.8 kg UO_2. The uncertainty that they assign to this estimate is ±43%, which was derived from the individual uncertainty estimates.

Several different techniques were used to arrive at these estimates, including gross gamma measurements, radiochemical analysis, video inspection and engineering analysis. The technique most commonly used was a gross gamma exposure rate measurement performed inside a pipe, used with computer modeling of the fuel deposits. The methods were appropriately applied and the estimate of record, 25.8 kg UO_2 ±43%, appears to be reliable.

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GPU Nuclear. July 3, 1991. TMI-2 Post-Defueling Survey Report for the Reactor Coolant System.

REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE REACTOR VESSEL

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report (PDSR) (GPU Nuclear, 1993) discusses the process for estimating the amount of UO_2 remaining in the TMI-2 Reactor Vessel (RV). The quantity of fuel estimated to be present in the RV is 925 kg, with a one-sigma uncertainty of 40%.

This study initially used a video inspection and analysis study to arrive at a fuel estimate of 630 kg. When it became apparent that this video estimate was not sufficiently accurate, a method of passive neutron measurements was adapted. These measurements were performed while the Reactor was being drained of water, and the water draining was used to isolate the RV into nine individual zones that could be analyzed separately. The analysis of these measurements led to an estimate of 1322 kg. A review committee was asked to review the data from the neutron measurement study, and this committee recommended several corrections to the measurements. The application of these corrections resulted in the final estimate of record of 925 kg.

MEASUREMENT METHODOLOGY

The measurement of residual fuel remaining in the RV was originally performed using video interpretation. The internal regions of the RV were thoroughly surveyed using video cameras while the RV was full of water, and the video images were analyzed by engineers to determine the volume of fuel deposits. Although the video estimates did not result in the final estimate of record, the recorded images and analyses were important in guiding the neutron measurements that were performed later.

Passive neutron measurements during RV draindown were used to arrive at the estimate of record. The measurements used three ^4He neutron detectors to measure neutrons emitted by residual fuel. Since the gamma radiation field was very high in the RV, it was necessary to provide gamma shielding around the detectors. Thus a shield employing 15-cm of lead and steel was fabricated to surround the detectors. The resulting detector assembly, consisting of shields, detectors, and electronics, weighed approximately two tons and had to be handled by crane.

The passive neutron measurements were performed while the RV was being drained of water. The water would be drained down to a previously defined level, and held at this level while the detectors counted neutrons. This technique isolated the detectors, to a large degree, from neutrons emitted by fuel below the water level. The method of incrementally lowering the water level thus allowed the neutron measurements to be made in a number of individual zones. Nine zones were chosen to conveniently handle the measurements. Since it was important to know exactly where the water level was in order to properly identify the zones, a staged precision bubbler was employed that could read out the water level accurate to within 0.23 inches.

At each of the nine measurement locations, the detector was first operated with the water level at the bottom of the measurement zone. Count times were long, typically more than a day, to collect a significant number of counts. The neutron source was then lowered into position and the detectors operated again to ensure correct neutron detection. The neutron source was then raised out of the RV, and the RV water level was lowered to the bottom of the next zone, and the detector again operated. Finally, the neutron source was returned to position near the detector and the detector again operated for a source check. This procedure provided two sets of counts in each zone: the first with the water level at the bottom of the zone, and the second with the water level at the bottom of the next lower zone. This counting scheme provided a means for subtracting out the effects of fuel in other zones. After these four measurements, the detector was lowered to the counting position in the next zone and the cycle of four measurements were repeated. This procedure was repeated for all nine zones, and the count data were used to determine the fuel quantities.

ANALYSIS METHODOLOGY

The video images were analyzed by engineers who estimated the volume of fuel deposits that were identified in the video. The inspections generally arrived at good estimates for two of the dimensional sizes of each deposit, but the depth was usually difficult to determine from the inspections. Conservative estimates of depth were used. The appearance of the deposits were then compared to samples that had been analyzed, and similar samples were used to convert the observed deposits to a mass of UO_2. The sample's density and uranium content were used to make this determination. The final result of the video inspections was an estimate of 609 kg of UO_2 remaining in the reactor vessel. An additional 21 kg were estimated to reside in the hot and cold leg nozzles, so this quantity was added in to give an estimate of 630 kg of fuel in the RV.

The video assays had several limitations, including difficulty of access into some areas of the RV that could contain fuel, resolution of the video cameras operating underwater, and the difficulty of using two-dimensional images to derive three-dimensional volumes. Thus no estimate of the uncertainty was made for this estimate, and a neutron measurement technique was subsequently used to arrive at the estimate of record.

Extensive analysis was required to convert the recorded detector count data to the fuel mass for each region. The analysis had to account for the fact that the counts recorded at any detector location were due to neutrons emitted from fuel in the measurement zone and neutruns emitted from fuel in the zones above the detector. In order to correct for this effect, two counts were taken with the detector at each location: one with the water level at the bottom of the measured zone, and one with the water level at the buttom of the next lower zone. All of these counts gave a set of coupled equations that could then be solved to find the nine count rates, each resulting from the neutrons emitted by fuel in the appropriate measurement zone.

After corrected zone count rates were determined, the computer codes Microshield and QADGP were used to model a zone's fuel deposits and intervening shielding, then estimate the number of neutrons that would reach the detector. At first glance it would seem to be a poor application of software, using a gamma shielding code to model the transport of neutrons. These tools were chosen as an easy means of calculating exponential attenuation. A neutron attenuation study had been performed by placing varying thicknesses of steel and lead between the AmBe neutron source and the detectors. The study found that the change in neutron count rate caused by both the lead and the steel followed an exponential relationship, identical to the exponential attenuation of gamma radiation. Thus, by carefully choosing the attenuation factor in the computer codes, the codes could be used to predict the neutron count rates caused by modeled amounts of fuel and shielding. The advantage of this technique over traditional neutron transport codes is the simplicity of code input and speed of execution. The disadvantage of this technique is the omission of neutron scattering effects.

The computer modeling for each zone started with modeling the fuel deposits that were identified during the video inspection and using this model to calculate the count rate. In all nine cases this modeling predicted a lower count rate than actually measured. In order to increase the modeled count rate for a given zone, fuel was added to the model until the predicted count rate reached the measured value. Two different schemes were used for adding the additional fuel: placing the additional fuel in a credible location close to the detector, and placing it in a credible location far from the detector. Placing fuel near the detector would require a small amount of additional fuel to meet the measured count rate, whereas placing the fuel far from the detector would require a larger additional quantity. Thus a low estimate for fuel in a given zone was determined by summing the video estimate plus the fuel added close to the detectors, and a high estimate was found as the sum of the video estimate plus the fuel added far from the detectors. The initial estimate of record was taken as the average of the low and high estimates. The results of this initial analysis are given in Table 1.

Table 1. Fuel Mass per RV Zone, Initially Estimated by Passive Neutron Measurements

Zone Number	Estimated Fuel Mass (kg)
1	10
2	225
3	150
4	99
5	154
6	387
7	113
8	89
9	95
RV TOTAL	1322

The modeling study's conversion of fuel quantities to neutron count rates depended on an assumed value for the neutron emission rate from fuel deposits. The licensee determined this value experimentally, by counting the neutrons emitted by fuel debris samples extracted from the core. The value of 0.66 n/s per g UO_2 was derived from these measurements and used to find the estimate of record. An uncertainty of ±6% was assigned to this value.

The passive neutron measurements were reviewed by a distinguished committee, and this review identified five biases that affected the initial analysis. These biases include:

Boron Variations

The committee felt that the residual fuel in the RV contained more boron than the fuel in the samples analyzed for the neutron emission rate study. This means that the residual fuel emits more neutrons than assumed by the licensee's study, so the initial measurement study overestimated the fuel in the RV by 15% in zones 1-5. (The committee report's convention was to label a bias as positive if it resulted in a high estimate, so this was a bias of +15%.)

UO_2 Particle Size

The particle size of fuel debris in the RV was probably finer than the samples used in the neutron emission study, also affecting neutron yield. The review estimated that this biased the initial study by +45% in zones 1-5.

Calibration Error

The committee felt that using the AmBe calibration source caused a 5% under-estimate in all nine zones. This bias was due to the imperfect match of energies between the AmBe source neutrons and neutrons emitted by the fuel.

Data Analysis

When the neutron emission rate study was performed, the licensee omitted the data from one of the fuel samples because the uranium fuel content of the sample looked unreliable. The review committee felt that this sample's result should not have been rejected, and that the rejection caused a +5% bias in all nine zones.

Neutron In-scattering

The modeling performed for estimating the fuel distribution did not account for neutrons emitted by the fuel that reached the detectors by scattering. This effect caused a 20% bias in all nine zones.

The numerical biases identified by the review committee were assigned to the fuel masses shown in Table 1 on a zone-by-zone basis to arrive at the estimate of record. The correction factor derived from these biases was 80% for zones

1-5 and 20% for zones 6-9. Table 2 shows the fuel masses in the estimate of record.

Table 2. Estimate of Record for the Reactor Vessel

Zone Number	Estimated Fuel Mass (kg)
1	6
2	125
3	83
4	55
5	86
6	323
7	94
8	74
9	79
RV TOTAL	925

For the initial measurement estimate, the licensee had assigned an uncertainty of ±15%. Upon the review committee's recommendation, the final uncertainty was ±40%. This uncertainty was based on the difficulty of accounting for neutrons scattered in the RV, the variety of forms for the debris, the difficulty in modeling the neutron scatter and absorption in the complex steel structures, and the uncertainty in the quantity of boron during the measurements.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

The use of the video inspection and the careful analysis of the videotapes were important contributors to understanding the distribution of fuel in the RV and for guiding the neutron measurement study. The visual estimate was also valuable as a "ballpark" figure for guiding the study: when the passive neutron measurement study gave a result more than double the visual estimate, this was a signal to search for biases in the measurements.

The passive neutron measurement scheme was a good one for developing a more accurate estimate. The hardware and counting scheme were well designed, and the use of the RV draindown to divide the RV into measurement zones was an excellent strategy. Performing two counts at each location with the water level at two different positions, then solving the system of coupled equations for each zone's effective count rate was also good procedure. The detectors were allowed to count at each location for long time periods to ensure statistically significant data collection.

The use of gamma shielding codes for modeling fuel distribution was one weakness of this analysis. It is true that neutron attenuation by shielding follows an exponential relationship, which can be calculated in a gamma shielding code, but neutron scatter does not behave in this manner and could not be modeled by the codes used in this study. Using a neutron transport code such as MCNP would have been a more difficult modeling task, but it would have avoided many of the problems caused by the resulting rough treatment of scattering effects.

The determination of neutron emission rates was another weak point in the analysis, but it was probably a weakness that could not be easily resolved. The method of measuring the neutron emission rates from the fuel samples was the best possible solution, but it was essentially impossible to find samples that would be representative of the residual RV fuel in terms of particle size and boron content. On the other hand, the biases that were identified and quantified by the review committee to account for these effects are appropriate. One shortcoming of the licensee's neutron emission study was the assignment of a very small uncertainty, ±6% to the measured emission rate: one of the critical components of this figure was the uranium mass in the samples, and these values were no more accurate than 15%. An uncertainty in the neutron emission rate of 25% would have been more appropriate.

Most of the biases identified by the review committee were pertinent and reasonably quantified, but we disagree with the "Data Analysis" bias. The data point that was rejected by the licensee did look suspicious, both in terms of the consistency of its mass with respect to the other eight samples, and in terms of the measured count rate. The rejection of this sample was justifiable. The bias, of course, was only 5%, so it was a negligible effect.

SUMMARY AND CONCLUSIONS

The estimate of record of fuel remaining in the TMI-2 RV is 925 kg UO_2, with a one-sigma error bar of ±40%. This estimate was determined using video inspection and passive neutron measurements that were performed while the RV was in a gradual, incremental, water draindown. The count data were used with computer modeling to determine the quantity of fuel in the reactor vessel. A subsequent review by the review committee helped identify biases in the original study and correct for the biases to arrive at a more realistic figure.

The measurement scheme was appropriate for the problem being solved and the conduct of the measurements displayed a good degree of engineering skill and ingenuity. Some parts of the analysis were weak, particularly the use of gamma shielding codes to model the neutron transport. The submission of the study results to a review committee showed good judgement and incorporating the committee's correction factors was appropriate.

The 925 kg of fuel estimated to reside in the RV represents about 84% of the 1097 kg estimated to remain in the TMI-2 facility. The RV is the only TMI-2 location that has enough fuel to warrant a criticality safety study.

Consequently the analysis that is documented in this PDSR is an important one. On the other hand, approximately 94.000 kg UO_2 remained in the RV at the end of the accident, so the estimated residual quantity demonstrates the enormous task that was performed in defueling the RV. It is reasonable to assume that the estimated quantity of residual fuel will not move from the RV for the duration of the Post-Defueling Monitored Storage period.

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REVIEW OF THE TMI-2 POST-DEFUELING SURVEY REPORT FOR THE REACTOR VESSEL HEAD ASSEMBLY

INTRODUCTION

This review of the licensee's Post-Defueling Survey Report on the React Vessel (RV) head (GPU, 1990) discusses the procedures used to estimate the amount of UO_2 remaining in the RV head assembly. The RV head assembly originally covered the RV and served as a containment barrier for the Reactor Coolant. It was removed from the RV in July 1984 and is currently stored on the 347'6" elevation of the Reactor Building, on a head storage stand.

The quantity of fuel estimated to be present in the RV head assembly is 1.3 kg with an uncertainty of +80% and -65%. This estimate was obtained by performing gamma scanning measurements on three leadscrews, combining the data to create a composite figure of a leadscrew and extrapolating the data to the entire head.

MEASUREMENT METHODOLOGY

The RV head assembly consists of:

1) 69 leadscrew motor housings that extend from the thermal barrier into the underhead area;
2) 69 leadscrew tubes (LSTs);
3) 69 leadscrews (61 are connected to control rods and eight are connected to axial power shaping rods) that extend from the thermal barrier to the tip of their bayonets;
4) a dome; and
5) a flange.

The RV head assembly sits on a head storage stand located on the 347'-6" elevation of the Reactor Containment Building. Because of the high dose rates underneath the RV head assembly, direct measurements of the fuel were not made. Instead, the total quantity of fuel was estimated based on radiochemical assays of three of the 69 leadscrews (L9, B8 and H8) that were removed from the RV head assembly and a small section (approximately nine inches) of leadscrew support tube (H8). The leadscrews and the LSTs were radiochemically analyzed by Battelle Columbus Laboratories, Scientific Applications, Inc. (SAI) and Babcox and Wilcox (B&W). A simplified model was used to determine the quantity of fuel remaining, where the RV head assembly surfaces were divided into three categories:

1) leadscrew;
2) inside surfaces (those areas encompassing the leadscrews, the inside of the leadscrew support tubes and the inside of the leadscrew motor housings); and
3) outside surfaces (encompassing the flanges, the dome, the outside of the leadscrew motor housings and the outside of the leadscrew support tubes).

The measurements of the leadscrews were used to estimate the fuel content on the remaining leadscrews. The measurements of the leadscrew support tube were used as the means to relate measured and inferred surface fuel values for the leadscrews to the other surfaces: the exposed underhead surfaces and the hidden or inside underhead surfaces.

Leadscrews

The fuel content of the leadscrews was taken directly from the composite leadscrew. The 61 leadscrews connected to the control rods were fully extended through the plenum during the accident. The eight leadscrews connected to the axial power shaping rods were only extended 75%. Because the E9, B8 and H8 leadscrews that were used for the measurements are control rod leadscrews, using the average fuel amounts for these leadscrews was deemed conservative, thus overestimating the fuel quantity on the axial power shaping rod leadscrews.

The estimate of the quantity of fuel on the leadscrews was determined by calculating the average ^{144}Ce (a known fuel analog) activity on the threaded sections of the leadscrews and dividing by the average cerium-to-fuel (Babel 1988) ratio for the threaded part of the leadscrews. The 66 lead-screws currently in the RV head assembly were estimated to contain 0.39 kg of fuel.

Inside and Outside Surfaces

Because the radiochemical analysis that was performed on the leadscrew support tube did not always report ^{144}Ce or ^{154}Eu (the known fuel analogs), the use of a different isotope was chosen. The assumption was made by the licensee that ^{137}Cs activity follows the same trend as ^{144}Ce. The radiochemical analysis of the section of the leadscrew support tube indicated that the outside activity of ^{137}Cs was approximately twice that on the inside. Thus, this ratio was used to determine the quantity of fuel remaining on these surfaces. The effective average fuel area density of 628 $\mu g/cm^2$ was calculated for inside surfaces of the RV head assembly areas. The effective average fuel area density for outside surfaces was calculated to be 1256 $\mu g\ UO_2/cm^2$, since the outside surface activity was conservatively estimated to be twice that of the inside activity.

ANALYSIS METHODOLOGY

The areas of the RV head assembly were calculated to result in a total surface area of 1.44E6 cm^2. The area was determined by approximating the RV head assembly components by geometrical shapes. The areas were than categorized as inside or outside surfaces. Those components that were considered to be inside surfaces included the area encompassing the leadscrews, the inside of the leadscrew support tubes and the inside of the leadscrew motor housings. The outside surface areas include the flanges, the dome, the outside of the leadscrew motor housings and the outside of the LSTs. The estimate of record

of the amount of UO_2 remaining in each of the components was determined by multiplying the total surface area of the component by the corresponding fuel value per square centimeter. The effective average fuel area density of 628 $\mu g\ UO_2/cm^2$ was calculated for inside surfaces of the RV head assembly areas and the effective average fuel area density of 1256 $\mu g\ UO_2/cm^2$ was used for the outside surface activity.

The licensee determined that the possibility of gravel-like deposits being trapped in the RV head assembly components was highly unlikely due to gravity and RV head assembly orientation. Therefore the contribution from this type of deposit was deemed to be insignificant, however, a value of 0.01 kg was assigned.

The estimate of record was determined to be 1.3 kg, based on the following distribution:

Leadscrew Motor Housings	0.12 kg
Leadscrew Support Tubes	0.46 kg
Dome and Flange	0.28 kg
Leadscrews	0.39 kg
Gravel-like Material	0.01 kg
TOTAL	1.3 kg

This estimate was given with an uncertainty of +80% and -65%.

REVIEW OF MEASUREMENTS AND ANALYSIS METHODOLOGY

The measurement methods used to determine the amount of UO_2 remaining on the leadscrews in the RV head assembly were reasonable. The inability to use ^{144}Ce or ^{154}Eu as an analog for fuel on the remaining surfaces of the RV head assembly was however unfortunate, since cesium has never been shown to be a good analog for fuel. It has been shown that cesium tended to dissolve in the Reactor Coolant water and thus to become distributed to a large extent throughout the Reactor Building and parts of the auxiliary and fuel-handling buildings (those areas where the water traveled). Because of the relative solubility of the cesium as opposed to the cerium, it is more likely to overestimate the quantity of fuel in the RV head assembly when used as an analog for the fuel. This is based on the assumption that the cesium in the water would have plated out on the components of the RV head assembly, whereas the fuel not being so readily dissolved, would not have plated out to as great an extent. Another way to validate the use of the cesium as a fuel analog would be to assume that the quantity of fuel is the same as that on the non-threaded section of the composite leadscrew. This value, 215 $\mu g\ UO_2/cm^2$, is substantially smaller than the effective average fuel area density of 628 $\mu g\ UO_2/cm^2$ for the inside surfaces, or 1256 $\mu g\ UO_2/cm^2$ for the outer surfaces.

The analysis methods of determining the surface area of the component and then multiplying by a fuel density number based on measurements of a leadscrew support tube were reasonable. There is no reason to believe that the leadscrew support tube was not typical of the surfaces in the RV head assembly, or that removal of the LST altered the amount of surface contamination on the tube.

The method used to obtain an estimate of the error did not appear to appropriately reflect the uncertainties. A more conservative approach would be to add δx and δy as the upper bound. However, for these values, the exact handling of errors is not crucial.

CONCLUSIONS

The measurement and analysis methods used to determine the amount of UO_2 remaining in the RV head assembly were reasonable, although it would have been preferable to have based the estimate of fuel on the inside and outer surfaces of the RV head assembly on an isotope that showed a link with the fuel, rather than using ^{137}Cs. However, as discussed above, the use of ^{137}Cs in determining fuel quantities appears to be conservative. The method used to obtain an estimate of the error did not appear to appropriately reflect the uncertainties; however, the exact handling of errors is not crucial.

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