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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:

The Post-Defueling Survey Report (PDSR) for the 'A' and 'B' Once-Through Steam Generators (OTSGs) was submitted by GPU Nuclear letter 4410-90-L-0019, dated March 14, 1990. The attached revision to that PDSR was necessitated by the results of analysis of "fluff" samples acquired from the OTSG J-legs and lower heads subsequent to the initial submittal of the PDSR. Included are a revision summary and the revised OTSG PDSR with change bars denoting the revisions.

Sincerely,

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

EDS/dlb

cc: T. T. Martin - Regional Administrator, Region I
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TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS

(REVISION 1)

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TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS

(REVISION 1)

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS (REVISION 1)

REVISION SUMMARY

The TMI-2 Post-Defueling Survey Report (PDSR) for the 'A' and 'B' Once-Through Steam Generators (OTSGs) has been revised due to the latest fluff sampling acquisition from the OTSG J-Legs and lower heads. The details for this revision are contained in Engineering Calculation 4249-3224-91-007.

The estimate of record of the amount of uranium dioxide (UO_2) reactor fuel remaining in the 'A' and 'B' OTSGs has been increased from 4.1 kg to 7.1 kg ±33% in the 'A' OTSG and from 51.4 kg to 55.2 kg ±17% in the 'B' OTSG for the following reasons:

1. New samples were taken from the OTSG lower heads and J-Legs which indicate that the fuel debris remaining in the OTSGs is similar to the fuel debris previously taken from the lower head of the reactor vessel.

2. Results from fuel debris samples taken from the lower head of the reactor vessel show more Cs-137 activity per gram of debris than did the fuel debris sample data used in the original calculation.

3. A different fuel debris density and weight percent fuel fraction was also incorporated. The revised values were based on the latest fluff samples taken from the lower head and J-Legs (Reference Engineering Calculation 4249-4400-91-004).

Revision of the Isotopic Cs-137 Activity

The original calculation used Cs-137 activity data determined for fuel debris samples taken from the pressurizer. The latest fuel samples taken from the OTSGs show a higher activity per gram which results in a greater calculated exposure rate for each Microshield computer run. The original calculation used 3.665 E+2 microcurie per gram for Cs-137 while the new sample data results for the lower head debris indicate 1.1290 E+3 microcuries per gram.

Revision of the Fuel Debris Density Values

The fuel sample data used in the original calculation were based on a combination of both pressurizer and reactor vessel material densities of 2.0 and 2.5 grams per cm^3. The latest sample data taken from the lower head and J-Legs of the OTSGs indicates that the debris material density is closer to 3.2 g/cm^3. Therefore, the revised calculation utilized the new material density of 3.2 g/cm^3 for all locations in the OTSGs.

Revision of the UO_2 to Fuel Debris Ratio

The latest fluff sample data for fuel debris taken from the J-Legs and lower heads of the OTSGs indicate that the UO_2 to fuel debris ratio was 0.575 or 57.5% by weight. This new fuel fraction was the average value determined from twenty-eight (28) samples obtained from the OTSGs (Reference GPUNC Calculation 4249-4400-91-004).

Conclusion

The above changes result in a small increase in the estimate of record for the OTSGs from 55.5 kg to 62.3 kg UO_2 with an uncertainty of approximately ±16% (one sigma).

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS

SUMMARY

The estimates of record of the amount of uranium oxide (UO_2) remaining in the Three Mile Island Unit 2 (TMI-2) 'A' and 'B' once-through steam generators (OTSGs) are 7.1 kg ± 33% and 55.2 kg ± 17%, respectively. All statistical uncertainties are expressed as ± one sigma limits for combined increments calculated or taken to be one sigma (defined as one standard deviation). The UO_2 is distributed as follows:

	'A' OTSG	'B' OTSG
Upper Tube Sheet	1.4 kg ± 21%	36.0 kg ± 18%
Tube Bundle	1.7 kg ± 48%	9.1 kg ± 48%
Lower Head	1.4 kg	2.2 kg
RCP-1 J-Leg	1.5 kg ± 55%	1.9 kg ± 55%
RCP-2 J-Leg	1.1 kg	6.0 kg
TOTAL	7.1 KG ± 33%	55.2 kg ± 17%

The 'B' upper tube sheet was characterized by neutron activation measurements. Fuel estimates of the 'A' upper tube sheet, the 'A' and 'B' OTSG tube bundles, the lower heads, and the J-legs were projected using gamma radiation measurement data. Earlier fuel estimates based on independent transuranic analysis of surface scrapings (Reference 1) and on direct alpha particle measurements of a limited portion of the 'A' OTSG tube bundle (Reference 2) indicate that the present values are conservatively high.

The quantities of UO_2 remaining in the 'A' and 'B' OTSGs are <0.5% and <5.5%, respectively, of the anticipated residual UO_2 inventory* for the entire TMI-2 facility in Mode 2.

Independent foil activation measurements were performed by a Battelle Northwest Laboratory group on both OTSG lower head and J-leg locations. These measurements confirmed the relatively low fuel deposition in these locations (Reference 4).

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* The anticipated residual UO_2 inventory is as defined in the PDMS Safety Analysis Report and the PDMS Programmatic Environmental Impact Statement and is based on the assumption that the defueling program goal to remove more than 99% of the original core inventory of UO_2 is achieved.

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

1.0 INTRODUCTION

This report presents the analysis of the residual inventory of uranium dioxide (UO_2) in the Three Mile Island Unit 2 (TMI-2) 'A' and 'B' once-through steam generators. It is one in a series of reports generated to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 5). All statistical uncertainties are expressed as $\pm$ one sigma limits (defined as one standard deviation).

Section 2.0, "Background", describes the physical attributes of the 'A' and 'B' OTSGs and their relationship to the accident and subsequent cleanup activities. The boundaries for this report are also discussed.

Section 3.0, "Methods", describes how fuel measurements and sample data were used to produce the estimates of record. Copper activation foils were primarily used to determine the estimate of record of the amount of fuel on the 'B' OTSG upper tube sheet. Separate copper foil activation calibrations were performed in the 'A' and 'B' OTSG upper heads with a known neutron source (Am-Be). The calibration was used to compensate for uncertain scattering and neutron absorption (Reference 6). Fuel measurement data for the 'A' upper tube sheet, the OTSG tube bundles, the lower heads, and the J-Legs consisted primarily of gamma radiation measurements, which were correlated to actual reactor coolant system (RCS) sample analysis results to produce the estimate of record. The results of using gross gamma measurements to characterize fuel present as films and as tube blockages was compared to earlier fuel analysis by direct alpha measurements and by scrapings taken from the 'A' and 'B' OTSG upper head manway and inspection hole cover plate inserts (Reference 1). Direct alpha film measurements of the top 20 feet of nine 'A' OTSG tubes also were used for comparison (Reference 2).

Section 4.0, "Analysis", explains how the estimates of record of fuel in the 'A' and 'B' OTSGs were calculated based on the fuel measurement and sample analysis data and compared to calculated exposure rates for known activity concentrations modeled with the Microshield computer program (Reference 7).

Section 5.0, "Conclusion", presents the estimates of record for the amount of UO_2 remaining in the 'A' and 'B' OTSGs, and states supporting rationale leading to the conclusion that the estimates are reasonable based upon the available sample analysis data and the fuel measurement techniques employed.

2.0 BACKGROUND

The OTSGs are vertical, straight tube and shell boilers in which the reactor coolant (the heat source) is on the tube side and the secondary coolant is on the shell side (Figure 1). They are made of high strength carbon steel and all surfaces that contact the primary coolant are stainless steel or inconel. The TMI-2 'A' and 'B' OTSGs were used for transferring heat from the reactor coolant system to the secondary system.

As a result of the TMI-2 accident, fuel was transported through the hot legs into the OTSGs by a variety of pathway mechanisms which include drain and refill of the steam generator and inlet pipings, forced circulation from the coolant pumps, natural circulation, and fluid movement resulting from energy releases into the reactor vessel (Reference 8).

Fuel fines and sediment material were transported to the OTSGs by reactor coolant through the hot legs to the steam generator upper head and tube sheet. Much of the larger particulate material settled out on the upper tube sheet with the finer material being transported through the tube bundles to the lower head and outlet piping.

This Post-Defueling Survey Report (PDSR) includes characterization of the 'A' and 'B' OTSG upper tube sheets, tube bundle regions, lower heads, and the associated coolant outlet J-legs.

Radiological Environment

As a result of the March 1979 accident, fuel debris and fission products were released and deposited in the 'A' and 'B' OTSGs. Most of the larger debris material settled out on the upper tube sheets with smaller material being transported to the outlet piping. During late 1987, the debris material on the upper tube sheets was removed by vacuuming and pick-and-place operations. The total person-hours and person-rem expended for defueling the 'A' upper tube sheet was 108 and 4.3, respectively. The total person-hours and person-rem expended for defueling the 'B' upper tube sheet was 153 and 9.4, respectively. Currently, radiation levels in the 'A' OTSG upper tube sheet range from 1-3 R/hr. In the 'B' OTSG, levels range from 30-100 R/hr. The general area exposure rate in the 'A' and 'B' D-rings ranges from 50-800 mR/hr. In general, higher exposure rates and significant loose surface contamination exists in the 'B' D-ring area.

The total person-hours and person-rem expended for the OTSG residual fuel measurement program were:

OSTG Lower Head and J-Legs	Person-hours	Person-rem
'A'	164	7.4
'B'	140	8.5

Upper Tube Sheet	Person-hours	Person-rem
'A'	160	7.1
'B'	348	21.9

The methodology covered by this report is considered to be optimized toward keeping radiation exposure as low as reasonably achievable (ALARA), due to the nature of the 'A' and 'B' OTSGs. Due to the radiologically hostile nature of the OTSGs and their environment, further refinements are not considered to be beneficial in light of the cost of exposure and labor.

3.0 METHODS

As part of the Special Nuclear Material (SNM) Accountability Program at TMI-2, gamma measurement surveys and neutron activation measurements were performed for fuel characterization of the 'A' and 'B' OTSG upper tube sheets, tube bundles, lower heads, and associated cold legs during September 1988 to mid-January 1989.

SNM accountability of the OTSGs presented a difficult task due to system inaccessibility and high radiation levels, requiring the selection of indirect fuel measurement techniques. Each entire OTSG, including the upper tube sheets, tube bundle region, lower head, and associated cold legs, required fuel characterization. Due to the inaccessibility and presence of primary coolant in the tube bundle region, lower head, and cold legs, an alternative method for estimating fuel was developed. This method consisted of obtaining gross gamma measurements, correlating this data to predicted gamma exposure rates for fission products sequestered in surface films, and for clustered fuel debris blocking tubes with a composition assumed to be similar to samples collected from the OTSG upper tube sheet, and estimating fuel content. Different types of debris in various geometries were modeled using the Microshield computer program to generate gamma exposure rates based on known isotopic quantities. The actual in situ gamma radiation measurements were then compared with the computer-generated values to estimate fuel content.

OTSG Characterization

The OTSGs were characterized in two phases. The first included measurements of the lower heads and associated outlet J-legs. The second phase included measurements of the upper tube sheets and the tube bundle region.

Phase One

During phase one of the measurement program, CoPhysics Corporation, under contract to GPU Nuclear, fabricated several fuel measurement strings to obtain gamma measurements in the 'A' and 'B' OTSG lower heads and associated J-legs. The fuel measurement strings consisted of 100-ft long by 0.5-in. diameter polypropylene tubes which contained, at the front ends, two Geiger-Müller (GM) probes spaced at 6-ft intervals. The fuel measurement strings were deployed through a guide tool mounted on the steam generator upper tube sheet and were pushed down through a 56-ft long steam generator tube (0.56-inch inside diameter) to the lower head. They were then pushed an additional 15 ft further into the associated J-legs. Additionally, miniature lights and a Welch Allyn videoprobe were deployed through adjacent tubes to document placement of the strings within the respective RCS component and within any observed debris.

Each fuel measurement string also contained copper foils for independent measurement of fuel-related material by a DOE national laboratory. The foil analysis was performed and reported by the Pacific Northwest Laboratory (Reference 9).

During deployment of the fuel measurement strings, stabilized assay meters (SAM-2s) were connected to the GM probes to obtain dual gamma radiation readings at every 2 ft in the tube bundle region and at 1-ft intervals in the lower heads and J-legs.

The fuel measurement strings were visually inspected to verify the string positioning and contact with debris material within the steam generator lower heads and J-legs. The debris in the lower heads and J-legs was a low-density material and was easily suspended while moving the videoprobe which resulted in poor visibility. This low-density material was uniformly distributed with a few areas containing granular material with a higher density and greater fission product content that corresponded well to the higher exposure rates.

Phase Two

Characterization of the OTSG upper tube sheets in phase two of the measurement program involved placement of copper foils inside the 'A' and 'B' OTSG upper tube sheets to estimate the amount of residual fuel remaining by neutron activation of copper foils. Activation is generated by neutrons emitted from residual fuel. Following activation, the foils were removed and counted in a coincidence system. This system determines the quantity of β^+ emissions from the activated copper foils. This process was repeated with an Am-Be neutron source to calibrate the foil and detector system for the scattering and neutron loss environment of the OTSG upper heads. Using the foil activation data, the average neutron activation flux was calculated to determine the fuel estimate of record for the upper tube sheets.

Characterization of the tube bundles for fuel debris blockages involved the deployment of a single measurement string into a total of 52 evenly-spaced steam generator tubes out of 15,531 tubes. The probing locations were selected based on the effective radius of sensitivity of a gross gamma detector probe to a 1-inch long debris blockage. An 8-inch effective horizontal radius was calculated by extensive modeling with the Microshield computer program which resulted in 52 deployment locations needed for full coverage characterization of the 9.5-ft diameter tube bundle region. The one inch blockages were assumed to be vertically displaced 5.5 inches from a given measurement point and always at a horizontal radius of 8 inches. This geometry will maximize fuel estimates by this method since the assumed location is roughly midway between measurement points. The Phase 2 fuel measurement string contained six GM probes at the front end, spaced at 1-ft intervals. The string was deployed into a steam generator tube similar to that used in phase one, and gamma measurements were recorded at 5-ft intervals down through the total 56-ft tube length.

The gamma readings from all GM probes were collected simultaneously at each 5-ft interval in the tube using 2 IBM personal computers, computer-operated Counter/Timer Boards, LabTech Notebook™ software (Reference 10), and preamplifier-amplifier-discriminators (PADs) connected to the GM probes. Two computer systems were installed to enable remote data collection from outside of the TMI-2 Reactor Building. The host computer system was installed in the Reactor Building along with a nuclear instrument module regulated power supply (NIM BIN) containing six PADs and a high-voltage power supply. The remote computer system was installed in the TMI-2 Command Center and connected to the host computer system by standard twisted pair cables and two Black Box Short Haul modems (Reference 11). PC-Anywhere™ (Reference 12) software was used to provide remote access to the host computer to operate the data collection system.

The data collection system installed in the host computer included LabTech Notebook™ DAS Software and four Metrabyte Counter/Timer Boards™ (Reference 13). LabTech Notebook™ provided easy-to-use menu-driven data acquisition capabilities along with real time analysis, display, and process control. In our application, LabTech Notebook™ was used to collect data using two of the four Metrabyte Counter/Timer Boards™. The Metrabyte Counter/Timer Boards™ are short slot boards with 5 independent, 16 bit programmable counters which operate up to 7 MHz for event counting, timing, and frequency counting. The input voltage levels used by these boards are standard TTL level 0 to 5 volts. A universal terminal panel was used to interface with the PADs. The Canberra PADs Model 814A (Reference 14) were internally modified to provide an output pulse amplitude of 3 volts. The counter/timer setup combined with the PADs worked very well.

The fuel measurement strings used in phase two of the measurement program were fabricated, assembled, and calibrated in the Rad Instrument Shop at TMI. The GM probes were connected with RG178U coaxial cable and installed into polypropylene tubing. At the end of each string, the coaxial cables were terminated into a utility box which provided strain relief for the smaller diameter cables, signal output connection, and a common high-voltage input. Each GM probe in a fuel measurement string was calibrated on a range using a 600 uCi Cs-137 source and a calibration curve was generated correlating exposure rate to counts per minute.

4.0 ANALYSIS

The data collected during phase one of the measurement program was reduced and evaluated and fuel quantities in the 'A' and 'B' OTSG lower heads and J-legs were estimated by comparing the actual in situ exposure rates to calculated exposure rates for known activity concentrations modeled with the Microshield computer program. The gamma radiation profiles for both the 'A' and 'B' OTSG lower heads and J-legs are shown in Figures 2 and 3.

In addition to the gamma radiation measurements performed, visual inspection of the lower heads and J-legs was used to confirm string positioning relative to contact with debris material and for estimating debris depth. The debris observed in the 'A' OTSG was a very low-density material and was easily suspended while moving the videoprobe which resulted in poor visibility. This low-density material was basically uniformly distributed with a few areas that were fluff zones containing material with crust-like surfaces. These areas (the 8 foot locations in Figure 2) corresponded well to the higher dose rates (Reference 15).

The debris observed in the 'B' OTSG lower head was also a very low-density material of minimal depth (less than 1/8 inch). The debris material in the J-legs was fairly uniformly distributed. A few areas exhibited variable debris depths in the RCP '2B' J-leg (the 9 foot location in Figure 3) which contained small gravel-like debris and corresponded to the highest exposure rate (Reference 16).

The amounts of fuel in the 'A' and 'B' OTSG lower heads and J-legs were estimated using several Microshield calculations to model the debris material and geometry in these areas. The composition of the debris was based on analysis of samples taken from the J-legs of both steam generators (Reference 17).

Additionally, for calculation of the source volume, it was assumed (based on Phase 1 visual inspections) that the debris material in the OTSG lower heads was a uniformly distributed low-density material of minimal depth within a 6-foot diameter source. In the J-legs, the debris material was taken to be uniformly distributed (based on Phase 1 visual inspections), with a few areas varying in depth throughout a 9-foot segment of piping (Reference 18). Based on these assumptions, the fuel estimates for the 'A' OTSG lower head and J-legs were 1.4 kg and 2.6 kg, respectively, and for the 'B' OTSG lower head and J-legs were 2.2 kg and 8.0 kg, respectively. Tabulated fuel estimates are shown in Table 1.

The foil activation data collected during phase two of the measurement program for characterization of the 'A' and 'B' OTSG upper tube sheets were evaluated. Four copper foils were placed in the 'B' OTSG and two copper foils were placed in the 'A' OTSG. A copper foil was also exposed in the Reactor Building (RB) above the D-ring to determine the amount of background activation due to environmetnal neutrons.

The difference in activation flux between the 'B' OTSG upper tube sheet and the RB background gives the activation flux due to residual fuel on the 'B' OTSG upper tube sheet. The estimate of record of residual fuel on the 'B' OTSG upper tube sheet was based on two determinations with resultant fuel estimates of 36.0 kg ± 18% and 35.3 kg ± 16% (Reference 6). The 36.0 kg value was chosen to be the estimate of record. This compares to previously determined values by visual-inspection of 28.9 kg (Reference 20) and gross gamma of 53.9 kg (Reference 21).

From the 'A' OTSG foil calibration by the Am-Be source, an effective flux was determined. Based on a comparison of exposure rates between the 'A' and 'B' OTSG upper tube sheets, 3 R/hr and 126 R/hr, respectively (Reference 19), and the 36 kg fuel estimate for the 'B' OTSG upper tube sheet, the 'A' OTSG upper tube sheet fuel estimate of record is 1.4 kg ± 21%. This value is based on correcting the exposure rate measured at 2 ft above the 'A' OTSG tube sheet to the exposure rate predicted at 1 ft for disc source geometry of the 'B' OTSG tube sheet. The calculated value is 5 R/hr. The ratio of the calculated 'A' OTSG value to the measured 'B' OTSG value of 1 ft separation was used to provide the estimate of record (1.4 kg) from the assayed quantity of 36 kg for the 'B' OTSG.

The measurement data collected for characterization of the 'A' and 'B' OTSG tube bundles was reduced and evaluated to determine the possibility of debris blocking the OTSG tubes. Fuel quantities were estimated by comparing the actual in situ exposure rates to calculated exposure rates based on a one-inch long debris blockage and known activity concentration modeled with the Microshield computer program. It was assumed that the debris material present in a potential blocked tube was most similar in activity and density to the 'B' OTSG upper tube sheet debris material. As previously mentioned, gamma measurements were obtained in the OTSG tube bundles using fuel measurement strings which consisted of six GM probes installed in polypropylene tubing and connected to associated electronics. Calibration response curves were generated for each GM probe in each measurement string. The response characteristics of each GM probe were very uniform and linear within the 100-5000 mR/hr range. Therefore, the response curves for each GM probe in a single string were averaged and calibration surveys and response equations were generated. The use of the averaged calibration data, along with the Lotus 1-2-3 Computer Program™ (Reference 22), facilitated the data reduction of approximately 7500 data points.

Probe number one in all fuel measurement strings was shielded with cadmium to eliminate any response to high energy beta and excess response to low energy gamma. Fuel measurement strings were deployed into the tube bundles in intervals to overlap measurements with probes 1 and 6. The GM probe count data for probes 1 and 6 were reviewed and correction factors for over-response were calculated. These correction factors were applied to probes 2-6 and incorporated into the Lotus 1-2-3 spreadsheet. The 'A' and 'B' OTSG GM probe measurement locations, mean measured exposure rates, and standard deviations are shown in Figures 4 and 5.

Analysis of the 'A' and 'B' OTSG fuel measurement gamma probing data indicates that there are no significant bright spots (i.e., high exposure rates above background) within the tube bundle region that are attributable to fuel blockages. In comparing the radiation profiles for both steam generators, the exposure rates in the dry region of the 'B' OTSG were approximately a factor of 4 higher than the 'A' OTSG. Similarly, the exposure rates in the wet region of the 'B' OTSG were approximately a factor of 2 higher than the 'A' OTSG. High radiation areas within the upper 6 feet of the 'B' OTSG tube bundle are largely due to the contribution from the upper tube sheet. Additionally, high exposure rates were also associated with the water interface, possibly corresponding to radioactive cation capture in the form of a "bath tub" ring. Exposure rates for other areas were relatively uniform and within the calculated deviation.

Previous measurements of fission products and transuranics were performed (Reference 1) on the inside stainless steel surfaces of one upper head nominal 16 inch manway insert for each OTSG. The results (Reference 23) were used to predict that 0.2 and 2.2 kg of UO_2 could be evenly plated out in films in the 'A' and 'B' OTSGs, respectively. Direct alpha measurements (Reference 2) of the top 20 feet of nine roughly evenly-spaced 'A' OTSG tubes were performed. Based on the alpha measurements, approximately 0.09 kg of UO_2 were predicted to be held on all internal surfaces of the 'A' OTSG tubes. This value is in reasonably good agreement with the earlier projection based on the 'A' OTSG manway insert plate.

The second potential fuel compartment, namely fuel debris in blocked OTSG tubes, was evaluated using gross gamma measurements within approximately 52 evenly-spaced tubes in each generator. The gamma field within a tube is taken to be the sum of background and of possible fuel debris located close enough to be measured. Background includes cosmic ray, contamination external to the OTSG, contamination on the probe string (Figure 6), Cs-137 activity on the inside surface of the tubes, and the 0.07 uCi/cc Cs-137 activity in the primary water wetting the lower portion of the tubes. The Cs-137 activity per unit stainless steel surface (Reference 1), reduced for the lower corrosiveness of the inconel tube material, was used to predict the general exposure rate within the tube bundles from the tubes (Reference 23). The values are approximately 670 and 375 mR/hr for the 'A' OTSG dry and wet parts and 1185 and 705 mR/hr for the 'B' OTSG dry and wet parts, respectively. Using these background values and the contamination on the probes and in the water, no statistically significant positive values in the 'A' and in the 'B' OTSGs were indicated.

The inferential method adopted for determining fuel is a modification of the conventional calculation for the lower limit of detection (LLD). Normally, the LLD is calculated for 95% confidence. A smaller confidence level of approximately 60% was used to partly compensate for two geometry assumptions that lead to unlikely maximum values. They are:

1. Blocked tubes only occur at the maximum effective radius of 8 inches.

2. The point of blockage is always vertically displaced from a measurement location by 5.5 inches.

From earlier discussions, all or nearly all of the gross gamma fields within the OTSGs were due to Cs-137 activity on the primary side surfaces. A leadscrew taken from the TMI-2 reactor vessel head was characterized for adherent films (Reference 24). Variation of factors of 2-3 are reported within distances of a few feet. Due to this expected variability, the OTSG gamma measurements were averaged for each five-foot increment of insertion. Standard deviations were calculated for the variances between the 52 tube results for given insertions, $\sigma_{(A-L)}$. The estimate of record was based on converting the $\sigma_{(A-L)}$ values to UO_2 weight (Reference 25), multiplying by 52 to account for all space at a given depth of insertion, and summing the results for insertion increments A to L as:

$$UO_2 = 52 \sum_{i=A}^{L} \sigma_i = \text{estimate of record.}$$

The amount of UO_2 predicted to be present in the tubes as films was not added to the total since it was assumed that the gross gamma method already accounted for this increment.

A large percentage of the variation in the rate could be attributed to variations in corrosion layer thickness largely containing Cs-137. It was assumed that all variances are due to fuel.

Based on the previously stated assumptions, the modeled steam generator fuel debris blockage, and the corresponding dose rates, fuel estimates for the 'A' and 'B' OTSG tube bundles were calculated. The estimates of record of total fuel debris deposited in the 'A' and 'B' OTSG tube bundles are 1.7 kg and 9.1 kg, respectively (Reference 25).

5.0 CONCLUSION

The use of the fuel measurement strings for obtaining gamma measurements and the copper foils for neutron activation proved to work very well considering the rough handling and strenuous work conditions encountered in the Reactor Building. Overall, the fuel measurement strings facilitated the deployment of the GM probes into the difficult-to-access areas of the OTSGs and were less dose-intensive to personnel deploying equipment and performing measurements than alternate methods such as probing statistically significant tube quantities of approximately 7000 tubes per OTSG. The estimates of record of the total amount of UO_2 remaining in the 'A' and 'B' OTSGs are approximately 7.1 kg ± 33% and 55.2 kg ± 17%, respectively. The uncertainty associated with these fuel estimates (except for the 'B' OTSG upper tube sheet foil activation fuel estimate of 36.0 kg ± 18%) is based on modeling geometry errors, source calibration errors, GM probe response errors, and cesium-fuel ratio errors. The total uncertainty is based on the square root of the sum of the squares of the individual uncertainties associated with each component of the analysis. The measurement uncertainty associated with the above fuel estimates varies from 18 to 55% depending on the component.

This estimate of record is derived from existing measurement and sample analysis data. It is expected to remain static since it is not expected that additional quantities of water from the reactor vessel will be added or cycled through the OTSGs.

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14. Pe Amp/Amp/Disc Model 814/A. Canberra Industries, Inc. One State Street. Meridan, CT 06450.

15. M. E. Greenidge. 'A' OTSG Lower Head and J-Leg Fuel Measurements. Memorandum 4730-88-6057 to G. R. Eidam. Middletown, PA: GPU Nuclear Corporation. 23 September 1988.

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17. Evaluation of Fluff Samples - OTSG 'A' & 'B'. Calculation No. 4249-4400-91-004. Middletown, PA: GPU Nuclear Corporation. 8 April 1991.

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19. GPU Nuclear Radiological Survey. File Code No. RB600-3841-87. Middletown, PA: GPU Nuclear Corporation. September 1987.

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21. P. J. Babel. OTSG Upper Tube Sheet Reactor Fuel. Calculation No. 4530-3224-88-019, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 27 July 1988.

22. Lotus 1-2-3. Lotus Development Corporation. 55 Cambridge Parkway. Cambridge, MA 02142.

23. Calculation # 4800-3224-89-130 OTSG Tube Bundle Dose Rate. 30 November 1989.

24. K. J. Hoffstetter, et al. Chemical Analysis and Test Results For Sections of the TMI-2 H-8 Leadscrew. TPO/TMI-103. February 1984.

25. OTSG Tube Bundle Fuel Estimates. Calculation No. 4800-3224-89-006. Middletown, PA: GPU Nuclear Corporation. 15 May 1989.

TABLE 1

ONCE-THROUGH STEAM GENERATOR

FUEL ESTIMATES OF RECORD

	'A' OTSG	'B' OTSG
Upper Tube Sheet	1.4 kg ± 21%	36.0 kg ± 18%
Tube Bundle	1.7 kg ± 48%	9.1 kg ± 48%
Lower Head	1.4 kg	2.2 kg
RCP-1 J-Leg	1.5 kg ± 55%	1.9 kg ± 55%
RCP-2 J-Leg	1.1 kg	6.0 kg
TOTAL	7.1 kg ± 33%	55.2 kg ± 17%

FIGURE 1

TMI-2 ONCE-THROUGH STEAM GENERATOR

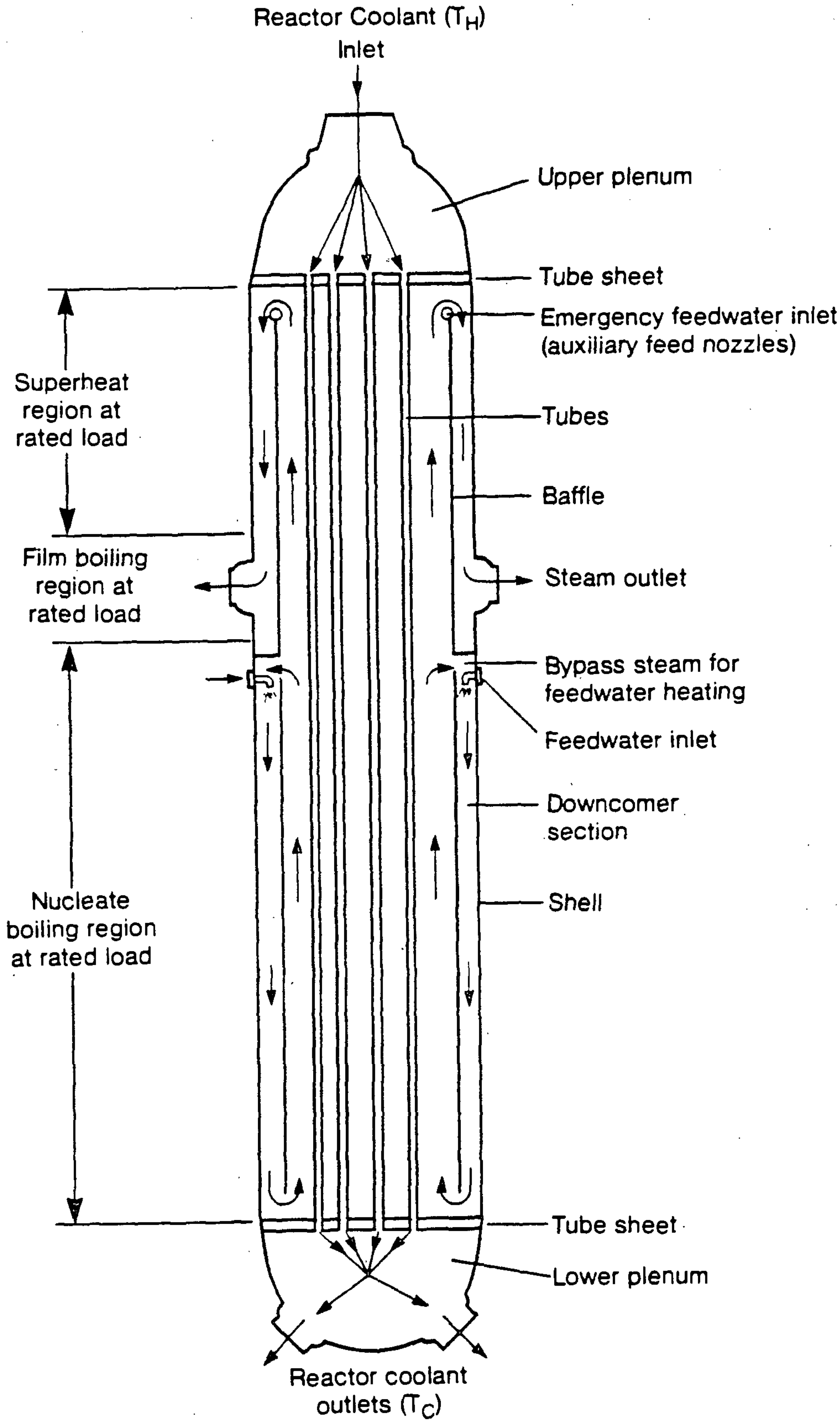
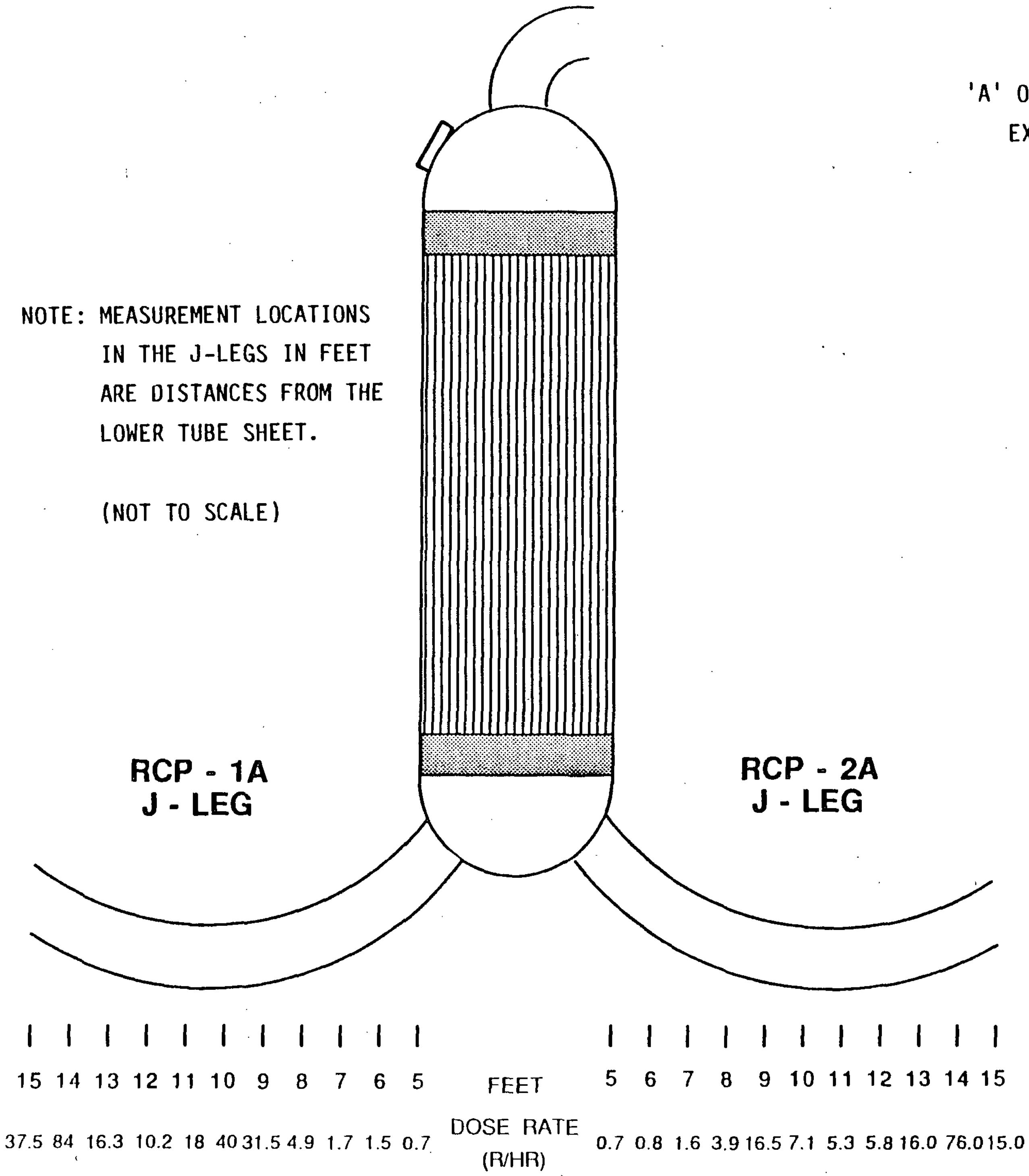
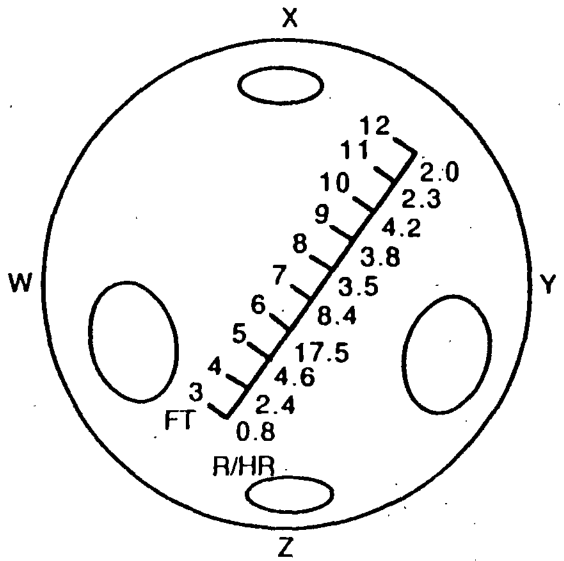


FIGURE 2
'A' OTSG LOWER HEAD AND J-LEG
EXPOSURE RATE PROFILES





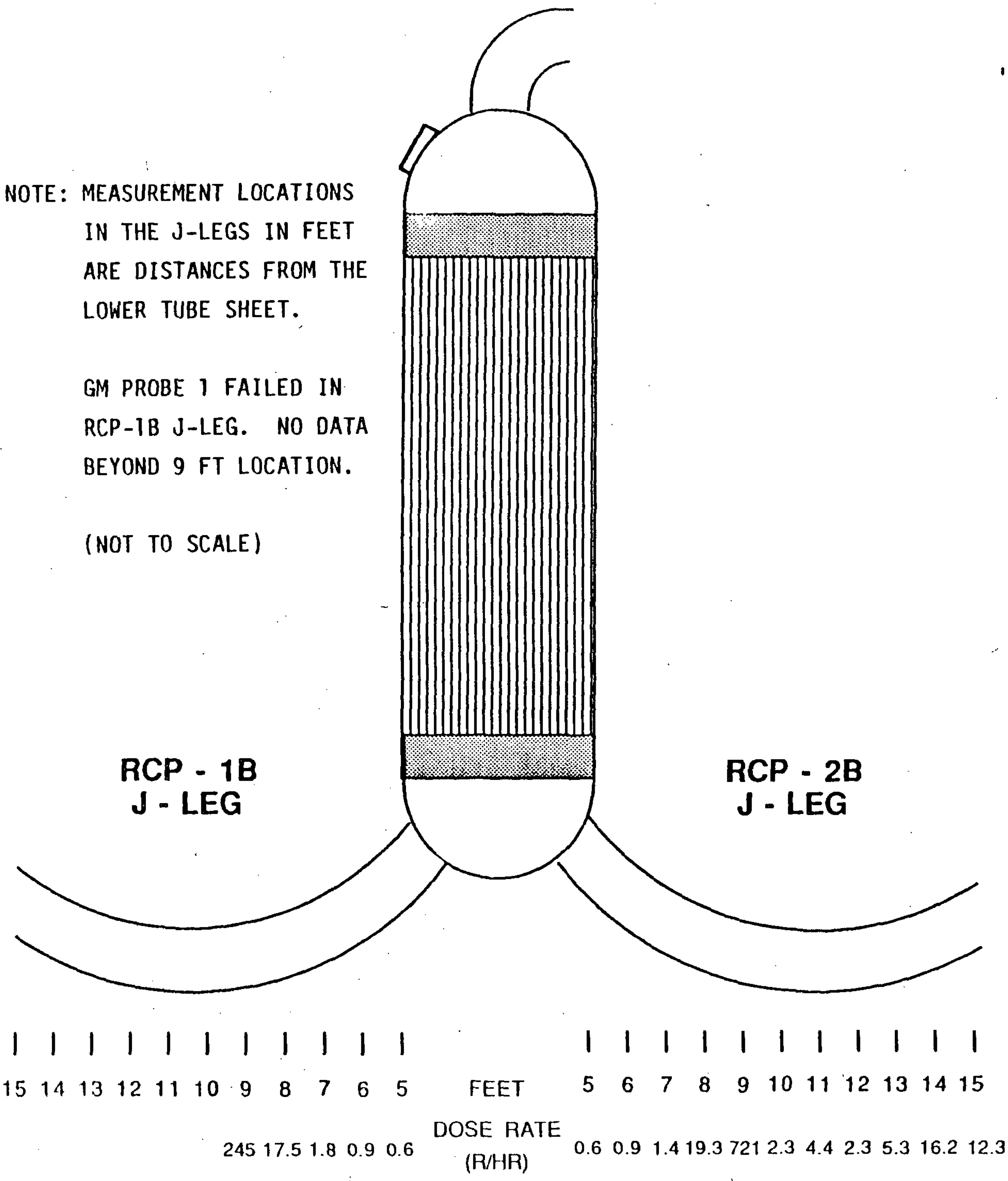
LOWER HEAD - PLAN VIEW

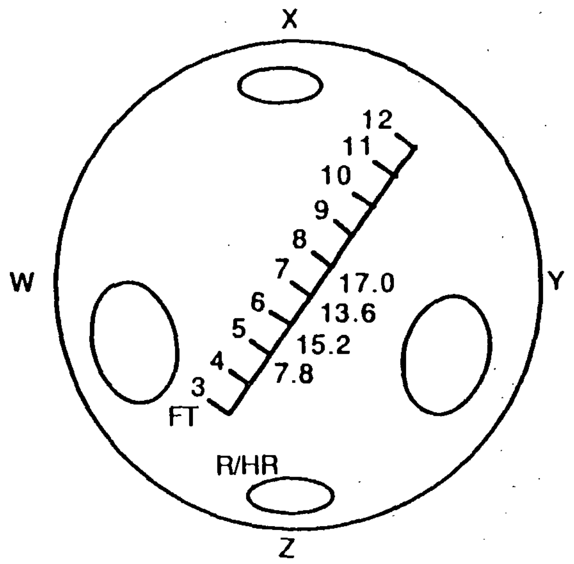
FIGURE 3
'B' OTSG LOWER HEAD AND J-LEG
EXPOSURE RATE PROFILES

NOTE: MEASUREMENT LOCATIONS IN THE J-LEGS IN FEET ARE DISTANCES FROM THE LOWER TUBE SHEET.

GM PROBE 1 FAILED IN RCP-1B J-LEG. NO DATA BEYOND 9 FT LOCATION.

(NOT TO SCALE)



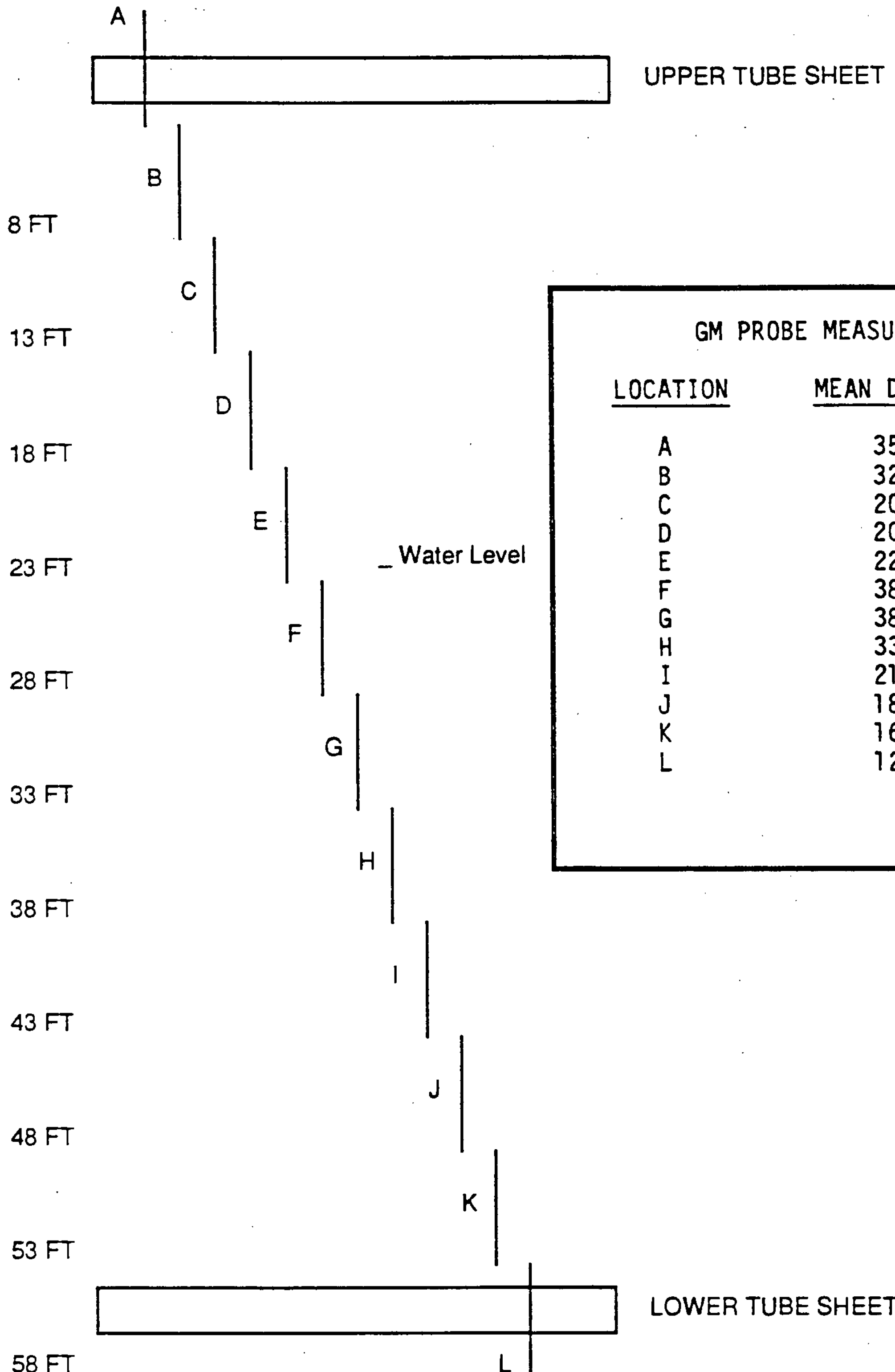


LOWER HEAD - PLAN VIEW

FIGURE 4

'A' OTSG TUBE BUNDLE

GM PROBE MEASUREMENT LOCATIONS



GM PROBE MEASUREMENT DATA

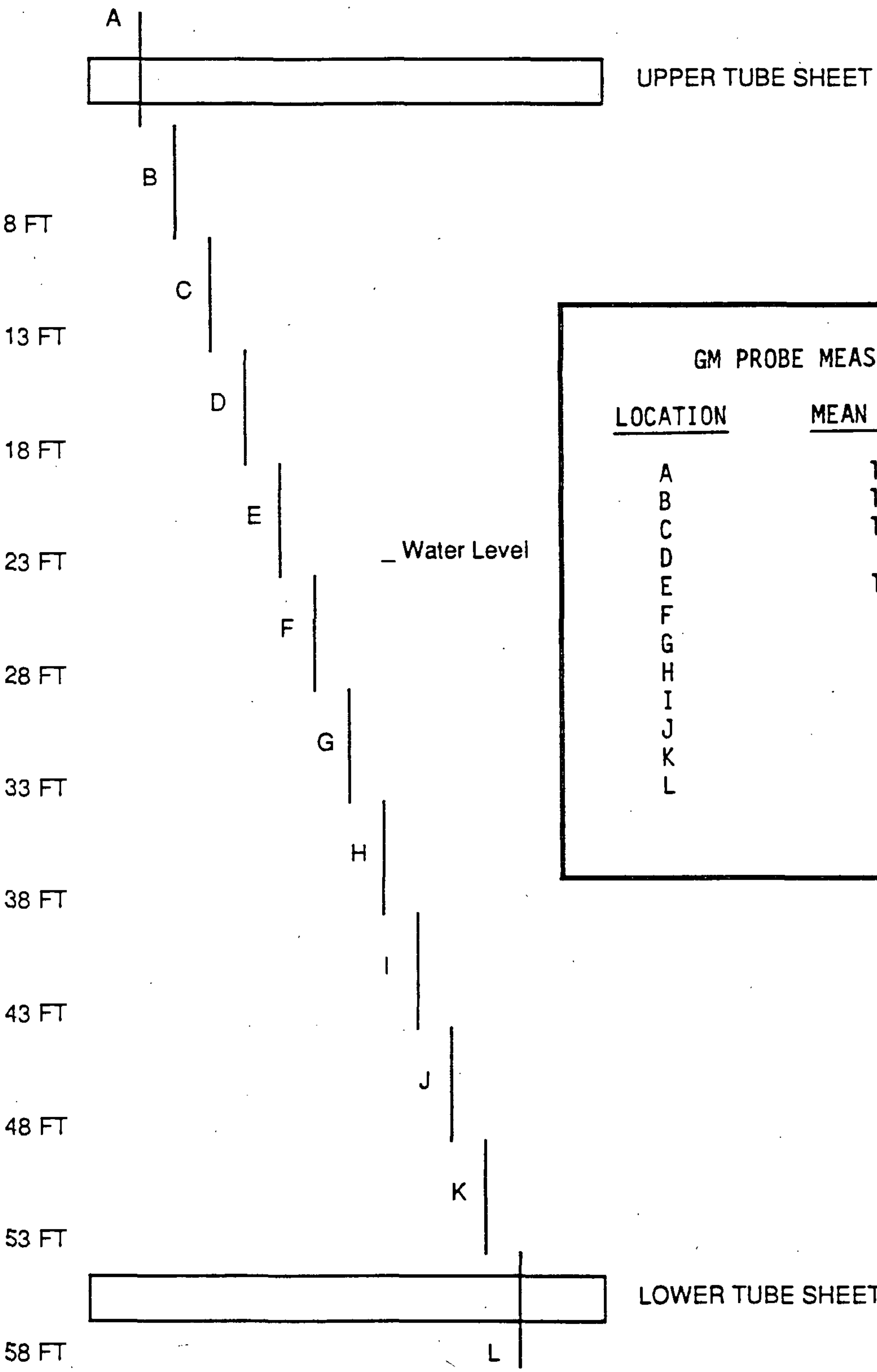
LOCATION	MEAN DOSE RATE (mR/hr)
A	350 ± 59
B	320 ± 37
C	207 ± 26
D	204 ± 25
E	228 ± 24
F	387 ± 19
G	384 ± 22
H	334 ± 23
I	215 ± 19
J	187 ± 17
K	164 ± 13
L	123 ± 21

NOTE: Numbers in feet are the distances below the top surface of the upper tube sheet.

FIGURE 5

'B' OTSG TUBE BUNDLE

GM PROBE MEASUREMENT LOCATIONS



GM PROBE MEASUREMENT DATA

LOCATION	MEAN DOSE RATE (mR/hr)
A	1290 ± 295
B	1270 ± 244
C	1213 ± 247
D	679 ± 150
E	1311 ± 288
F	771 ± 124
G	606 ± 50
H	540 ± 46
I	354 ± 27
J	352 ± 25
K	359 ± 24
L	318 ± 28

NOTE: Numbers in feet are the distances below the top surface of the upper tube sheet.

FIGURE 6
GAMMA PROBE CONTAMINATION

GPU Nuclear

RADIOLOGICAL SURVEY ☐TMI-1 ■TMI-2 ☐OC

Survey Information	Instrument Data		Air Sample ●	File Code Number: RB 700 88J 25693
	Contamination Survey	Radiation Survey		Smearable Contamination
Location 347' B D-RING UNIT 2 R.B.	Inst N/T — Inst N/T	Inst R02-A — Inst N/T	Date N/A Time	
Reason Job coverage RWP#	S/N — S/N	S/N 684 — S/N	Activity	
Date 12-21-88 Time 13:25	Cal.Due — Cal.Due	Cal.Due 3-2-89 — Cal.Due	Sample #	
Tech. GEO. HOLTZ	Eff. — Eff.	B.C.F. 2 — B.C.F.	Note: Radiation dose rates in mr/hr are general area, contamination results are dpm/100cm² unless otherwise noted.	
Survey Request# N/A	Bkg — Bkg	Tech GEO. HOLTZ		
Reviewed by	Tech	Note: Contact Readings Circled ☐ → Smear Location		

Location	βγ	α	Comments
	N/T		

Remarks:

K. Norman DEC 22 1988