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US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Three Mile Island Nuclear Station Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
SNM Accountability

Dear Sir:

By NRC letter dated October 17, 1985, GPU Nuclear was granted exemption from certain requirements for periodic inventory and reporting of the special nuclear materials (SNM) balance for Three Mile Island Unit 2 (TMI-2). As a condition of the exemption, GPU Nuclear was required to conduct an assessment of the SNM remaining at TMI-2 following the completion of the defueling effort. This assessment was referred to in the exemption as the "post-defueling survey." GPU Nuclear letter 4410-88-L-0162 dated September 30, 1988, submitted the initial Post-Defueling Survey Report (PDSR) sub-reports.

As stated in that submittal, the PDSR documents the GPU Nuclear assessment of the amount of residual SNM in the various facilities, systems, and components of the plant and describes the methodology utilized to determine the quantity of SNM in each case. The attached PDSR (Enclosure 1) transmits the post-defueling survey results for the TMI-2 Reactor Vessel (RV). In addition, Enclosure 2 provides replacements for the "Overview/Summary" (properly titled the "Executive Summary"), "SNM Plan Summary," and "Introduction" sections of the PDSR document. The PDSR document is officially complete.

The RV PDSR sub-report completes the final assessment of the quantity of residual SNM at TMI-2 for accountability purposes. A total SNM accounting is ongoing; submittal of the Material Balance Report (DOE/NRC Form 742) is scheduled for April 30, 1993.

A001

The exemption from the annual physical inventory requirements (10 CFR 70.51(d)) granted to GPU Nuclear in the above referenced NRC letter cites no condition or date of expiration of that exemption. Accordingly, since the residual inventory is fixed, as reported in the PDSR, GPU Nuclear concludes that the exemption from 10 CFR 70.51(d) continues in force. Please advise us if that interpretation is incorrect.

Sincerely,

Robert L Long

R. L. Long
Director, Corporate Services/TMI-2

EDS/dlb

Attachment

cc: T. T. Martin - Regional Administrator, Region I
M. T. Masnik - Project Manager, PDNP Directorate
L. H. Thonus - Project Manager, TMI
F. I. Young - Senior Resident Inspector, TMI

ENCLOSURE I

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE REACTOR VESSEL

Approved: J J Byrne 1/18/93
Manager, TMI-2 Engineering

Approved: Robert E Rogan 1/21/93
TMI Licensing Director

TMI-2 INTERNAL REVIEWS

Author: E. D. Schrull 1/18/93

Responsible Technical Reviewer: J. J. Byrne 1/18/93

PDSR Committee Chairman: R. E. Rogan 1/21/93

SUMMARY

The estimate of record of the amount of uranium dioxide (UO_2) remaining in the Reactor Vessel (RV) is 925 kg with an uncertainty of ± 40% expressed as one-standard deviation. The estimate of record is based on underwater video inspections and passive neutron measurements of the residual fuel adjusted to account for measurement bias. The passive neutron measurement technique utilized the results of the video inspections to divide the RV into nine zones which separated the major fuel deposits by elevation. Water was removed from the flooded RV in stages that corresponded to the zones so that the water could be used as a shadow shield to separate the fuel deposits by zone. This technique simplified analysis and improved the accuracy of the passive neutron measurement.

Because of the technical complexities involved in the passive neutron measurement program, including complex geometries, structural interferences, the presence of neutron absorbers and boron-induced (α,n) reactions, a distinguished group of scientists, chaired by Dr. N. Rasmussen, Massachusetts Institute of Technology, and including Dr. H. Menlove, Los Alamos National Laboratory, and Dr. G. Knoll, University of Michigan, was commissioned to conduct an independent review of the measurement data and the estimate of residual fuel. The RV estimate of record was determined by adjusting the passive neutron measurement estimate to consider the measurement biases identified by the Rasmussen Committee.

The total amount of residual UO_2 (i.e., 925 kg) is about 50% larger than the video estimate. The difference between the earlier video estimate and the estimate of record is attributed to the indeterminate uncertainty associated with the visual estimates that were developed from review of the videotapes collected during the video inspection effort.

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE REACTOR VESSEL

1.0 INTRODUCTION

This report presents the analysis of the Three Mile Island Unit 2 (TMI-2) Reactor Vessel (RV) uranium dioxide inventory. It is the final report in a series of reports generated to fulfill the requirements of the TMI-2 SNM Accountability Program. Unless noted, all statistical data uncertainties are expressed in terms of one standard deviation (one sigma or 1σ).

A video inspection of the flooded Reactor Vessel (RV) internals was completed in early 1990. The results of that survey and others were presented in the Defueling Completion Report (DCR) (Reference 1) which was supplemented following the NRC-sponsored Post-Lower Head Sampling Program (Reference 1a). While the fuel estimates were acceptable for the purposes of the DCR, the video inspection of the RV was not considered to be of sufficient quality for final Special Nuclear Material (SNM) accountability. For this reason, passive neutron measurements were performed to provide the final estimate of the quantity of residual fuel within the RV as well as to estimate the uncertainty. Because of the technical complexities involved in the passive neutron measurement program, including complex geometries, structural interferences, the presence of neutron absorbers and boron-induced (α,n) reactions, a distinguished group of scientists, chaired by Dr. N. Rasmussen, Massachusetts Institute of Technology, and including Dr. H. Menlove, Los Alamos National Laboratory, and Dr. G. Knoll, University of Michigan, was commissioned to conduct an independent review of the measurement data and the estimate of residual fuel.

This document outlines the results and uncertainties, as well as the specific measurement and analytical tools used to achieve the estimate of record of the residual fuel in the RV. In this report, residual fuel refers to UO_2 and fuel debris refers to a composite material that was the product of the accident.

Section 2, "Background," describes the accident scenario and subsequent cleanup activities. Also discussed are the location and condition of fuel that was relocated during the accident.

Section 3, "Methods," describes the video analysis, the passive neutron measurement technique and equipment used, and the results of the Rasmussen Committee's review that was incorporated in the estimate of record of residual fuel in the TMI-2 RV.

Section 4, "Analysis," explains in detail how the estimate of record of fuel in the RV was calculated and discusses the supporting data, assumptions made, and the assigned uncertainties.

Section 5, "Conclusion," presents the estimate of record and uncertainty for the amount of residual UO_2 remaining in the RV and supporting rationale leading to a conclusion that the estimate is reasonable based upon the available data and analysis performed.

2.0 BACKGROUND

During the March 1979 accident, the TMI-2 nuclear fuel was overheated due to a loss of coolant. Core temperatures were sufficient to remove the cladding by melting or gross oxidation. This process advanced for about 174 minutes after initial loss of coolant. At this time, the "2B" reactor coolant pump (RCP) operated for a few minutes transporting cold water to the core causing further thermal-mechanical degradation of the cladding and pellets. About 30 minutes later, the Emergency Core Coolant System was used to refill the RV, Reference 1.

At about 225 minutes, instruments indicated, and later analysis revealed, a major core relocation. Visual inspections conducted during defueling indicated a flow of molten core material to the Lower Core Support Assembly (LCSA). Approximately 16 hours after the start of the accident, the RCP "1A" was restarted effectively ending the thermal degradation process.

One effect of the accident was to shatter the ceramic uranium dioxide pellets, UO_2, and to mix UO_2 with other metal oxides creating a gravel-like debris bed covering a mass of resolidified composite material. Subsequent defueling operations shattered the resolidified material into gravel-like material visually indistinguishable from the other debris.

After defueling, the residual fuel was observed to be composed of finely divided material, thin crusts, and solid masses. Small amounts of debris were left in hard to access locations, and a light "dusting" of residual fines was widely deposited on horizontal surfaces after the last underwater vacuum cleaning operation. Crusty resolidified material was distributed along surfaces that tended to channel the molten fuel. Solid masses of the same material were observed in a few LCSA peripheral coolant flow holes that could not be defueled. The UO_2 content of the fines and crusty material is similar; with a larger bulk density for the crusty material.

3.0 METHOD

3.1 VISUAL EXAMINATION METHOD

A license condition for transition from Facility Mode 1 (i.e., active defueling) to Facility Modes 2 and 3 (i.e., final preparation for PDMS) was confirmation that the RV had been defueled to the extent reasonably achievable. Video examinations of the RV performed during and subsequent to the completion of defueling were used to quantify the amount of fuel left in the TMI-2 RV. The results of those visual examinations were presented in References 1 and 1a.

Approximately 133,000 kg of core debris, including approximately 94,000 kg UO_2, remained in the RV after the accident. Extensive defueling operations were performed in the RV to remove core debris resulting from the accident. The determination of residual quantities of core debris and its location in the RV was an ongoing process from the start of defueling. Techniques included video inspection and sample acquisition and analysis.

Extensive visual examination of nearly all accessible parts of the RV internals were made during and following RV defueling. The resulting videotapes were analyzed to infer the amount, form, and location of the residual fuel in the RV (References 1 and 1a). The physical extent of debris deposits was mapped in three dimensions, using known reference points or landmarks as dimensional indicators. Given good lighting conditions, the vertical and lateral extent could be estimated fairly accurately, but depth (i.e., dimension along the line of sight) was much less easily determined. The physical distribution was then used to estimate volume. Surface texture and other subtle factors were used to identify and compare the subject debris deposit to other similar material for which sample analysis data existed. The sample data for fuel material was then used to estimate the density and composition of the deposit.

The engineers who worked on the project developed a conservative, yet realistic, estimate of the quantity of residual fuel. For instance, if a flow hole in an LCSA plate could not be seen by the video cameras, the engineer was to assume that the hole was completely full of fuel debris. For this reason, when the visual residual fuel value was reported, it was assumed to represent a conservative upper bound of the quantity of residual fuel in the TMI-2 RV. However, the visual estimate was not presented as the final residual fuel value for the TMI-2 RV due to its indeterminate range of uncertainty. Rather, passive neutron measurements were conducted as part of the SNM Accountability Program.

3.2 PASSIVE NEUTRON MEASUREMENT TECHNIQUE

Passive neutron measurements were performed to provide the final estimate of the quantity of residual fuel within the TMI-2 RV. The passive method equates the measured number of neutrons released by the core debris to the remaining fuel quantity. Passive neutrons are generated by spontaneous fission of the fuel debris and by (α,n) reactions with oxygen and boron. The alpha particles are largely produced by the decay of the transuranics.

It was postulated that the video estimates could be used to predict neutron yields for the visualized deposits. Residual fuel quantities could then be assessed by comparing neutron measurements with predictions. For simple systems, the product of the measured neutron/second (n/s) and the predicted kg UO_2 per n/s would provide the fuel quantity of record. Unfo tunately this simple analysis was complicated by geometry, and by the resolution of the video inspection method (References 2 and 2a).

The video-estimated fuel deposits were not uniform. Correlating deposits with locations revealed that the thickness of stainless steel shielding the fuel was also variable. Partially known, non-uniform deposits behind absorbers of varying thickness provide a complex fuel/source distribution and neutron absorption quality. Therefore, it was determined that a single measurement made near the center of the dry RV would not be useful to resolve the complex source term for an appropriate fuel estimate.

The visually-determined fuel distribution was used to divide the vessel into regions that could be usefully separated by water level. The implementation of a controlled draindown divided the RV internals into nine vertical zones as shown in Figure 1. Zone elevations are given in Table 1. Multiple passive neutron measurements, assisted by the RV water shield, were used to reduce the complexity for a better fuel assay. The water shield was created by draining the RV to successive zone elevations.

The distances between the entryway into the RV, the top of the chimney, and various zones were based on physical measurements and engineering drawings of the pressure vessel and vessel internal components.

3.3 PASSIVE NEUTRON MEASUREMENT BIASES

The passive neutron measurement technique is subject to systematic biases that could significantly affect the resultant estimate of record. The technical complexities involved in the passive neutron measurement program, including complex geometries, structural interferences, the presence of neutron absorbers and boron-induced (α,n) reactions, make these systematic biases difficult to quantify with any reasonable accuracy.

A distinguished group of scientists, chaired by Dr. N. Rasmussen, Massachusetts Institute of Technology, and including Dr. H. Menlove, Los Alamos National Laboratory, and Dr. G. Knoll, University of Michigan, was commissioned to conduct an independent review of the measurement data and the estimate of residual fuel. The Rasmussen Committee concluded that measurement biases were present in the passive neutron measurement technique (Reference 3). These biases and their impact on the passive neutron measurement estimate are discussed in Section 4.3.

3.4 HARDWARE FOR THE PASSIVE NEUTRON MEASUREMENT

Four assemblies comprise the hardware used for the passive neutron measurements (Reference 2). They are the detector and stand assembly; the contamination control box; the calibration source deployment system; and the RV liquid level monitor.

3.4.1 DETECTOR AND STAND ASSEMBLY

The detector assembly consisted of three He-4 detectors and coupled preamplifiers that were shielded by about 15 cm of lead and steel. The shield was used to reduce the radiation level near the detectors to about 0.5 R/h. The assembly was designed to be deployed by the RB polar crane using a 50' long steel chain. A special signal/power cable bundle was fabricated to interconnect the detector's preamplifiers to their amplifiers, power supplies, and a computer.

A stand assembly was designed to support the 2 ton detector assembly. The stand was located on the 347' El. between the tool decon facility and the "B" D-ring shield wall. Close proximity and clear visual space between the stand and associated electronics located in the contamination control box aided adjustments.

3.4.2 CONTAMINATION CONTROL BOX

A housing was provided for a computer, a NIM-bin containing several modules, an RS232 modem, and an electrical cable terminator plate. The housing was defined as the contamination control box (CCB).

The main purpose of the CCB was to shield against electrically-generated noise and to allow flexibility in terminating the various signal/power cables in a contamination-free environment.

3.4.3 CALIBRATION SOURCE DEPLOYMENT SYSTEM

For calibration and operational checks, an Am-241-Be neutron source was lowered into the RV through three stainless steel tubes. The tubes extended down from the shielded work platform to the bottom of the RV. The tube used for calibration was located inboard of the Core Support Structure (CSS)-Upper Core Support Assembly (UCSA)-LCSA.

An 1/8" stainless steel ball and bead chain was used to support the source. Incremental chain length distance uncertainties were less than 1/4". The position of the source was monitored by calibrating the ball and bead chain for length. The calibration marks were white balls against a flat black painted chain. Readout of source position was by a video monitor located in the command center which was connected to a camera attached to the reel assembly. Readout in the RB was not needed as the source controller was located within clear sight of the source reel.

3.4.4 RV LIQUID LEVEL INDICATOR

A staged precision bubbler was used for liquid level readout. The heart of this device was a pressure transducer that was rated for an accuracy of 0.1%. A water depth of about 11.5' was covered by a 5 psig transducer, and the liquid level was followed during pump-out by moving the bubbler elevation in large stages. The error determined by quadrature*, anywhere in this span, was 0.23 inches.

4.0 ANALYSIS

4.1 VISUAL ESTIMATE

The visual estimate of the residual fuel in the TMI-2 RV was 630 kg. The documentation in References 1 and 1a provides a description of the analytical

* Square root of the sum of the squares of the individual errors.

results of the video examination of the TMI-2 RV. The data presented showed a residual fuel inventory of 609 kg. However, for comparison with the passive neutron measurement estimate and the estimate of record, the 21 kg extant in the hot and cold leg nozzles (Reference 4) was added to achieve the stated 630 kg (Figure 2).

As shown on the archived videotapes, the residual fuel in the TMI-2 RV consists primarily of finely divided, small particle-size sediment in inaccessible holes, crevices, and corners; surface films; and resolidified material either tightly adherent to the RV components or inaccessible for defueling. Approximately 50 samples of residual core debris were analyzed for density and fuel (i.e., UO_2) content. An average density and representative fuel content was determined for each type of material. A computer model of the RV and its internal components was developed which defined the dimensional characteristics of the remaining RV internals and, more importantly, the spaces where core debris could reside. Then, based on observation, the actual core debris was spatially represented in the computer model.

Computer software was developed which used the information contained in the geometry model to calculate the volume of modelled debris. Other visual factors such as surface texture, shape, apparent hardness or porosity, friability, color, and location were used to categorize the residual core debris into one of the three types discussed above. A mass estimate was then obtained by multiplying the estimated core debris volume by the average density and UO_2 percentage corresponding to the particular type of material (i.e., resolidified material, loose debris, or surface film).

The benefit of the video assay was not only in identifying where fuel deposits were located but also in identifying where no fuel deposit greater than the video camera resolution of about 1 mm existed. However, the video assay had its limitations, such as, lack of access to all areas of the RV, translation of a two-dimensional image into a three-dimensional portrait, and the resolution of the video cameras. For these reasons, no uncertainty was ascribed to the video estimate and, more importantly, the video estimate was not considered sufficiently accurate to be used as the sole basis for the estimate of record for SNM accountability purposes. The SNM Accountability Program required a more technically exact measurement technique; thus, the passive neutron measurement technique was developed.

4.2 PASSIVE NEUTRON MEASUREMENT ANALYSIS

The passive neutron measurements produced raw neutron count rates for a series of locations within the RV under known conditions. Fuel increments in each zone were determined by satisfying coupled simultaneous sets of equations that represented the series of measurements made in each zone (Reference 2). A neutron measurement was made with the water level at the bottom of one zone (e.g., Zone 3) and the detector located within that zone. The water level was then dropped to the bottom of the next zone (e.g., Zone 4) and with the detector in the same location (e.g., within Zone 3), another neutron measurement was made. The detector was then lowered to the next zone (e.g., within Zone 4) and the measurement process was repeated. Neutron attenuation in stainless steel, lead, and water was represented by simple exponentials with increasing thickness. Gamma ray transport codes were used with material and density selected to force the codes to match the empirical attenuation characteristics.

The fuel locations and a portrait of the remaining RV internal structure defined by the video inspection were used to produce minimum and maximum fuel values. The minimum fuel value was attained by adding fuel to areas nearest the detector until the actual measured neutron count rate equalled the computer-modelled count rate. Conversely, the maximum fuel value was determined by adding fuel to areas furthest from the detector until the same match was achieved. The average of the minimum and maximum values for each zone was the passive neutron measurement estimate. The passive neutron measurement estimate of residual fuel in the TMI-2 RV was 1322 kg.

4.3 EVALUATION OF MEASUREMENT BIASES

Since the passive neutron estimate was more than twice the visual estimate, which was considered conservative, the Rasmussen Committee was commissioned to evaluate the measurement data, the passive neutron measurement technique, and residual fuel estimate.

The Rasmussen Committee concluded that there were five systematic measurement biases in the passive neutron measurement technique and analysis. These biases, related to boron variations, UO_2 particle size, calibration error, data analysis, and in-scattering of neutrons, were difficult to quantify with any reasonable accuracy.

4.3.1 BORON VARIATIONS

Neutron yield from the TMI-2 core debris is primarily a function of three distinct components: spontaneous fission of transuranics, (α,n) reactions from oxygen, and (α,n) reactions from boron. The dominant neutron source term is (α,n) reactions from boron; the neutron yield from this component is proportional to the thickness of the boron layer on the surface of the UO_2. The Rasmussen Committee concluded that a bias in the passive neutron measurement was the inability to fully drain the borated water without a buildup of boron on the residual fuel. This resulted in an increase in the neutron yield from residual fuel in Zones 1 through 5. The Rasmussen Committee ascribed an estimated bias of +15% for this effect in Zones 1 through 5.

4.3.2 UO_2 PARTICLE SIZE

Fuel particle size has a strong affect on neutron yield. A UO_2 pellet will yield orders of magnitude less neutrons per gram than UO_2 powder with a surface coating of boron. Further, small dust-size particles would adhere to all surfaces; whereas, larger particles would tend to drop to the bottom of the vessel. Therefore, the Rasmussen Committee concluded that the passive neutron measurements were biased because the neutron yield calibrations used TMI-2 fuel samples that were comprised of a non-representative lesser quantity of fine particles. Defueling videos provided extensive observations of fine powder deposits in Zones 1 through 5. Therefore, the Rasmussen Committee ascribed a total estimated bias of +45% from these effects to Zones 1 through 5.

4.3.3 CALIBRATION ERROR

A measurement bias was introduced by using an AmBe source for the primary calibrations. The calibration technique caused the detector to be more efficient for the larger-size particles extant in the bottom regions of the RV. An additional source of calibration error was the variation in angular distribution of the neutrons from the AmBe source as compared to the UO_2 sample. The bias for these effects was estimated at -5% on a total vessel basis.

4.3.4 DATA ANALYSIS

Nine grab samples of fuel debris were used for neutron calibration. The results from one of the samples was omitted from the calculation of neutron yield. Including this sample in the average would increase the neutron yield by approximately 5%. Therefore, the bias for this effect was estimated at +5% on a total vessel basis.

4.3.5 NEUTRON IN-SCATTERING

The passive neutron measurement technique corrected the data for neutron absorption and out-scatter, but no credit was given for neutron in-scatter. A bias was introduced through the use of gamma-ray computer codes that do not account for neutron scatter in non-line-of-sight structural steel. The Rasmussen Committee ascribed an estimated bias of +20% on a total vessel basis for this effect.

4.4 ESTIMATE OF RECORD

Since the residual fuel result of the passive neutron measurements was so much larger than the visual estimate and considering the Rasmussen Committee's discussion of systematic biases, it was decided that the estimate of record of the residual fuel in the TMI-2 RV for SNM accountability purposes would be determined by adjusting the results of the passive neutron measurements to account for the biases discussed in Section 4.3. The biases were assigned to specific RV zones and then summed per zone (Table 2). The passive neutron measurement estimate was adjusted accordingly resulting in a zonal estimate of record (Table 3). The estimate of record of the residual fuel in the TMI-2 RV is 925 kg. Figure 2 provides a comparison of the three fuel estimates, i.e., visual estimate, passive neutron estimate and estimate of record, for the TMI-2 RV on a per zone basis.

4.5 ANALYSIS OF UNCERTAINTY

The passive neutron measurement report (Reference 2) includes an uncertainty estimate of approximately ± 15 % for the estimated non-bias adjusted RV content of 1322 kg. The Rasmussen Committee judged the uncertainty to be ± 40% when bias adjustments are made to the passive neutron estimate. This uncertainty was based on the following assumptions:

1. Approximately two-thirds of the neutrons counted are not related to the amount of uranium present;

2. The fuel source material is in a wide variety of forms; each with a different neutron yield per gram of uranium;

3. The highly asymmetrical location of the steel complicates evaluation of the in-scattering effect; and

4. The distribution of boron changes during water removal.

It is noteworthy that if two-sigma uncertainties (i.e., 95% confidence level) are considered, the estimate of record (925 kg) with its estimated two-sigma uncertainty of 80% certainly bounds the visual estimate (630 kg) plus any reasonable associated uncertainty, and essentially bounds the unadjusted passive neutron estimate of 1322 kg plus its two-sigma uncertainty (30%). Figure 3 provides a visual representation of the uncertainty evaluation.

Therefore, it appears reasonable to adopt the Rasmussen Committee's assignation of an uncertainty of ± 40% to the TMI-2 RV estimate of record.

5.0 CONCLUSION

The estimate of record of the amount of uranium dioxide (UO_2) remaining in the TMI-2 RV is 925 kg. This estimate has an associated uncertainty of ± 40%. Since the RV has been drained to the extent that less than 10 gallons of water remain, no fuel movement into or out of the RV is anticipated, and this quantity of residual fuel is considered fixed.

This estimate of record is based on a detailed, non-destructive visual examination and physical measurement of the RV residual fuel that was adjusted in accordance with the recommendations of a panel of distinguished scientists and evaluated by a group of independent external reviewers. It is understood that the bias adjustments are somewhat judgemental due to the difficulty experienced in their quantification. However, the reviewers were in general agreement regarding both the estimate of record and its associated uncertainty. Considering the information provided by the visual estimates and the circumstances under which the passive neutron measurements were made, the estimate of record of 925 kg represents a realistic estimate of the quantity of residual fuel extant in the TMI-2 Reactor Vessel and the consensus of the reviewers commissioned to assess the results for the TMI-2 SNM accountability project.

REFERENCES

1. GPU Nuclear letter 4410-90-L-0012, "Defueling Completion Report, Final Submittal," dated February 22, 1990.

1a. GPU Nuclear letter 4410-90-L-0026, "Results Of Post-Lower Head Sampling Program Cleanup," dated April 12, 1990.

2. GPU Nuclear memorandum, 4240-92-114, "Results of Passive Neutron Measurement Program," dated August 21, 1992.

2a. Letter, C. Distenfeld to R. Rogan, dated September 18, 1992, transmitting the report, "TMI-2 Reactor Vessel Fuel Assay."

3. GPU Nuclear memorandum, 5400-92-0045, "Final Transmittal of Rasmussen Review Committee Reports," dated August 28, 1992.

4. GPU Nuclear memorandum, 4850-90-008, "Fuel Assessment Of Hot, Cold, and Core Flooding Nozzles, Tool, Tool Rack, and Tool Decon Facility," dated February 12, 1990.

5. GPU Nuclear letter, C312-92-2080, "TMI-2 Reactor Vessel Criticality Safety Analysis," dated December 18, 1992.

TABLE I

ZONE ELEVATIONS

	ZONE	ELEVATION	DEPTH
0.	top of chimney	337.43'	00.00'
1.	top of flange	322.48'	14.95'
2.	top of hot leg	317.00'	20.43'
3.	bottom of cold leg nozzles	313.96'	23.47'
4.	bottom of CSS	312.06'	25.36'
5.	mid-plane of UCSA	305.17'	32.25'
6.	bottom of USCA	298.29'	39.14'
7.	top of forging	296.91'	40.52'
8.	top of IGSP	295.74'	41.69'
9.	empty	291.07'	46.36'

TABLE 2

CORRECTION FACTORS FOR MEASUREMENT BIAS

CORRECTION FACTOR	BIAS	APPLICABLE ZONES
Boron Variations	+15%	1 - 5
UO_2 Particle Size	+45%	1 - 5
Calibration Error	- 5%	1 - 9
Data Analysis	+5%	1 - 9
In-Scattering	+20%	1 - 9

SUMMATION OF BIASES

Zones 1 - 5	Bias = 80%
Zones 6 - 9	Bias = 20%

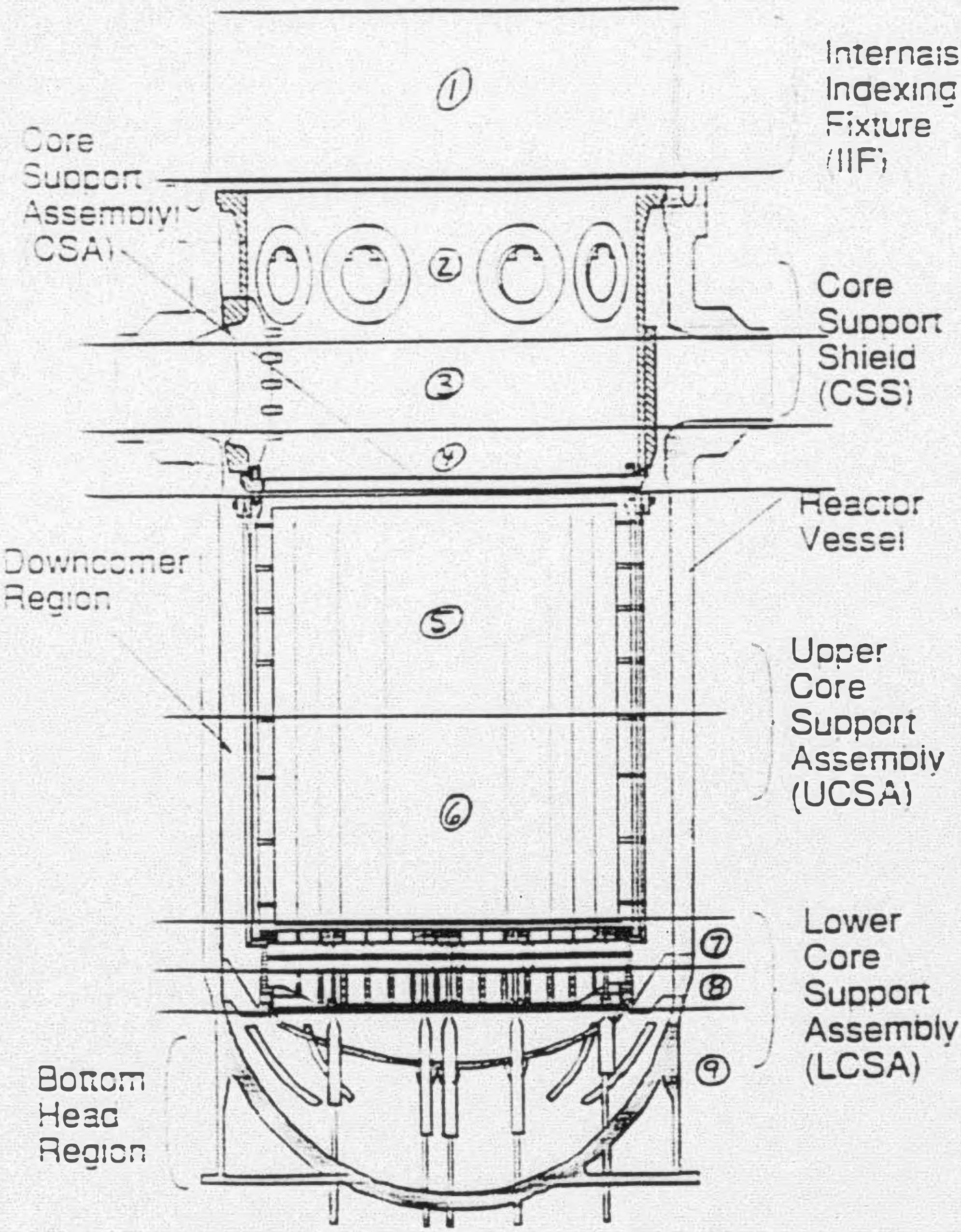
TABLE 3

REACTOR VESSEL CALCULATION OF THE ESTIMATE OF RECORD

ZONE	PASSIVE NEUTRON MEASUREMENT ESTIMATE	CORRECTION FACTOR	ESTIMATE OF RECORD
1	10	1.8	6
2	225	1.8	125
3	150	1.8	83
4	99	1.8	55
5	154	1.8	86
6	387	1.2	323
7	113	1.2	94
8	89	1.2	74
9	95	1.2	79
TOTALS	1322		925

FIGURE 1

RV Vertical Zones

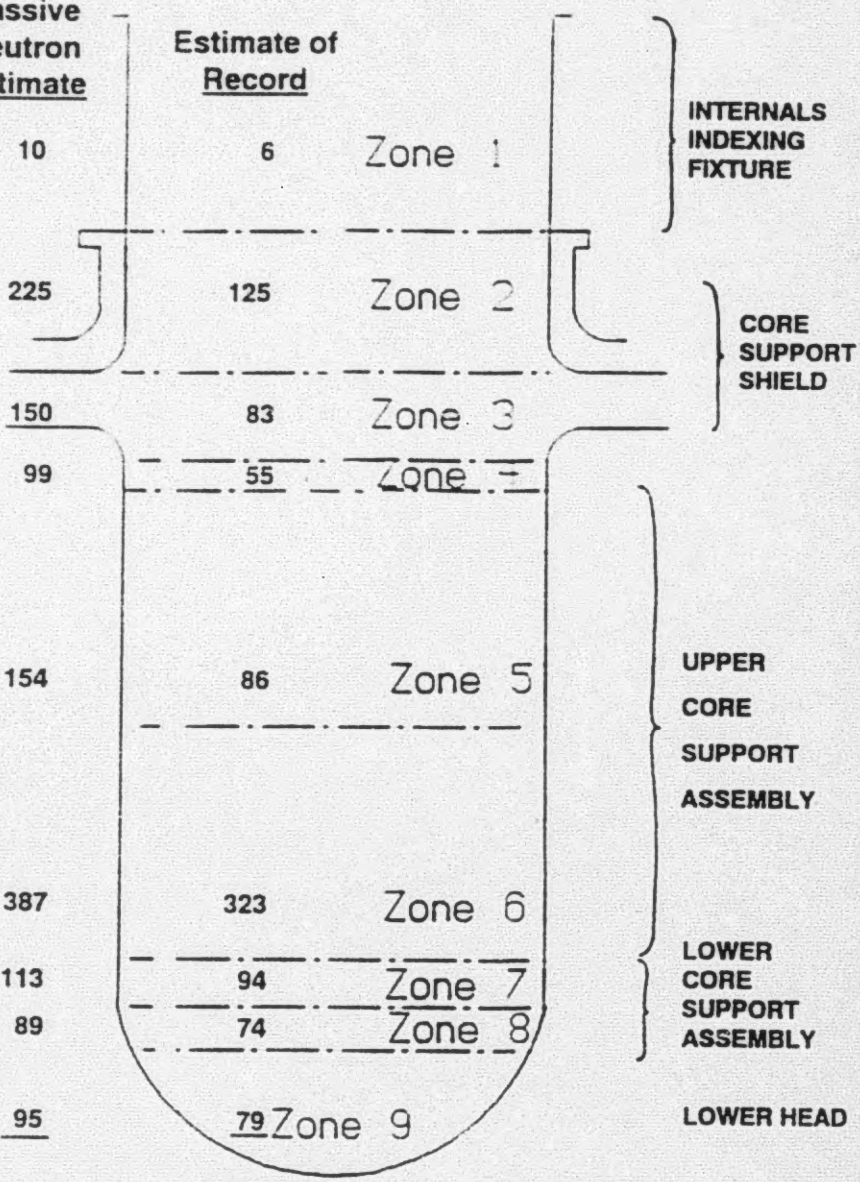


Reactor Vessel Cutaway View

FIGURE 2

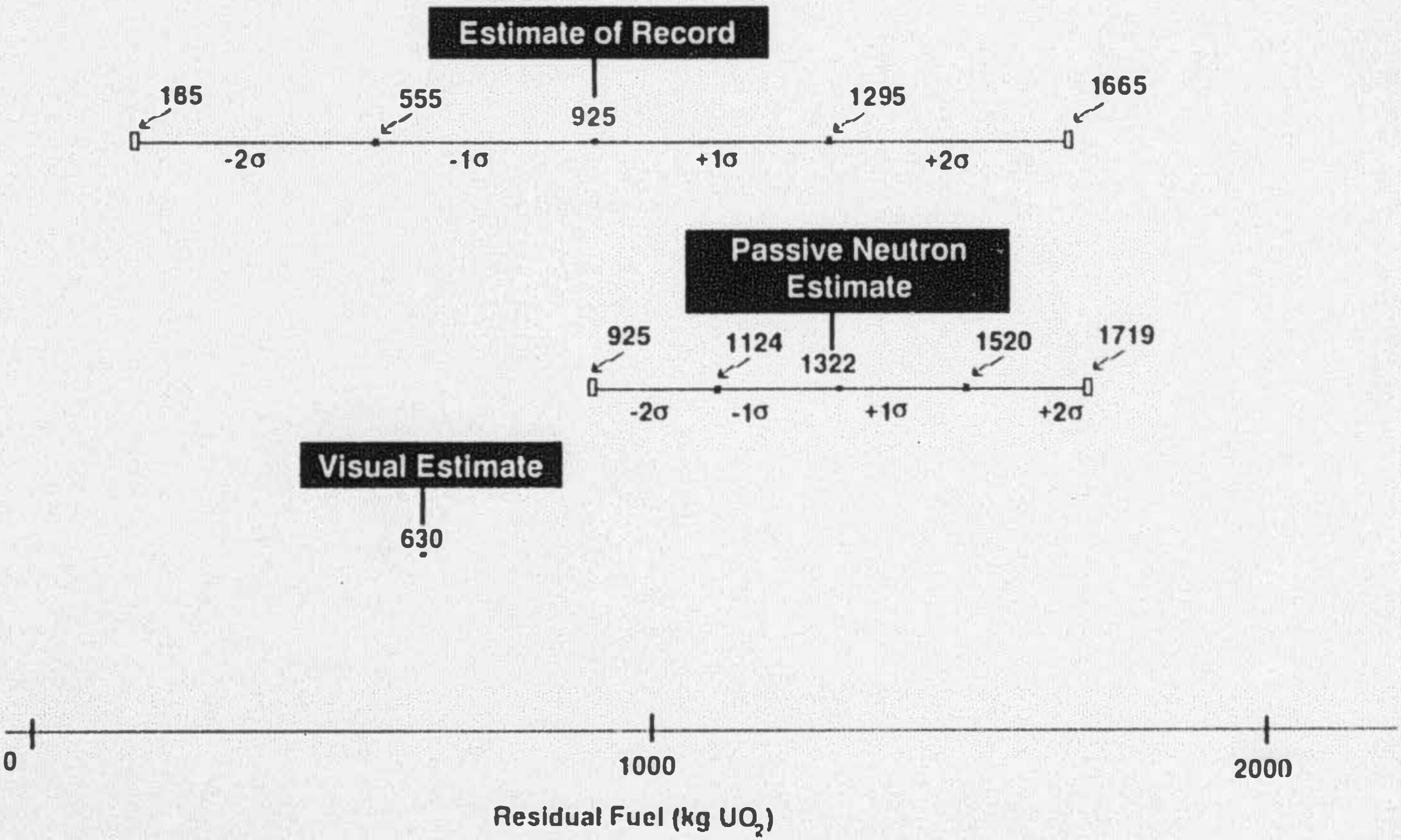
Comparison of RV Fuel Estimates

	Visual Estimate	Passive Neutron Estimate	Estimate of Record
Zone 1	10	10	6
Zone 2	62	225	125
Zone 3	24	150	83
Zone 4	27	99	55
Zone 5	8	154	86
Zone 6	244	387	323
Zone 7	93	113	94
Zone 8	133	89	74
Zone 9	29	95	79
Totals	630 kg	1,322 kg	925 kg ± 40% (555 - 1,295)



Estimate of Record = 925 kg

FIGURE 3
Evaluation of Estimate of Record Uncertainty



ENCLOSURE 2

EXECUTIVE SUMMARY

The initial Three Mile Island Unit 2 (TMI-2) core contained 94.0 tonnes (94,000 kg) of uranium dioxide (UO_2). Three months of full power operations, defueling, and fuel shipment to the Idaho National Engineering Laboratory reduced the inventory to approximately 1100 kg UO_2[1] or about 1.2% of the original core loading. The purpose of this document is to discuss the process used to determine the total amount of UO_2 remaining in the TMI-2 facility, and to document the final fuel estimate of record.

Table 1 depicts the final residual fuel distribution of the TMI-2 facility. The total quantity of residual fuel was approximately 1100 kg with an uncertainty of about ± 370 kg. The uncertainty (one standard deviation - 1σ) was calculated as the square root of the sum of squares of the individual uncertainties. In general, the individual uncertainties were the standard deviations of the "estimates of record." For the Auxiliary and Fuel Handling Buildings (AFHB), Reactor Building Miscellaneous Components, Plenum, and the Reactor Vessel (RV) Head, the uncertainties were reported as asymmetric quantities. For these cases, the averages of the upper and lower error bounds were taken as the standard deviation. The uncertainty in the residual fuel in the RV dominated the total uncertainty. It is useful to note that the total quantity of residual fuel outside of the RV was about 170 kg; an amount less than one-half of the standard deviation of the residual fuel in the RV.

The collection of PDSR sub-reports provide a comprehensive and systematic accounting of all

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1 Given an average fuel enrichment of 2.24 wt % U-235, the total quantity of contained U-235 at TMI-2 is less than 22 kg.

residual fuel in the TMI-2 facility. The results of each survey of an area, system, or component containing special nuclear material (SNM) are detailed in a separate PDSR sub-report and a compilation of these results forms the basis for the final SNM accounting for TMI-2.

To facilitate the review and comprehension of this compilation of sub-reports, a section describing the SNM Program Plan and the specific requirements and considerations for measurements, quality assurance, and program management is included. The Introduction section describes the format of the PDSR sub-reports, provides a graphic illustration of the SNM areas, systems, and components being surveyed, and explains the various methods and techniques used to measure SNM and analyze the data. Each sub-report provides specific analytical and/or measurement results, the "estimate of record" of the residual fuel for the given area, system, or component, and a discussion of the measurement methods and analytical techniques utilized, as well as the level of uncertainty of the results. Each sub-report received an independent external review by Dr. R. Brodzinski, Battelle Pacific Northwest Laboratories, Dr. F. Tingey, University of Idaho, and either Dr. T. Sampson or Dr. P. Russo, Los Alamos National Laboratory.

The fuel assay process extended over several years. The earlier PDSR sub-reports tended to cover the fuel assay results for specific components (e.g., the RV Plenum). More recent reports served to group together components common to a specific location, such as the AFHB and the Reactor Coolant System. For ease and continuity of discussion, the detailed data, measurement information, calculations, and analysis were not included in the sub-reports. The individually referenced support material was archived for retrieval by interested parties.

TABLE 1

POST-DEFUELING DISTRIBUTION OF RESIDUAL UO_2 IN THE TMI-2 FACILITY

PDSR SECTION TITLE	UO_2 kg	UNCERTAINTY kg
Plenum	2.1	± 1.9
Letdown Coolers	3.7	± 2.0
Pressurizer	0.3	± 0.2
Reactor Vessel Head	1.3	± 0.9
Reactor Building Basement	1.3	± 0.7
"A" and "B" OTSG	62.3	± 9.7
Auxiliary and Fuel Handling Buildings	11.5	± 5.8
Reactor Building Miscellaneous Components	64.0	± 26.9
Reactor Coolant System	25.8	± 11.1
Reactor Vessel	925	± 370
TMI-2 Facility Total Fuel	**1097**	**± 371**

SUMMARY OF TMI-2 SPECIAL NUCLEAR MATERIAL ACCOUNTABILITY PLAN

PURPOSE

The Special Nuclear Material (SNM) Accountability Plan was initially issued in April 1987. The purpose of the SNM Accountability Plan was to define the method of SNM accountability, the Quality Assurance Program incorporated in the SNM Accountability Program, the areas, systems, and components to undergo formal SNM measurement, and those that did not require SNM assessment. The Plan also described the integration of programmatic ALARA in SNM assessment activities.

As defined in the SNM Accountability Plan, the post-defueling survey was a process by which the entire TMI-2 plant was surveyed to identify areas known to contain, or that could contain, residual SNM and to quantify that SNM. The quantity of SNM in each applicable area was determined. The accomplishment of the required SNM measurements and associated engineering analyses, and a determination of the estimate of record of the total quantity of residual fuel at TMI-2, constitute completion of the post-defueling survey. The SNM Accountability Plan describes the process by which the post-defueling survey was conducted.

BACKGROUND

The March 1979 accident resulted in significant damage to the core and the subsequent release of fuel and fission products into the Reactor Coolant System (RCS) and other closely related systems. The post-accident TMI-2 core consisted of loose fuel pellets, solidified fuel, structural metal components (e.g., end fittings), loose rubble, and partial fuel assemblies. This collection of material is generically referred to as core debris. As a result of the core condition, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

Core debris was removed from the TMI-2 reactor, loaded in special containers, and shipped to the Department of Energy Idaho National Engineering Laboratory (DOE INEL) facility in Idaho. Each shipment was accompanied by a Nuclear Material Transaction Report (DOE/NRC Form 741) which recorded the net weight of the contents of each canister and a best available physical description of the contents. A statement that quantification of the amount of SNM in each canister was not possible also accompanied each shipment as an annotation on the DOE/NRC Form 741.

The canister contents were a mixture of SNM and other core debris and structural materials. There was no feasible method at TMI-2 to determine the exact content of fuel in each canister by isotope with gram specificity. Therefore, SNM accountability for TMI-2 was based on the total measured SNM remaining in the plant after defueling was complete. A final plant inventory of residual SNM will be reported on the DOE/NRC Material Balance Report (DOE/NRC Form 742).

In October 1985, GPU Nuclear, the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) entered into an agreement (References 1 and 2) that final SNM accountability for TMI-2 would be performed after defueling was completed and would be based upon a thorough post-defueling survey of TMI-2. This post defueling survey would quantify, as accurately as possible, the amount of residual SNM in plant areas, systems, and components. Implied in this agreement was the understanding that the post-defueling survey would involve all areas, systems, and components where SNM could possibly have been deposited as a result of the 1979 accident and subsequent cleanup activities.

SUMMARY

Formal SNM assessment activities began in 1987 and continued until 1992. SNM measurements were performed as areas, systems, and components were defueled and/or placed into an isolated configuration that ensured no fuel transport in or out after the SNM survey was completed.[1] The intent of this isolation configuration was to ensure no transport of SNM into or out of an area, system or component previously surveyed which could significantly alter the SNM assessment.

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[1] An exception to this was the RV draindown to the fuel transfer canal (FTC). No special precautions were taken to prevent fuel transport to the FTC. However, little fuel was in fact transferred.

Some SNM measurements were performed on areas, systems, or components that were not in an isolated configuration. This approach was not preferred and these measurements were utilized for SNM accountability only when analysis demonstrated that there could be no significant transport of SNM into or out of the area, system, or component after the completion of the measurement. When areas, systems, or components could be physically isolated prior to SNM measurement, the isolation configuration was selected to enhance SNM detection with due regard for system boundaries, piping configuration, and measurement requirements. Following SNM measurements, the configuration was and will continue to be administratively and physically controlled.

A review of the entire TMI-2 facility was conducted to determine where SNM may have been deposited as a result of the 1979 accident and subsequent cleanup activities. Locations were placed into three categories: Category A - locations where SNM is definitely deposited; Category B - locations where it can be reasonably postulated that SNM may be deposited; and Category C - locations where it can be shown that SNM was not deposited. Appendix 1 categorizes each area, system, and component of concern.

All Category A and B areas underwent SNM assessment. Category C areas were identified as not requiring SNM assessment based upon authoritative analysis of the TMI-2 accident (Reference 3) and a review of subsequent cleanup activities.

SNM accountability at TMI-2 was a complex task. Inaccessibility of some areas, systems, and components, high area radiation backgrounds, complex geometries, and the required indirect measurement of fuel complicated physical measurement of SNM quantities. Therefore, several alternative techniques for performing measurement of SNM quantities were used. Reference 5 describes those techniques.

SNM assessment was an Important to Safety (ITS) activity; therefore, the TMI-2 Recovery QA Plan applied to all SNM assessment activities. The SNM Accountability Plan, SNM measurement procedures, and Unit Work Instructions underwent Quality Assurance/Quality Control (QA/QC) review and concurrence. Measurement equipment was maintained and calibrated in accordance with QA/QC requirements. Individual SNM assessment activities included QA/QC verification of essential parameters, as deemed necessary. Records of SNM assessment activities and associated analyses were subject to QA/QC monitoring and auditing. Engineering calculations for SNM assessment were performed in accordance with the TMI-2 Engineering Procedure 4000-ENG-7310.02, "Engineering Calculations."

POST-DEFUELING SURVEY REPORTS

The collection of Post-Defueling Survey Reports (PDSRs) documents the estimates of record of residual SNM in each Category A and B area, system, and component at TMI-2. The quantity of residual SNM was determined through measurements, sampling, and engineering analysis. Each PDSR sub-report contains, as a minimum, a description of the boundaries of the sub-report, a detailed description of the area, system, or component to be surveyed, a description of its role in the accident and/or subsequent cleanup activities, specific measurement results, an explanation of the measurement method(s) used to obtain fuel measurement data, a discussion of the analysis techniques utilized, the estimate of record of residual fuel, and its associated uncertainty.

DETERMINATION OF SNM IN RADIOACTIVE WASTE AND SAMPLE SHIPMENTS

The total amount of SNM shipped offsite as radioactive waste and/or as samples was determined in accordance with approved ITS procedures. This amount of SNM shipped was then recorded on the appropriate shipping forms and the DOE/NRC Form 741. The quantities of SNM reported as shipped was included in the calculation which finalized the SNM accountability process at TMI-2.

FINAL SNM ACCOUNTABILITY

Final SNM accountability for the TMI-2 facility was determined by compiling the results of individual sub-reports (i.e., "estimates of record") of the quantity of residual SNM for each area, system, or component. The amount of fuel shipped to the DOE INEL was determined by subtracting the sum of the final plant inventory and the amount of SNM shipped as radioactive waste from the total plant inventory of SNM as reported on the most recent SNM Material Balance Report (DOE/NRC Form 742) as corrected for decay.

Last reported inventory

- Decay correction

- Final plant inventory

- SNM shipped as samples and radwaste

= SNM shipped to INEL in fuel, filter and knockout canisters

SNM MEASUREMENTS

SNM measurements were performed on TMI-2 Category A and B areas, systems, and components (Appendix I). A determination of the residual SNM in each location was based upon individual SNM measurements performed using approved procedures, or by examination and analysis of previously performed fuel characterization measurements. SNM measurements were performed in accordance with an approved procedure or Unit Work Instruction when existing data from previous fuel characterization measurements were insufficient for final SNM accountability. Generic procedures were utilized for measurements which applied standard techniques (e.g., gamma spectroscopy). Unit Work Instructions were utilized for measurements that required special, one-of-a-kind techniques (e.g., OTSG tube film SNM measurement). Data sheets attached to each SNM measurement document were used to record the data required for post measurement analysis.

SNM measurement documents were reviewed and approved in accordance with TMI-2 Unit Procedure 4000-ADM-1218.02, "TMI-2 Document Evaluation, Review and Approval." SNM measurement documents required internal concurrence in accordance with TMI-2 Procedure 4000-PLN-4420.02, "SNM Accountability Plan." SNM measurement documents were archived in CARIRS.

SNM MEASUREMENT TECHNIQUES

Post-defueling SNM assessment at TMI-2 was a complex task. Several different measurement techniques were used. Technique selection for each measurement depended upon the configuration of the area, structure, or component assayed, physical access limitations, area radiation dose rates, and the likely nature of the form of SNM deposits. Extensive use was made of gamma scintillation counting, neutron detectors, video inspections, radiochemical analysis of samples, and alpha detection. Several areas were surveyed utilizing two or more techniques. A description of the most common measurement techniques and measurement technique selection criteria can be found in Reference 4.

QUALITY ASSURANCE FOR SNM ACCOUNTABILITY

The results of the SNM Accountability Program were the basis for final SNM accountability at TMI-2. Therefore, the SNM accountability activities were classified as "Important to Safety" and complied with the TMI-2 Recovery QA Plan.

Final SNM accountability was based on a determination of the quantity of residual SNM remaining in TMI-2 areas, systems, and components after defueling was completed. The determination of the residual SNM quantities was based upon measurements performed utilizing QA/QC approved procedures or upon measurement packages that contained previously performed fuel characterization measurements reviewed and approved by QA/QC on an after-the-fact basis.

SNM measurement activities performed via approved procedures or Unit Work Instructions were reviewed by QA/QC for inclusion of hold/witness points. Specific activities (e.g., critical items) were also identified by the document preparer which required QA/QC verification. Activities requiring QA/QC observation required coordination to ensure that QA/QC support was timely and available for the specific activity.

SNM engineering calculations were performed in accordance with TMI-2 Engineering Procedure 4000-ENG-7310.02, "Engineering Calculations." Data utilized in the engineering calculations were obtained from the completed QA approved procedure or Unit Work Instruction data sheets or from QA reviewed and approved data acquisition measurement packages. Calculations were independently verified in accordance with the referenced procedure. SNM measurement packages that identified residual SNM deposits greater than 1 kilogram were submitted for an independent review.

Computer codes utilized to quantify residual SNM were verified by the Fuel Measurement and Analysis Section by benchmarking with accepted industry codes. Verification was documented. Only verified and approved computer codes were utilized. Approved code versions were controlled to preclude unauthorized modification. Code versions utilized in engineering calculations were specifically identified.

Equipment utilized to quantify residual SNM via a QA-approved procedure or Unit Work Instruction was calibrated and operated in accordance with these procedures. Essential equipment identification information (e.g., type, size, configuration) and performance data (e.g., counts, duration of count, location of detector) were recorded on the data sheets.

CONFIGURATION CONTROL OF AREAS, SYSTEMS, OR COMPONENTS CONTAINING RESIDUAL SNM

In order to ensure that the SNM measurement process was accurate, controls were established to ensure that SNM was not "double counted." Double counting could have occurred if SNM was relocated out of an area, system, or component that had already been measured into a radioactive waste shipment or an area, system, or component still requiring SNM assessment.

Where possible, areas, systems, and components were physically isolated prior to SNM measurement. However, some SNM measurements were performed on areas, systems, or components that were not in an isolated configuration. In these instances, the measurements were accompanied by an analysis which demonstrated that there could be no significant transport of SNM into or out of the area, system, or component after the completion of the measurement.[2]

Administrative controls were utilized, as required or appropriate, to maintain physical isolation of areas, systems, or components that have undergone SNM measurement so that transport of SNM into or out of the area, system, or component was precluded. The type of administrative controls depended upon the nature of the area, system, or component.

Piping systems and components connected to piping systems (e.g., pumps, tanks) were isolated utilizing "red" tags via the TMI-2 Administrative Procedure 4000-ADM-3020.04, "Switching and Tagging Safety." Red tags prohibit the changing of position of a component (e.g., valves, electrical breakers). "Red tagged" components served as isolation barriers intended to prevent the transport of residual SNM into or out of areas, systems, or components. If an isolation barrier was removed (e.g., a valve was opened), the need for a repeat measurement of SNM in the affected piping was evaluated.

Open areas (e.g., reactor building basement floor, auxiliary building sump) were controlled by one of two methods to prevent fuel transport. Areas were maintained in a dry condition so that liquids could not displace residual SNM or, if subjected to liquid flow, the liquid was

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2 An exception to this was the RV draindown to the fuel transfer canal (FTC). No special precautions were taken to prevent fuel transport to the FTC. However, little fuel was in fact transferred.

sampled. Sampling of the liquid for the presence of residual SNM was performed in accordance with QA approved procedures.

ALARA FOR SNM ACCOUNTABILITY

The SNM Accountability Program was conducted in accordance with the "As Low As Is Reasonably Achievable" (ALARA) principle for radiation exposure. Conduct of the post-defueling survey, as an integral element of the SNM Accountability Program, resulted in the exposure of personnel to ionizing radiation as SNM measurements were performed in the auxiliary, fuel handling and reactor buildings. Personnel radiation exposures were maintained ALARA by limiting the number of measurements to those essential for SNM accountability and by planning each measurement to minimize personnel exposure.

The number of required SNM measurements was limited by utilizing, when possible, previously performed fuel characterization measurements. In addition, SNM measurements were limited to those areas, systems, or components which conceivably contained fuel.

The radiation exposure received by personnel performing SNM measurements was maintained ALARA by proper planning. Individual measurements were designed to include efficient use of time in radiation areas, incorporate lessons learned on dose minimization from previous measurements, and include the use of remote equipment, if possible.

REFERENCES

1. DOE letter WWB-100-85, "Accountability for the TMI-2 Core," W. W. Bixby to H. M. Burton, EG&G, dated October 8, 1985.

2. NRC letter, "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d), and 70.53," B. J. Snyder to F. R. Standerfer, GPUNC, dated October 17, 1985.

3. NSAC 80-1, "Analysis of Three Mile Island - Unit 2 Accident; Rogovin Report: Three Mile Island, A Report to the Commissioners and the Public."

4. TPO-TMI-187, "Instrument Selection for Residual Fuel Measurements," Rev. 0, dated January 1987.

5. TMI-2 Engineering Procedure 4000-ENG-7310.02, "Engineering Calculations."

6. TMI-2 Administrative Procedure 4000-ADM-3020.04, "Switching and Tagging Safety."

7. TMI-2 Administrative Procedure 4000-ADM-1218.02, "TMI-2 Document Evaluation, Review and Approval."

8. TMI-2 Unit Policy/Plan Procedure 4000-PLN-4420.02, "SNM Accountability Plan."

APPENDIX 1

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
AX001	RB Emergency Booster Pump	C
AX002	Access Corridor (Drains)	C
AX003	Access Area (Drains)	C
AX004	Seal Injection Valve Room	B
AX005	MU Pump - 1C	A
AX006	MU Pump - 1B	A
AX007	MU Pump - 1A	A
AX008	Spent Resin Storage Tank - 1B	B
AX009	Spent Resin Storage Tank - 1A	B
AX010	Spent Resin Storage Tank Pump	B
AX011	Aux. Sump Pump Valve Room	A
AX012	Aux. Bldg. Sump Tank Room	A
AX013	Evap. Cond. Tanks, Pumps	C
AX014	RC Evaporator Room	C
AX015a	Cleanup Filters Room	C
AX015b	Cleanup Filter After Room	C
AX016	Cleanup Demineralizer - 2A	C
AX017	Cleanup Demineralizer - 2B	C
AX018	Waste Transfer Pump Room	C
AX019	Waste Disposal Liquid Valves	A
AX020	RC Bleed Tanks - 1B, 1C	A
AX021	RC Bleed Tank - 1A	A
AX022	North Stairwell	C
AX023	Elevator Shaft	C

APPENDIX 1 (cont'd)

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
AX024	Aux. Bldg. Sump Filters	B
AX026	Seal Injection Filters (MU-F4A, 4B)	B
AX027	South Stairwell	C
AX101	Radwaste Disposal Panel	C
AX102	RB Sump Pump Filters	B
AX103	MCC 2-11EB	C
AX104	MCC 2-21EB	C
AX105	Substation 2-11E	C
AX106	Substation 2-21E	C
AX107	MCC 2-11EA	C
AX108	MCC 2-21EA	C
AX109	Nuc. Services Coolers and Pumps	C
AX110	Intermediate Coolers	C
AX111	Intermediate Cooling Pumps and Filters	C
AX112	Seal Return Coolers & Filters (MU-F-3)	B
AX113	Waste Gas Analyzer	C
AX114	MU&P Demineralizer - 1A	A
AX115	MU&P Demineralizer - 1B	A
AX116	MU Tank	A
AX117	MU&P Filters (MU-F2A&B and MU-F5A&B)	A
AX118	Spent Fuel Coolers	C
AX119	Spent Fuel Demineralizers	C
AX120	Spent Fuel Filters	C
AX121	Elevator Shaft	C
AX122	North Stairwell	C

APPENDIX 1 (cont'd)

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
AX123	Access Area	C
AX124	Concent. Liquid Waste Pump	B
AX125	Waste Gas Decay Tank - 1B	C
AX126	Waste Gas Filter Room	C
AX127	Waste Gas Decay Tank - 1A	C
AX128	Valve and Instrument Room	C
AX129	Deborating Demineralizer - 1B	B
AX130	Deborating Demineralizer - 1A	B
AX131	Misc. Waste Tank (WDL-T2)	B
AX132	Corridor between Unit 1 and Unit 2	C
AX133	South Stairwell	C
AX134	Misc. Waste Tank Pumps	B
AX135	Radwaste Disposal Control Panel	C
AX201	North Stairwell	C
AX202	Elevator Shaft	C
AX203	4160V Switchgear - 2-1E	C
AX204	4160V Switchgear - 2-2E	C
AX205	RB Purge Air Supply and Hy. Crot. Exh.	C
AX206	RB Purge Exhaust Unit B	C
AX207	RB Purge Exhaust Unit A	C
AX208	Aux. Bldg. Exhaust Unit B	C
AX209	Aux. Bldg. Exhaust Unit A	C
AX210	FH Bldg. Exhaust Unit B	C
AX211	FH Bldg. Exhaust Unit A	C
AX212	Decay Heat Surge Tank and Substation	C

APPENDIX 1 (cont'd)

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
AX213	Unit Substations and Access Area	C
AX214	Decon Facility	C
AX215	FH Bldg. Supply Unit	C
AX216	Aux. Bldg. Supply Unit	C
AX217	Access Area	C
AX218	Concent. Waste Storage Tank Room	B
AX219	Inst. Racks & Atmosph. Monitor	C
AX220	Caustic Liquids Mixing Area	C
AX221	Caustic Liquids Mixing Area Corr.	C
AX222	South Stairwell	C
AX223	Air Handling Units General Area	C
AX301	Elevator Shaft and Elevator Machine Rm.	C
AX302	North Stairwell	C
AX303	Elevator and Stairwell Access	C
AX401	Roof	C
AX402	Cooling Water Surge Tanks	C
AX403	Damper Room	C
AX501	RB Spray Pump - 1A	C
AX502	RB Spray Pump - 1B	C
AX503	DH Remov. Cooler and Pump - 1A	C
AX504	DH Remov. Cooler and Pump - 1B	C

APPENDIX 1 (cont'd)

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
FH001	Makeup Suction Valve Room	A
FH002	Access Corridor	C
FH003a	Makeup Discharge Valve Room	A
FH003b	Makeup Discharge Valve Room	A
FH004	Westinghouse Valve Room (Mini-Decay Heat)	C
FH005	Mini-Decay Heat Vault	C
FH006	Decay Heat Service Coolers	C
FH007	Neutral & Reclaimed Boric Acid	C
FH008	Neutralizer Tank Pump Room	C
FH009	Neutralizer Tank Room	C
FH010	Reclaimed Boric Acid Tank	C
FH011	Reclaimed Boric Acid Pump	C
FH012	Neutralizer - Tank Filters	C
FH013	Oil Drum Storage Area	C
FH014	Annulus	B
FH101	MU&P Valve Room	A
FH102	East Corridor	C
FH103	Sample Room	C
FH104	West Corridor	C
FH105	Model Room A	C
FH106	Monitor Tanks & Sample Sink Area	B
FH107	Trash Compactor Area	C
FH108	Truck Bay	C
FH109	Spent Fuel Pool A	B

APPENDIX 1 (cont'd)

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
FH110	SDS Spent Fuel Pool	B
FH111	Fuel Cask Storage	B
FH112	Annulus	B
FH201	East Corridor	C
FH202	West Corridor	C
FH203	Surge Tank Area	C
FH204	SPC Area	C
FH205	Annulus	C
FH301	Upper Spent Fuel Pool Area	C
FH302	SDS Operating Area	B
FH303	Upper SPC Area	C
FH304	Annulus	C
FH305	Spent Fuel Pool Access	C

APPENDIX 1 (cont'd)

AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
RB01	Letdown Coolers Cubicle	A
RB02	Reactor Building Sump	A
RB03	Reactor Coolant Drain Tank Cubicle	A
RB04	Reactor Building Basement (Floor)	A
RB05	Under Reactor Vessel	A
RB06	Letdown Line	B
RB11	Decay Heat Drop Line	A
RB21	Reactor Coolant Pumps	B
RB22	Horizontal RCS Piping	A
RB23	Reactor Vessel	A
RB31	Pressurizer	A
RB32	Steam Generators	A
	Upper Tube Sheet and Tube Blockages	A
	OTSG Tube Surfaces	A
	OTSG Lower Head/J-Legs	A
RB33	Core Flood Tanks - A&B and Drain Lines	A
RB34	Incore Guide Tubes	B
RB35	Plenum	A
RB36	Reactor Vessel Head	A
RB37	Reactor Coolant Hot Legs	A
RB38	Pressurizer Surge Line	A
RB39	Pressurizer Spray Line	B
RB40	Fuel Transfer Canal	B

POST-DEFUELING SURVEY REPORTS

INTRODUCTION

BACKGROUND

On October 17, 1985, the Nuclear Regulatory Commission granted GPU Nuclear Three Mile Island Unit 2 (TMI-2) an exemption from certain requirements for periodic inventory and reporting of special nuclear material (SNM) balance (Reference 1). As a condition of the exemption, GPU Nuclear was required to conduct an assessment of the SNM remaining at TMI-2 following the completion of the defueling effort. This assessment is referred to in the exemption as the "post-defueling survey."

Post-defueling surveys were performed by conducting measurements for residual fuel in those areas, systems, and components in which residual SNM was assumed to exist and for which no further defueling was planned. A compilation of these survey report packages, entitled "Post-Defueling Survey Reports" (PDSRs), documented the estimate of record of the amount of residual SNM and the methodology utilized to determine the quantity of SNM in each case. PDSR sub-reports have been prepared and submitted for all areas, systems, and components listed in Attachment 1. Attachment 2 graphically illustrates the areas, structures, and components covered by the PDSR sub-reports.

The survey results documented in the PDSR sub-reports were not intended to demonstrate the overall adequacy of TMI-2 defueling progress, nor should they be construed as a determining factor for transition to a defueled state as defined in Section 1.3 of the TMI-2 Technical Specifications; transition was based on the TMI-2 Defueling Completion Report which demonstrated the overall adequacy of the defueling progress in the context that inadvertent criticality is no longer possible and defueling has progressed to a point which, based on current technology and ALARA considerations, is defined as the logical endpoint for current defueling activities.

Each PDSR sub-report was submitted to the Nuclear Regulatory Commission (NRC) as measurements, analysis and the internal review and approval process were completed.

PDSR FORMAT

A standard format was utilized for all PDSR sub-reports. Certain adaptations were required based on the type of measurement techniques applied and the varying types of data generated. In general, each PDSR sub-report contains the following sections:

- Summary
- Introduction
- Background
- Methods
- Analysis
- Conclusion
- References
- Tables (as necessary)
- Figures and other attachments (as necessary)

The Summary section of each sub-report precedes the Introduction section and consists of a brief statement including the estimate of record of the amount of fuel remaining, the physical forms of the fuel, the locations of the fuel, a brief statement of the measurement methods used to measure the fuel, the analytical techniques utilized to determine the estimate of record, and a statement of any pertinent conclusions and recommendations for confirmatory measurements, if appropriate.

The Introduction section of each sub-report explains that the individual sub-report is one in a series of reports generated to fulfill the requirements of the SNM Accountability Program. This section also describes the content of the sections to follow.

The Background section of each sub-report describes the relevant area, system, or component by its physical dimensions and location, the boundaries of the sub-report, the intended function of the area, system, or component, its operations and relationship to the accident and/or subsequent cleanup activities, and how the fuel may have relocated to that area.

The Methods section of each sub-report briefly explains the method(s) utilized to measure fuel in each area, system, or component. Due to the complex structure of the components surveyed, the residual SNM quantity may be determined utilizing a combination of direct measurements, sample analyses, volumetric measurements, and engineering analyses. The sub-reports are supported by detail engineering calculations, vendor reports, and research data.

The Analysis section of each PDSR explains how the "estimate of record" of the amount of remaining fuel was calculated. This section describes the techniques utilized to analyze the available data and discusses supporting data, assumptions, calculations, key observations, various models, and the adequacy of the data. This section also addresses the transport analysis, when applicable, for non-isolated areas, systems, and components.

The Conclusion section of each sub-report states the estimate of record of the amount of fuel remaining in that area, system or component; assigns an appropriate uncertainty; discusses the adequacy and confidence level of the conclusion; and makes a statement defining why the estimate of record is reasonable based upon the available data.

METHODS USED TO ASSAY FUEL

The complex configuration of the TMI-2 plant and restricted access in certain areas increased the difficulty of accomplishing residual fuel assessment at TMI-2. Several different measurement methods were required in order to assess the residual fuel for different areas, systems, and components. Gamma detection, neutron detection, video inspections, radiochemical analysis of samples, and alpha detection were among the methods used to perform this assessment. Brief summaries of these methods are provided below.

Gamma Detection - Gamma detection uses either sodium iodide (NaI) scintillation or high purity germanium (HPGe) detector methods. NaI scintillation is used to detect the Ce-144 2.185

MeV gamma ray. The HPGe detector detects both Ce-144 and Eu-154 gamma rays. These isotopes are suitable tracers for fuel debris at TMI-2 and the quantity of fuel is based on either the Ce-144 to fuel ratio or the Eu-154 to fuel ratio (Reference 5). These ratios change by the radioactive decay of Ce-144 (284.3 days) and Eu-154 (8.8 years). The gamma detection method was generally used wherever the level of background radiation was moderate to low (50 mR hr^{-1}).

<u>Neutron Detection</u> - There are two kinds of neutron detection methods: passive counting and active neutron interrogation. The passive counting method detects neutrons directly, employing He-4 detectors for fast neutrons, and He-3 and BF_3 detectors for thermalized neutrons. The active neutron interrogation method detects induced fission neutrons. Sb-Be photoneutrons induce fission reactions. The induced fission neutrons are detected by an He-4 neutron counter. This photoneutron system is calibrated with a known amount of fuel placed at various locations in a mock-up. The neutron detection method is generally used wherever background radiation levels are high in order to confirm measurements made by other methods.

<u>Visual Inspection and Radiochemical Analysis of Samples</u> - If residual deposits are found, visual inspections and debris sampling and analyses may be performed. The quantity of fuel is assessed by debris volume and ratio of fuel to debris. Debris volume is estimated by visual inspections and configuration of components. Ratio of fuel to debris is estimated by debris sampling and analyses. These analyses include volume estimates, weight measurements, and sample analysis for fuel. This method is rarely used alone, but is used in conjunction with other

counting methods in order to help interpret those results.

Alpha Detection - A proportional alpha counter is a gas-filled chamber with two electrodes. This counter detects current between the electrodes in an amount proportional to the amount of ionization. This detector readout assembly is able to select alpha particles from other ionizing radiation by pulse height discrimination. The fuel ratio of alpha activity is 151 μCi/g of fuel, and has a half-life long enough to require only a small decay correction. The alpha detection method is used only for thin films, due to the short range of an alpha particle.

FUEL MEASUREMENT UNCERTAINTIES

Where possible, SNM accountability measurements were conducted in areas that had been physically isolated to prevent transport of SNM into or out of the area, system, or component. However, in some cases, it was not possible or feasible to conduct an SNM measurement of an area, system, or component in an isolated configuration. In these cases, the PDSR included an analysis which demonstrated that there could be no significant transport of SNM into or out of the area, system, or component after the completion of the SNM measurement.[1]

Some of the estimate of record quantities reported for residual SNM quantities are referred to -

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[1] An exception to this was the RV draindown to the fuel transfer canal (FTC). No special precautions were taken to prevent fuel transport to the FTC. However, little fuel was in fact transferred.

as the minimum detectable level (MDL). This means that the measurement technique did not detect a statistically significant number of events (counts) related to SNM. Therefore, the true quantity of SNM believed to be in the applicable area, system, or component is equal to or less than the reported MDL. For example, if the residual SNM quantity is reported to be an MDL of 3 kg, the true quantity of residual SNM could be any value from 0 to 3 kg.

Physical measurements of fuel quantities are subject to imprecisions. The accuracy of post-defueling survey SNM measurements was significantly impacted by the inaccessibility of SNM locations, high background gamma radiation dose rates, unknown SNM distribution characteristics and low neutron emission rates for TMI-2 fuel. The precision of the measurements was negatively impacted by the short half life of the preferred high energy fuel tracer isotope Cerium-144 (2.1 MeV gamma emitter) as compared to the lower energy of Europium-154 (1.1 MeV gamma emitter), which is a fuel tracer isotope with a longer half life (i.e., 8.8 years). Uncertainties were also due in part to the combined effects of counting statistics, representativeness of samples to the whole, high radiation interference backgrounds which elevate MDLs, complex and undefined fuel distribution geometries, lack of personnel access requiring use of remote measurement techniques, and varying signal absorption rates due to the presence of structural members and blanketing layers. The uncertainty of the total "estimate of record" tends to be dominated by the largest single measurement aggregate error or largest MDL. Laborious measurements at precision values much smaller than the largest aggregate error would have consumed measurement resources unnecessarily. The principles of ALARA suggest that measurement effort be proportioned according to the absolute uncertainty of fuel in any measurement location. The analysis of the Reactor Vessel residual fuel quantity yielded an uncertainty of ± 370 kg. Therefore, state-of-the-art measurements of other reactor coolant and auxiliary system components of much smaller fuel content (e.g., 3 kg) would have included errors that were diminishingly small compared to the overall uncertainty set by the Reactor Vessel. Since the component bias estimate errors were independent, they were combined in quadrature (i.e., the square root of the sum of the squares of the individual biases) to obtain their contribution to the overall error.

Because of the inability to control some of the variables described above, post-defueling survey measurements at TMI-2 were attended by significant uncertainties. These uncertainties were minimized to the extent possible by the judicious selection of measurement techniques and a graded application of resources. Areas, systems, and components known or believed to contain SNM quantities greater than 1 kg were subjected to significantly greater measurement resources than locations where less than 1 kg of SNM was indicated. Nonetheless, the variables which negatively impact precision and accuracy of post-defueling survey SNM measurements resulted in some relatively large uncertainties. The PDSR sub-reports include the estimate of record of residual SNM and associated uncertainty. Therefore, measurement reports contain a calculated uncertainty and/or qualitative discussion of the adequacy of the measurement and identify potential source(s) of measurement error. The use of the term "estimate of record" in this context conveys that the quantity reported is the most realistic value for the quantity of SNM in the area, system, or component based upon the available measurement data.

REFERENCES

1. NRC letter, "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d) and 70.53," dated October 17, 1985.

2. GPU Nuclear Administrative Procedure 4000-ADM-4420.02, "TMI-2 Core and Special Nuclear Material Accountability Program."

3. TPO/TMI-189, "Reactor Vessel Post-Defueling Special Nuclear Material Survey," Revision 0, January 1988.

4. TPO/TMI-187, "Instrument Selection for Residual Fuel Measurements," Revision 0, January 1987.

5. GPU Nuclear Technical Bulletin 86-41, "Ce-144, Eu-154, and Eu-155 as Tracers for Fuel Debris," Revision 2, February 1988.

6. GPU Nuclear Policy/Plan Procedure 4000-PLN-4420.02, "SNM Accountability Plan."

ATTACHMENT 1

LIST OF SECTIONS

SECTION TITLE	GPUN LETTER NUMBER	DATE OF SUBMITTAL
Plenum	4410-88-L-0162	09-30-88
Letdown Coolers	4410-89-L-0097	09-22-89
Pressurizer	4410-89-L-0097	09-22-89
Reactor Vessel Head	4410-90-L-0019	03-14-90
Reactor Building Basement	4410-89-L-0097	09-22-89
'A' and 'B' OTSG	C312-92-2064	08-20-91
Auxiliary and Fuel Handling Buildings	C312-91-2045	06-07-91
Reactor Building Miscellaneous Components*	C312-91-2052	06-18-91
Reactor Coolant System**	C312-92-2055	07-03-91
Reactor Vessel	C312-93-2004	02-01-93

* Includes Incore Guide Tubes in "A" D-ring, Fuel Transfer Canal/Transfer Tubes, Upper Endfittings, Core Flood System, Tool Decon Facility and the Drain Line from the Tool Decon Facility, Temporary RV Filter System, Defueling Water Cleanup System, Reactor Coolant Drain Tank, and Defueling Tools.

** Includes Cold Legs, Hot Legs, Reactor Coolant Pumps, Decay Heat Line, Core Flood Lines, and Pressurizer Lines.

ATTACHMENT 2