Nuclear

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June 7, 1991
C312-91-2045
C000-91-1353

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 is 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." GPU Nuclear letter 4410-88-L-0162 dated September 30, 1988, submitted the initial Post-Defueling Survey Reports (PDSRs).

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 transmits the post-defueling survey results for the Auxiliary and Fuel Handling Buildings.

The remaining PDSRs will be submitted as they are completed. A compilation of the individual PDSRs will form the basis for a final assessment of the quantity of residual SNM at TMI-2 for accountability purposes.

Sincerely,

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

EDS/dlb

A001

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

TMI-2 POST-DEFUELING SURVEY REPORT

FOR THE

AUXILIARY & FUEL HANDLING BUILDINGS

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SUMMARY

The estimate of record of the amount of uranium dioxide (UO_2) remaining in the Auxiliary and Fuel Handling Buildings (AFHB) is 11.5 kg with a range from 4.2 to 15.8 Kg, distributed approximately as follows:

Spent Fuel Pool - A (FH109)	3.8 kg UO_2
RC Bleed Holdup Tanks - B&C (AX020) . . .	3.5 kg UO_2
MU&P Demineralizer - 1A (AX114)	1.1 kg UO_2
MU&P Demineralizer - 1B (AX115)	0.1 kg UO_2
MU Suction Vavles (FH001)	0.5 kg UO_2
Seal Return Coolers (AX112)	0.3 kg UO_2
Makeup Tank (AX116)	0.3 kg UO_2
RC Bleed Holdup Tank - A (AX021)	0.3 kg UO_2
MU&P Valves (FH101)	0.3 kg UO_2
Makeup Pump - 1A (AX007)	0.2 kg UO_2
78 other AFHB Areas	1.1 kg UO_2
AFHB Total	11.5 kg UO_2

The above fuel distribution table shows that 63% of the UO_2 remaining in the AFHB is located in two (2) areas, the Spent Fuel Pool A (FH109) and the RC Bleed Holdup Tanks B&C (AX020). The majority of this 7.3 kg of UO_2 is in the form of silt/sediment which was deposited in its present location after flushing or washing fuel laden pipes or canisters. In determining the quantity of fuel present in each area, a system wide approach was taken. Based on this review, it was determined that approximately 67% of the residual fuel was located in two (2) systems, Makeup & Purification and Liquid Radwaste Disposal Systems, and the majority of the remainder of the 11.5 kg was contained in the Spent Fuel Pool A.

The AFHB is divided into one-hundred and thirty-one (131) individual areas covering five (5) elevations. Four (4) areas are

located on elevation 258'6" and forty-two (42) areas are located on elevation 280'6", see Figure 1. Forty-six (46) areas are located on elevation 305'0", see Figure 2. Twenty-eight (28) areas are located on elevation 328'0", see Figure 3. Eleven (11) areas are located on elevation 347'6" or above, see Figure 4.

Thirty-eight (38) individual areas in the AFHB were measured. Twenty-eight (28) areas were measured using Gamma Scintillation, four (4) areas were measured using sample analysis and the remaining six (6) areas using Gross Gamma Exposure Rate measurements. Another fifty (50) areas were analyzed for fuel (UO_2) based on their vulnerability during the accident and their service during the cleanup program at TMI-2. Forty-three (43) other areas were evaluated and were judged as not containing any fuel because they were not connected in any way to the process lines supporting the reactor coolant system, the spent resin storage system or the RB sump pumping system.

The estimate of record, 11.5 kg UO_2, represents approximately 1½% of the total fuel remaining at TMI-2. The AFHB is expected to remain static since most of the systems/areas have been drained and isolated, and the remaining active systems will not be subjected to the large flow rates necessary to transport fuel. Additional measurements in the AFHB are not considered to be justifiable based on the measurement results to date and the potentially very small quantity of fuel (UO_2) involved. After final draindown of the RCS, the AFHB will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE AUXILIARY AND FUEL HANDLING BUILDINGS

1.0 INTRODUCTION

This report presents the analysis of the amount of UO_2 remaining in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are shown in the floor plan elevation drawings included with this report as Figures 1, 2, 3 and 4. This report is one in a series of reports prepared to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 1). All statistical uncertainties are expressed as $\pm$ one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the AFHB and the part played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the measurement methods used to assay the residual fuel (UO_2) in each area in the AFHB containing process systems which support and/or interface with the primary reactor coolant system. The majority of areas were assayed using gamma spectroscopy featuring a shielded thallium activated sodium iodide (NaI) scintillation detector connected to a multichannel analyzer (MCA). Since the presence of Cerium-144(Pr-144) was not always identifiable, a minimum detectable level (MDL) calculation was prepared when appropriate.

Section 4, "Analysis", explains the methodology for arriving at the estimate of record of fuel (UO_2) in the AFHB using multiple measurements and discusses supporting data, assumptions made and calculations used.

Section 5, "Conclusion", presents the estimate of record, uncertainty and MDL values for the amount of SNM remaining in the AFHB and is based upon the available data.

2.0 BACKGROUND

The March 1979 TMI-2 accident resulted in significant damage to the reactor core and subsequent release of fuel particles and fission products into the Reactor Coolant System (RCS) and other closely interconnected systems. The damaged core consisted of loose fuel pellets, solidified fuel, structural metal components, loose rubble, and partial fuel assemblies. Therefore, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

The Auxiliary and Fuel Handling Buildings (AFHB) are divided into two parts that are separated by a common wall. One part, the Auxiliary Building, contains tanks, pumps, piping and other equipment for the processing and storage of water for the reactor and primary cooling system and for the treatment of radioactive wastes. The other part, the Fuel Handling Building, contains fuel handling and storage equipment and a limited number of tanks, filters, pumps and coolers, and other facilities. The general layout of the AFHB is shown in Figures 1, 2, 3 and 4. The interior of the AFHB was contaminated by radioactive material as a consequence of the accident and cleanup activities. Piping systems that interfaced with the RCS were also contaminated with radioactive water and/or core debris. Early estimates (Reference 2) reported that approximately 25 kg of UO_2 was transported to the AFHB during the accident sequence. In addition, certain defueling water processing activities were considered to have transported as much as an additional 15 kg of fuel from the RB into the AFHB for a total of 40 kg. Subsequent decontamination and water processing have reduced the AFHB residual fuel inventory to its current level of approximately 12 kg (UO_2).

The majority of the twenty-six (26) systems in the AFHB were either never contaminated or have been flushed and drained in preparation for post-defueling monitored storage (PDMS). Most of the remaining fuel (UO_2) is contained within two systems, the Makeup and Purification System and the Liquid Radwaste Disposal System, and is located in various components, tanks and pipes. This residual fuel is believed to be in the form of finely divided particles and sediment material with some minor amounts of fuel remaining as adherent films. The quantity of fuel (UO_2) is substantially less than the TMI-2 safe fuel mass limit (SFML) of 140 kg for fuel external to the Reactor Vessel (Reference 3); thus, within the AFHB, there is no potential for fuel accumulation which could result in a critical mass.

After draindown of the RCS and spent fuel pools, including the reactor vessel, the AFHB will be isolated from the Reactor Building and other miscellaneous buildings by administrative controls to ensure that containment isolation is maintained and primary system valves remain in the closed position except when authorized to be open to process individual batches of water on an as needed basis.

3.0 MEASUREMENT METHODS

Post-defueling SNM measurement at TMI-2 was a complex task. Several different measurement techniques were used. Technique selection for a particular area depended on the configuration of the area, structure or component to be assayed, physical access limitations, area radiation dose rates and the likely nature of the form and quantity of fuel deposits. In the AFHB, the measurement techniques were limited to Gamma Scintillation Counting for all except ten (10) areas. In the ten (10) areas where Gamma Scintillation was not used to measure fuel quantities, gross gamma exposure rates were measured in six (6) areas. In four (4) other areas fuel content was established using individual samples (fuel/resin) from each area.

Extrapolations were made of residual fuel in twenty-nine (29) areas that were based on measurements taken in other areas subjected to similar accident conditions. Generally the residual fuel remaining in those areas not specifically measured was assessed to be small, i.e., less than 300 grams, and insignificant in terms of the total fuel remaining at TMI-2. The results of every area measured are supported by a formal Engineering Calculation referenced in Table 1. Each Engineering Calculation generally contains a preapproved measurement plan including both test equipment setup and data that were collected during the measurement period, analysis of the data/measurement recorded and the results of the individual assay. In December 1986, the implementing procedure was formalized to assure that the measurements were performed in a proper, accurate, verifiable and approved manner and is described in References 4 and 5.

The following assay methods were utilized to measure thirty-eight (38) areas in the Auxiliary and Fuel Handling Buildings (AFHB).

3.1 Sodium Iodide (NaI) - Gamma Scintillation

The following twenty-two (22) areas were surveyed using a NaI (Tl) detector and gamma spectroscopy:

Area	Description
AX006	Makeup Pump (1B) Room
AX007	Makeup Pump (1A) Room
AX011	AB Sump Pump & Valve Room
AX012	AB Sump & Tank Room
AX015a/b	Cleanup Filter Rooms
AX018	Waste Transfer Pump Room
AX019	WDL Valve Room
AX026	Seal Injection Filter Room
AX112	Seal Return Cooler Room
AX116	Makeup Tank Room
AX117	MU&P Filter Room
AX501	RB Spray Pump - 1A
AX502	RB Spray Pump - 1B
AX503	DHR Cooler & Pump - 1A
AX504	DHR Cooler & Pump - 1B
FH001	MU Suction Valve Room
FH003a/b	MU Discharge Valve Rooms
FH008	Neutralizer Tank Pump
FH009	Neutralizer Tank
FH101	MU&P Valve Room

Sodium iodide (NaI) detectors were used extensively in the AFHB measurement program. These detectors are thallium activated scintillation crystals; i.e. photon interactions in the NaI (Tl) crystals produced light pulses that were detected, changed to an electronic signal, and amplified by a photo multiplier tube. This device measured fuel by detection of the 2.19 MeV gamma ray of Cerium-144 (Ce-144). Ce-144(Pr-144) was selected as a tracer for reactor fuel and a Ce-144-to-fuel (UO_2) ratio was developed using sample data (Reference 6). The 2.19 MeV gamma was the highest energy gamma-ray peak that was observable thereby allowing the use of the low resolution NaI (Tl) System. This device was initially swamped by the gamma flux. The problem was solved by using the smallest detector that would successfully collect the 2.19 MeV gamma ray and the maximum practical amount of shielding. New York University optimized the design for the TMI-2 application. It consisted of a 1/2-inch diameter by 3/4-inch long crystal, a 1/2-inch diameter photomultiplier tube and 35 to 80 pounds of tungsten shielding that could be assembled in various configurations to enhance various source/detector counting geometries. The detector had good sensitivity. However, Ce-144 decay (284 d half-life) introduced a significant loss of sensitivity over the time frame of the Recovery Program requiring a shift to a High Purity Germanium Detector System. Additional details of these measurement techniques are described in References 7 and 8.

3.2 High-Purity Germanium (HPGe) Spectrometer

The following six (6) areas were surveyed using high-purity germanium detector systems:

Area	Description
AX004	Seal Injection Valve Room
AX014	RC Evaporator
AX020	RC Bleed Holdup Tanks - B & C
AX021	RC Bleed Holdup Tank - A

AX218 Concentrated Waste Tank

FH112 Annulus - Elevation 305'

The HPGe detector was used to locate and quantify the residual fuel (UO_2) in the above mentioned areas in the AFHB. AX004 and FH112 contain portions of the Makeup and Purification System while the other four (4) areas contain portions of the Radwaste Disposal System. The HPGe detector was housed in a 2-inch thick cylindrical lead shield. The detector was connected to a preamplifier, amplifier and multi-channel analyzer (MCA) as described in Reference 9. The MCA and input-output device were at a remote location; as much as 100 feet of signal cable was connected between the amplifier and analyzer. The HPGe detector measurements relied on Ce-144 and Europium-154 (Eu-154) as analogs for the residual fuel. The ratios Ce-144-and Eu-154-to-fuel (UO_2) were 152.5 and 42.6 μCi g^{-1}, respectively, as of 8-1-87 (Reference 6).

The detector systems were source checked periodically to ensure proper operation using standard Ce-144 and Eu-154 sources. After the calibration data had been collected, the standard sources were removed and the acquisition of data was performed.

3.3 Sample Analysis Determination

Four (4) areas (MU&P Demineralizer A (AX114), MU&P Demineralizer B (AX115), Deborating Demineralizer B (AX129), and Deborating Demineralizer A (AX130)) were evaluated based on individual samples of resin media analyzed by radiochemical laboratory. The samples taken from the MU&P Demineralizers were sent to Oak Ridge National Laboratory for evaluation and chemical analysis (Reference 10). The samples taken from the Deborating Demineralizers were analyzed on site, and the results are shown in Reference 11.

3.4 GROSS GAMMA

Gross gamma directional surveys were performed on piping and components when exposure rates and accessibility permitted. Six (6) areas (Makeup Pump - 1C (AX005), AB Sump Filters (AX024), Concentrated Liquid Waste Pump (AX124), Model Room (FH105), Monitor Tanks (FH106), and Spent Fuel Pool A (FH109)) were analyzed using gross gamma directional surveys. The measurement technique utilized a portable gamma survey instrument consisting of a directional probe and readout device. TMI-2 core debris contained fission and activation products which produced gross gamma-ray exposure rate values based on the radioisotopic distribution of sample data. Using the reactor core debris sample data, a correlation was developed between the gross gamma-ray output and the quantity of fuel present (Reference 12). Computer modeling was used to predict exposure rates for a given volume of fuel under the appropriate geometry for a particular location. Field measurements were made for specific locations and compared with the computer modeled output. Comparison was accomplished on a point-by-point basis. The results were generally good and the data was used to determine the quantity of fuel present in any area or component.

4.0 ANALYSIS

The quantity of residual fuel (UO_2) in the Auxiliary and Fuel Handling Buildings (AFHB) was determined by performing a series of non-destructive measurements, as described in Section 3.0, Measurement Methods. The entire AFHB was reviewed to identify those areas that could contain fuel deposits. The Rogovin Report (Reference 13) was used as the authoritative basis for the review. For the purpose of this Post-Defueling Survey Report (PDSR), the term Special Nuclear Material (SNM) was utilized to describe the residual material derived from the original UO_2 fuel and should not be construed as the amount of contained U-235. The final estimate of record of residual SNM was performed employing gamma detection measurements and engineering analyses. Engineering analyses were performed in areas where the total quantity was believed to be insignificant and where accident flowpaths were established.

An insignificant quantity of residual fuel was defined as about 7 kg of UO_2 or 5% of the maximum safe fuel mass for the AFHB (Reference 3). All assessment measurements were performed using direct gamma readings or counting techniques and are supported by formal engineering calculations containing relevant data and data reduction. The result of the assessment was an "estimate of record" of residual fuel (UO_2) within a determined range for the entire AFHB. The method used to construct the overall uncertainty is discussed in Appendix F.

Two (2) major plant systems contained 67% of the residual fuel in the AFHB. The two (2) systems are the Makeup and Purification (MU&P) and the Liquid Radwaste Disposal System (WDL). Other supporting systems contained quantities of fuel as listed in Table 7. The analysis for each plant system is discussed in the Appendices attached to this PDSR.

The eight (8) plant systems (identified in Tables 4, 5, 6 and 7) evaluated for fuel content occupy sixty-nine (69) individual areas in the AFHB and are listed in the Appendices for each system. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan, Reference 1.

Another forty-three (43) areas, shown in Table 3, were judged as not requiring SNM assessment based upon authoritative analyses of the TMI-2 accident, Reference 13. These areas contain non-water processing equipment such as electrical switchgear, unit sub-stations, HVAC blowers and ducting, or no equipment at all, and served primarily as an access corridor to service other areas. There was no means to transport fuel into these areas, e.g., no piping connected to any of the processing systems that support operation of reactor coolant system.

The final nineteen (19) areas contain no residual fuel and are discussed in detail in Appendix E. The nineteen (19) areas were assessed to be free of fuel because they were completely isolated from all fuel or had been contaminated and have since been decontaminated by repeated cleaning/flushing.

The MU&P System occupies portions of eighteen (18) areas in the AFHB which are illustrated in Figures 5 and 6. Fifteen (15) of the areas were measured for residual fuel and three (3) unmeasured areas were analyzed for fuel based on their operational history during the TMI-2 accident and their service during the recovery phase. The three (3) unmeasured areas (FH002, FH004, and FH014) were assigned a residual fuel content of 0.160 kg of UO_2 which is probably a high estimate of their actual content. The entire MU&P System was determined to contain 2.81 Kg UO_2 ± 27% plus 0.60 Kg UO_2 (MDL value). This results in a range of values from 2.05 Kg to 4.17 Kg UO_2. Details of this residual fuel appraisal are discussed in

Appendix A of this PDSR.

The WDL System occupies portions of twenty-nine (29) areas in the AFHB which are illustrated in Figures 7, 8, 9, 10 and 11 except for AX101 which is not connected to any of the processing pathways. Twenty (20) of the areas were measured for residual fuel and nine (9) unmeasured areas were analyzed for fuel based on their exposure during the TMI-2 accident and their service during the cleanup program. The nine (9) unmeasured areas (AX013, AX016, AX017, AX101, AX102, AX128, AX131, AX134 and FH012) were determined to contain 0.107 kg of UO_2 which is considered a conservative value. The entire WDL System was determined to contain 4.13 Kg UO_2 ± 71% plus 0.11 Kg UO_2 (MDL value). This results in a range of values from 1.20 Kg to 7.17 Kg UO_2. Details of this residual SNM appraisal are discussed in Appendix B of this PDSR.

The WDS System occupies portions of seven (7) areas in the AFHB which are illustrated in Figures 12 and 13. Two of the areas, AX124 and AX218, were measured for residual fuel and five (5) unmeasured areas were analyzed for fuel based on their exposure during the TMI-2 accident and their service during the recovery phase. The five (5) unmeasured areas (AX008, AX009 AX010, FH010 and FH011) were estimated to contain 0.004 kg of UO_2 which is considered a conservative estimate of record of their SNM content. The entire WDS System was determined to contain approximately 0.01 kg which has been rounded to 0.0 kg of UO_2 for purposes of this report. This sum is less than 0.1% of the total UO_2 remaining at TMI-2. Details of this residual SNM appraisal are discussed in Appendix C of this PDSR.

Appendix D addresses a group of seventeen (17) areas where fuel deposits are probably located. One area, the Spent Fuel Pool A (FH109), underwent a survey in the flooded condition and was appraised to contain 3.8 kg of UO_2. This represents the largest single deposit in the AFHB. The balance of the areas evaluated in Appendix D contained an insignificant quantity of UO_2. Details of the residual SNM appraised for each area are discussed in Appendix D of this PDSR.

Appendix E addresses a group of nineteen (19) areas where fuel deposits could have been located if the function of each area had been misused. No areas were formally measured and Appendix E presents the rationale supporting a determination that each area does not contain fuel (UO_2).

This survey resulted in a total of eighty-eight (88) areas being measured or analyzed for fuel content in the AFHB. The individual appraisal for each area is shown in Tables 4, 5, 6 and 7 and Appendix E of this PDSR. Thirty-eight (38) areas were formally measured using techniques discussed in Section 3.0, Measurement Methods, while the remainder were appraised based on their envolvement during the accident and service during the TMI-2 Cleanup Program. Forty-three (43) areas, listed in Table 3, were evaluated and judged not to contain fuel because they were not connected in any way to the process lines supporting the reactor coolant system.

Measurement of those areas in the AFHB that were not measured during this survey would result in additional unwarranted occupational radiation exposure to measurement teams and RadCon technicians. Review of the current analytical results indicates that measurements would not be expected to change the estimate of record by more than one kilogram and would not improve the accuracy or precision of the existing estimate of record. Additionally, the relative amount of residual fuel (UO_2) in the AFHB, when viewed in the context of the projected total inventory of UO_2, does not warrant additional radiation exposures in the AFHB to attempt to improve the present residual fuel values.

Approximately 10% of the process piping was assumed to be embedded in concrete walls and floors and was not included in the areas of the AFHB for which fuel was inferred. Each summary table for each system contains a contribution for embedded piping; the total contribution is 0.21 kg of UO_2.

The term "uncertainty" in this Post-Defueling Survey Report is used to represent the error of each measured "estimate of record" and is taken to be one sigma. The uncertainty was determined by the project engineer for each individual measurement and associated engineering calculation. The overall uncertainty value was derived by taking into consideration errors in mathematical modeling, measurement equipment, counting, standard values (fissionable material), randomness, and uncertainty in background levels. A range of fuel (UO_2) values was established using the measured and estimated data, uncertainty values and MDL values in accordance with the method outlined in Appendix F.

This approach resulted in an overall AFHB estimate of record of 11.5 kg UO_2 with an overall range of values from 4.22 Kg to 15.78 Kg UO_2. This estimate of record was calculated by summing the operating system totals shown in Table 2 and discussed in detail in Appendix F. All residual fuel values determined to be less than 0.005 Kg of UO_2 have been rounded off to 0.00 Kg for purposes of this report. Tables 4, 5, 6 and 7 show the rounded-off values.

5.0 Conclusion

The estimate of record of the quantity of fuel (UO_2) remaining in the AFHB is 11.5 kilograms. Based on the data shown in Table 2 and the data reduction methods outlined in Appendix F, the range of the estimate of record extends from 4.22 Kg to 15.78 Kg UO_2. The MDL portion of the upper range value is 0.71 Kg UO_2.

This estimate of record is derived from the measured and analyzed data summarized in Table 2. The data shows that most of the remaining fuel is contained in two systems (the Makeup and Purification System and the Liquid Radwaste Disposal System) and the Spent Fuel Pool A. The amount and location of the residual fuel in the AFHB is expected to remain static because many of the systems will be drained and isolated, and the water movement through the remaining systems will be limited. Consequently, all of the fuel that might move due to significant flow rates has already been relocated to a tank, filter or other low velocity location. Additional measurements in the AFHB are not considered to be warranted based on ALARA considerations due to the small quantity of fuel (UO_2) estimated to be located residual to the AFHB. After final draindown of the RCS, the AFHB will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

The goal of the TMI-2 defueling program, to remove more than 99% of the original core inventory of approximately 94,000 kg, was achieved. In that context, the 11.5 Kg quantity of UO_2 remaining in the AFHB is approximately 1½% of the total residual UO_2 inventory for TMI-2.

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TABLE 1

ASSAY METHODS UTILIZED TO MEASURE AREAS IN THE AFHB

AREA	DESCRIPTION	ASSAY METHOD	CALCULATION NUMBER
AX004	Seal Injection Valve Room	HPGe w/Ce-144/Eu-154	4800-3211-89-103
AX005	Makeup Pump - 1C	Gross Gamma E530	4249-3232-91-001
AX006	Makeup Pump - 1B	NaI Gam Spec - Ce-144	4550-3211-88-009
AX007	Makeup Pump - 1A	NaI Gam Spec - Ce-144	4550-3211-87-027
AX011	AB Sump Tank Pumps & Valves	NaI Gam Spec - Ce-144	4200-3232-90-048
AX012	AB Sump, Pumps & Tank	NaI Gam Spec - Ce-144	4200-3232-90-048
AX014	RC Evaporator (WDL-Z-1)	HPGe	Reference 17
AX015a	Cleanup Filters WDL-F-6B/9B	NaI Gam Spec - Ce-144	4249-3232-90-070
AX015b	Cleanup Filters WDL-F-6A/9A	NaI Gam Spec - Ce-144	4249-3232-90-070
AX018	Waste Transfer Pumps	NaI Gam Spec - Ce-144	4249-3232-90-071
AX019	WDL Valve Room	NaI Gam Spec - Ce-144	4200-3232-90-049
AX020	RC Bleed Holdup Tanks - B & C	HPGe w/Eu-154	4800-3232-89-099
AX021	RC Bleed Holdup Tank - A	HPGe w/Eu-154	4800-3232-89-077
AX024	AB Sump Filters	Gross Gamma E530	4200-3232-90-064
AX026	Seal Injection Filters	NaI Gam Spec - Ce-144	4550-3211-87-017
AX112	Seal Return Coolers	NaI Gam Spec - Ce-144	4550-3211-87-025
AX114	MU&P Demineralizer - 1A	Sampled	4249-3211-90-069
AX115	MU&P Demineralizer - 1B	Sampled	4249-3211-90-069
AX116	Makeup Tank (MU-T-1)	NaI Gam Spec - Ce-144	4550-3211-87-038
AX117	MU&P Filters (F-2A/B & 5A/B)	NaI Gam Spec - Ce-144	4200-3233-90-053
AX124	Conc. Liquid Waste Pump	Gross Gamma HP220	4200-3233-90-050
AX129	Deborating Demineralizer - 1B	Sampled & used Ce-144	4550-3232-87-016
AX130	Deborating Demineralizer - 1A	Sampled & used Ce-144	4550-3232-87-016
AX218	Conc. Waste Tank (WDS-T-2)	HPGe w/Ce-144/Eu-154	4240-3233-90-061
AX501	RB Spray Pump - 1A	NaI Gam Spec - Ce-144	4249-3232-90-072
AX502	RB Spray Pump - 1B	NaI Gam Spec - Ce-144	4249-3232-90-072
AX503	DHR Cooler & Pump - 1A	NaI Gam Spec - Ce-144	4249-3232-90-072
AX504	DHR Cooler & Pump - 1B	NaI Gam Spec - Ce-144	4249-3232-90-072
FH001	MU Suction Valves	NaI Gam Spec - Ce-144	4200-3233-90-056
FH003a	MU Discharge Valves	NaI Gam Spec - Ce-144	4200-3233-90-057
FH003b	MU Discharge Valves	NaI Gam Spec - Ce-144	4200-3233-90-057
FH008	Neutralizer Tank Pump	NaI Gam Spec - Ce-144	4200-3233-90-054
FH009	Neutralizer Tanks (T-8A/8B)	NaI Gam Spec - Ce-144	4200-3233-90-054
FH101	MU&P Valves	NaI Gam Spec - Ce-144	4200-3233-90-055
FH105	Model Room	Gross Gamma HP220A	4200-3551-90-051
FH106	Monitor Tanks (SDS-T-1A/1B)	Gross Gamma HP220A	4200-3551-90-051
FH109	Spent Fuel Pool A	Gross Gamma LND716	4800-4420-89-095
FH112	Annulus - Elevation 305'	HPGe w/Ce-144/Eu-154	4240-3211-90-060

TABLE 2

SUMMARY OF AUXILIARY AND FUEL HANDLING BUILDING SNM INVENTORY

SYSTEM	DETERMINED FUEL (UO_2) QUANTITY			REFERENCE
	Measured Value	Uncertainty on Measured Values	MDL Value	
Makeup & Purification	2.81 kg	+27%, –27%	0.60 kg	Appendix A
Liquid Radwaste Disposal	4.13 kg	+71%, –71%	0.11 kg	Appendix B
Solids Radwaste Disposal	0.01 kg	+57%, –57%	–	Appendix C
Balance of Systems	3.80 kg	+34%, –92%	–	Appendix D
Other areas	0.00* kg	–	–	Appendix E
Subtotal	10.75 kg		0.71 kg	

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Total UO_2 Range: Low Value: 4.22 kg, High Value: 15.78 kg

Estimate of Record: 11.46 kg UO_2

* Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 kg UO_2.

TABLE 3

AREAS NOT REQUIRING SNM ASSESSMENT

DESIGNATION	NAME	EXPLANATION
AX002	Access Corridor	No water piping in area
AX003	Access Area	No water piping in area
AX022	North Stairwell	No water piping in area
AX023	Elevator Shaft	No water piping in area
AX027	South Stairwell	No water piping in area
AX103	MCC 2-11 EB	No water piping in area
AX104	MCC 2-21 EB	No water piping in area
AX105	Substation 2-11 E	No water piping in area
AX106	Substation 2-21 E	No water piping in area
AX107	MCC 2-11 EA	No water piping in area
AX108	MCC 2-21 EA	No water piping in area
AX121	Elevator Shaft	No water piping in area
AX122	North Stairwell	No water piping in area
AX123	Access Area	No water piping in area
AX133	South Stairwell	No water piping in area
AX135	Radwaste Disposal Control Panel	Same area as AX101 (see Appendix B)
AX201	North Stairwell	No water piping in area
AX202	Elevator Shaft	No water piping in area
AX203	4160 Switchgear 2-1 E	No water piping in area
AX204	4160 Switchgear 2-2 E	No water piping in area
AX205	RB Purge Air Supply	System design prevents fuel transport
AX206	RB Purge Exhaust - B	System design prevents fuel transport
AX207	RB Purge Exhaust - A	System design prevents fuel transport
AX208	AB Exhaust Unit B	System design prevents fuel transport
AX209	AB Exhaust Unit A	System design prevents fuel transport
AX210	FHB Exhaust Unit B	System design prevents fuel transport
AX211	FHB Exhaust Unit A	System design prevents fuel transport
AX213	Unit Substation	No water piping in area
AX215	FHB Supply Unit	System design prevents fuel transport
AX216	AB Supply Unit	System design prevents fuel transport
AX217	Access Area	No water piping in area
AX222	South Stairwell	No water piping in area
AX223	Air Handling Units	System design prevents fuel transport
AX301	Elevator Shaft	No water piping in area
AX302	North Stairwell	No water piping in area
AX303	Elevator and Stairwell Access	No water piping in area
AX401	Roof	No water piping in area
AX403	Damper Room	System design prevents fuel transport
FH013	Oil Drum Storage	No water piping in area
FH102	East Corridor	No water piping in area
FH104	West Corridor	No water piping in area
FH201	East Corridor	No water piping in area
FH202	West Corridor	No water piping in area

TABLE 4

SUMMARY OF MAKEUP AND PURIFICATION SYSTEM SNM INVENTORY

AREA	DESCRIPTION	DETERMINED FUEL QUANTITY (KG)	UNCERTAINTY	CALCULATION/ REFERENCE
AX004	Seal Injection Valve Room	0.03	± 58%	4800-3211-89-103, Rev. 0
AX005	Makeup Pump - 1C	0.00	–	4249-3232-91-001, Rev. 0
AX006	Makeup Pump - 1B	0.07	MDL	4550-3211-88-009, Rev. 0
AX007	Makeup Pump - 1A	0.23	MDL	4550-3211-87-027, Rev. 1
AX026	Seal Injection Filters	0.00	MDL	4550-3211-87-017, Rev. 1
AX112	Seal Return Coolers	0.30	MDL	4550-3211-87-025, Rev. 1
AX114	MU&P Demineralizer - 1A	1.06	± 44%	4249-3211-90-069, Rev. 0
AX115	MU&P Demineralizer - 1B	0.13	± 25%	4249-3211-90-069, Rev. 0
AX116	Makeup Tank	0.31	± 95%	4550-3211-87-038, Rev. 1
AX117	MU&P Filters	0.06	± 68%	4200-3233-90-053, Rev. 0
FH001	MU Suction Valves	0.46	± 85%	4200-3233-90-056, Rev. 0
FH003a	MU Discharge Valves	0.01	± 100%	4200-3233-90-057, Rev. 0
FH003b	MU Discharge Valves	0.10	± 100%	4200-3233-90-057, Rev. 0
FH002	Access Corridor			
FH004	W Valve Room	0.16*	± 50%	PDSR Appendix A
FH014	Annulus - Elevation 281'			
FH101	MU&P Valves	0.32	± 71%	4200-3233-90-055, Rev. 0
FH112	Annulus - Elevation 305'	Note 1		4200-3211-90-060, Rev. 0
Misc.	Embedding Piping	0.17	± 95%	PDSR Appendix A

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	Measured Values	Uncertainty on Measured Values	MDL Values
Estimate of Record	2.81 Kg	(+ 27%, – 27%)	0.60 Kg

*Fuel quantity was determined using measured data for areas with similar service.
NOTE 1 – Measured fuel quantity was included in the WDL System total.
NOTE 2 – Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO_2.
NOTE 3 – See Appendix F for details of data reduction.

TABL

SUMMARY OF LIQUID RADWASTE DISPOSAL SYSTEM SNM INVENTORY

AREAS	DESCRIPTION	DETERMINED FUEL QUANTITY (KG)	UNCERTAINTY	CALCULATION/ REFERENCE
AX011	AB Sump Tank Pumps & Valves	0.00	-	4200-3232-90-048, Rev. 1
AX012	AB Sump, Pumps & Tank	0.10	± 104%	4200-3232-90-048, Rev. 1
AX013	Evaporator Cond. Tanks & Pumps	0.00*	-	PDSR Appendix B
AX014	RC Evaporator	0.00	MDL	Reference 17
AX015a/b	Cleanup Filters	0.10	MDL	4249-3232-90-070, Rev. 0
AX016	Cleanup Demineralizer - 2A	0.00	MDL	4249-3232-90-070, Rev. 0
AX017	Cleanup Demineralizer - 2B	0.00	MDL	4249-3232-90-070, Rev. 0
AX018	Waste Transfer Pumps	0.00	MDL	4249-3232-90-071, Rev. 0
AX019	WDL Valve Room	0.01	MDL	4200-3232-90-049, Rev. 1
AX020	RC Bleed Holdup Tanks - B & C	3.50	± 83%	4800-3232-89-099, Rev. 0
AX021	RC Bleed Holdup Tank - A	0.31	± 46%	4800-3232-89-077, Rev. 0
AX024	AB Sump Filters	0.02	± 36%	4200-3232-90-064, Rev. 0
AX101	Radwaste Disp. Control Panel	0.00	-	PDSR Appendix B
AX102	RB Sump Pump Filters			PDSR Appendix B
AX131	Miscellaneous Waste Holdup Tank	0.10*	± 104%	PDSR Appendix B
AX134	Misc. Waste Holdup Tank Pump			PDSR Appendix B
AX128	Instrument & Valve Room	0.01*	± 95%	PDSR Appendix B
AX129	Deborating Demineralizer - 1B	0.00	-	4550-3232-87-016, Rev. 0
AX130	Deborating Demineralizer - 1A	0.00	-	4550-3232-87-016, Rev. 0
AX501	RB Spray Pump - 1A	0.01	± 76%	4249-3232-90-072, Rev. 0
AX502	RB Spray Pump - 1B	0.01	± 59%	4249-3232-90-072, Rev. 0
AX503	DHR Cooler & Pump - 1A	0.01	± 100%	4249-3232-90-072, Rev. 0
AX504	DHR Cooler & Pump - 1B	0.01	± 92%	4249-3232-90-072, Rev. 0
FH008	Neutralizer Tank Pump	0.00	-	4200-3233-90-054, Rev. 0
FH009	Neutralizer Tanks	0.00	-	4200-3233-90-054, Rev. 0
FH012	Neutralizer Tank Filters	0.00*	-	PDSR Appendix B
FH101	MU&P Valve Room	Note 1		4200-3233-90-055, Rev. 0
FH112	Annulus - Elevation 305'	0.01	± 55%	4240-3211-90-060, Rev. 0
Misc.	Embedding Piping	0.04	± 95%	PDSR Appendix B

* * * * * * * * * * * * * * * *

	Measured Values	Uncertainty on Measured Values	MDL Values
Estimate of Record	4.13 Kg	(+ 71%, - 71%)	0.11 Kg

*Fuel quantity was determined using measured data for areas with similar service.
NOTE 1 - Fuel quantity was included in MU&P System total.
NOTE 2 - Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO_2.
NOTE 3 - See Appendix F for details of data reduction.

TABLE 6

SUMMARY OF SOLIDS RADWASTE DISPOSAL SYSTEM SNM INVENTORY

AREA	DESCRIPTION	DETERMINED FUEL QUANTITY (KG)	UNCERTAINTY	CALCULATION/ REFERENCE
AX008	SRST (WDS-T-1B)	**		PDSR Appendix C
AX009	SRST (WDS-T-1A)	**		PDSR Appendix C
AX010	Spent Resin Transfer Pump	**		PDSR Appendix C
AX124	Conc. Liquid Waste Pump	0.00*	–	4200-3233-90-050, Rev. 0
AX218	Conc. Waste Tank (WDS-T-2)	0.01*	± 57%	4240-3233-90-061, Rev. 0
FH010	Reclaimed Boric Acid Tank	0.00	–	PDSR Appendix C
FH011	Reclaimed Boric Acid Pump	0.00	–	PDSR Appendix C
Misc.	Embedded Piping	0.00	–	PDSR Appendix C

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	Measured Values	Uncertainty on Measured Values	MDL Values
Estimate of Record	0.01 Kg	(+ 57%, – 57%)	0 Kg

*AX124 AND AX218 FUEL QUANTITY WAS DETERMINED BY ACTUAL MEASUREMENTS IN BOTH AREAS.
**SNM INVENTORY WAS INCLUDED IN THE DEMINERALIZER VALUES, SEE APPENDIX C.
NOTE: Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO_2.

TABLE 7

SUMMARY OF BALANCE OF SYSTEMS CONTAINING SNM INVENTORY

AREA	DESCRIPTION	PRIMARY SYSTEM	DETERMINED FUEL QUANTITY (KG)	UNCERTAINTY	CALCULATION/ REFERENCE
AX113	Waste Gas Analyzer Room	WDG	0.00	–	PDSR Appendix D
AX118	Spent Fuel Cooler Area	SFC	0.00	–	PDSR Appendix D
AX119	Spent Fuel Demin. K–1	SFC	0.00	–	PDSR Appendix D
AX120	Spent Fuel Filters F–1A/1B	SFC	0.00	–	PDSR Appendix D
AX125	Waste Gas Decay Tank – 1B	WDG	0.00	–	PDSR Appendix D
AX126	Waste Gas Filter F–1	WDG	0.00	–	PDSR Appendix D
AX127	Waste Gas Decay Tank – 1A	WDG	0.00	–	PDSR Appendix D
FH105	Model Room	SNS	0.00*	–	4200–3551–90–051
FH106	Monitor Tanks & Sample Sink	SDS	0.00*	–	4200–3551–90–051
FH109	Spent Fuel Pool A	SFC	3.80*	+34%, –92%	4800–4420–89–095
FH110	Spent Fuel Pool B	SDS	0.00	–	PDSR Appendix D
FH111	Fuel Cast Storage	SFC	0.00	–	PDSR Appendix D
FH205	Annulus – Elevation 328'	DWCS	0.00	–	PDSR Appendix D
FH302	SDS Operating Area	SDS	0.00	–	PDSR Appendix D
FH303	New Fuel Storage Area	DWCS	0.00	–	PDSR Appendix D
FH304	Annulus – Above 347'6"	Electrical	0.00	–	PDSR Appendix D
FH305	Spent Fuel Pool Access Area	DWCS	0.00	–	PDSR Appendix D

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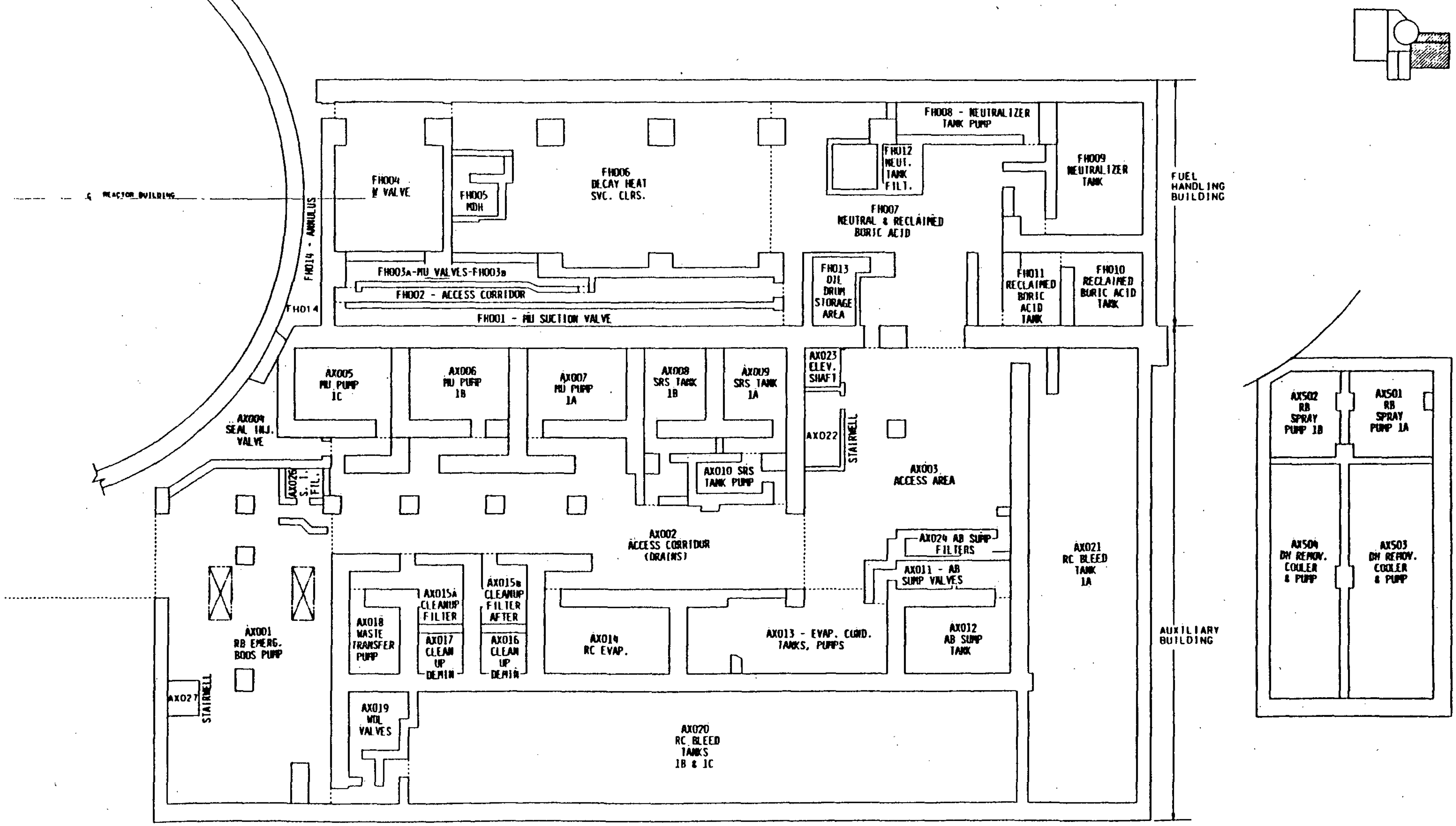
	Measured Values	Uncertainty on Measured Values	MDL Values
Estimate of Record	3.80 Kg	+34%, –92%	0 Kg

*Areas were measured for fuel using gross gamma directional surveys.
NOTE: Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO_2.

FIGL I

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 280'6" & 258'6"



FIGU… 2

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 305'

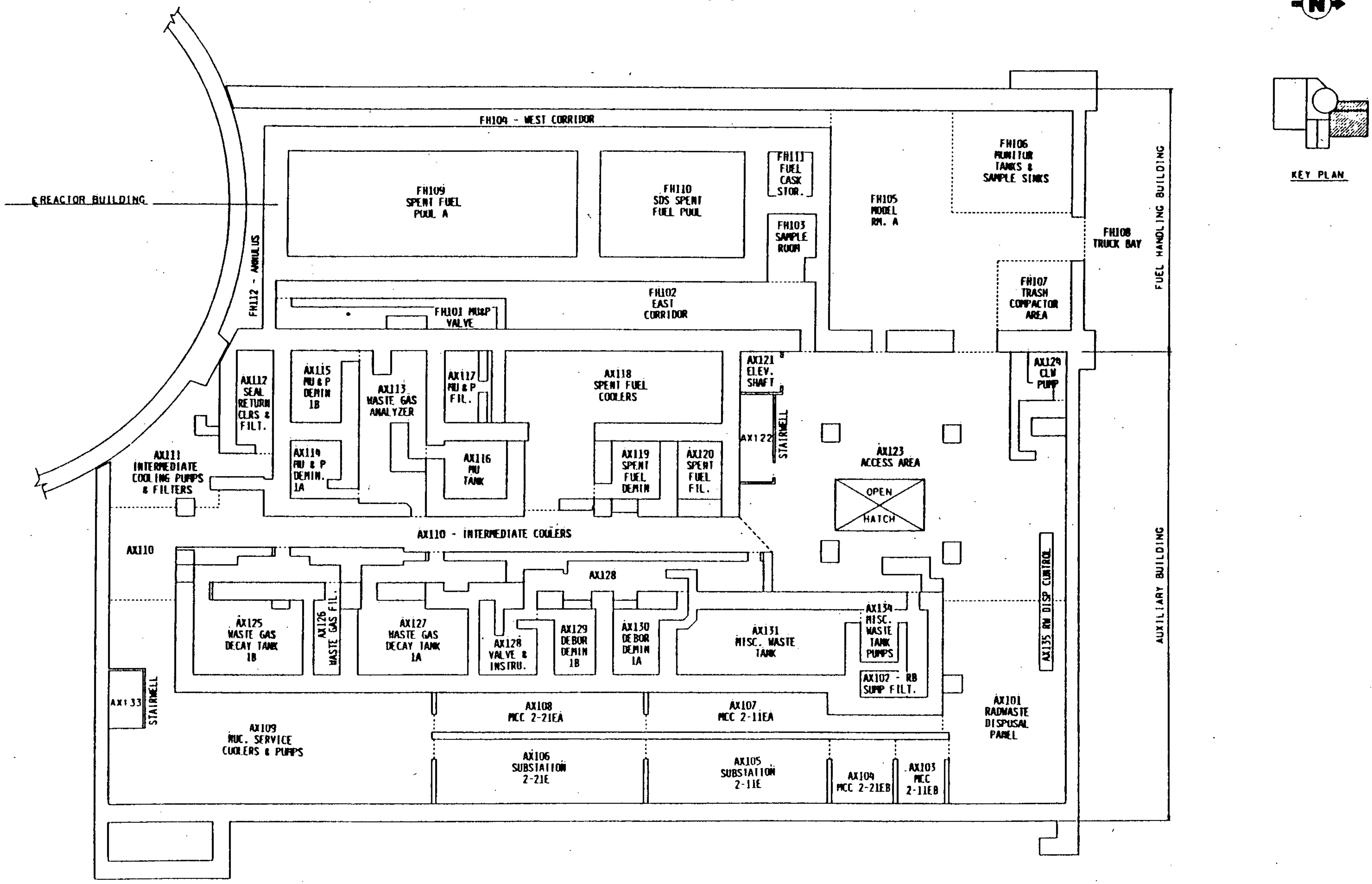


FIGURE 3

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 328"

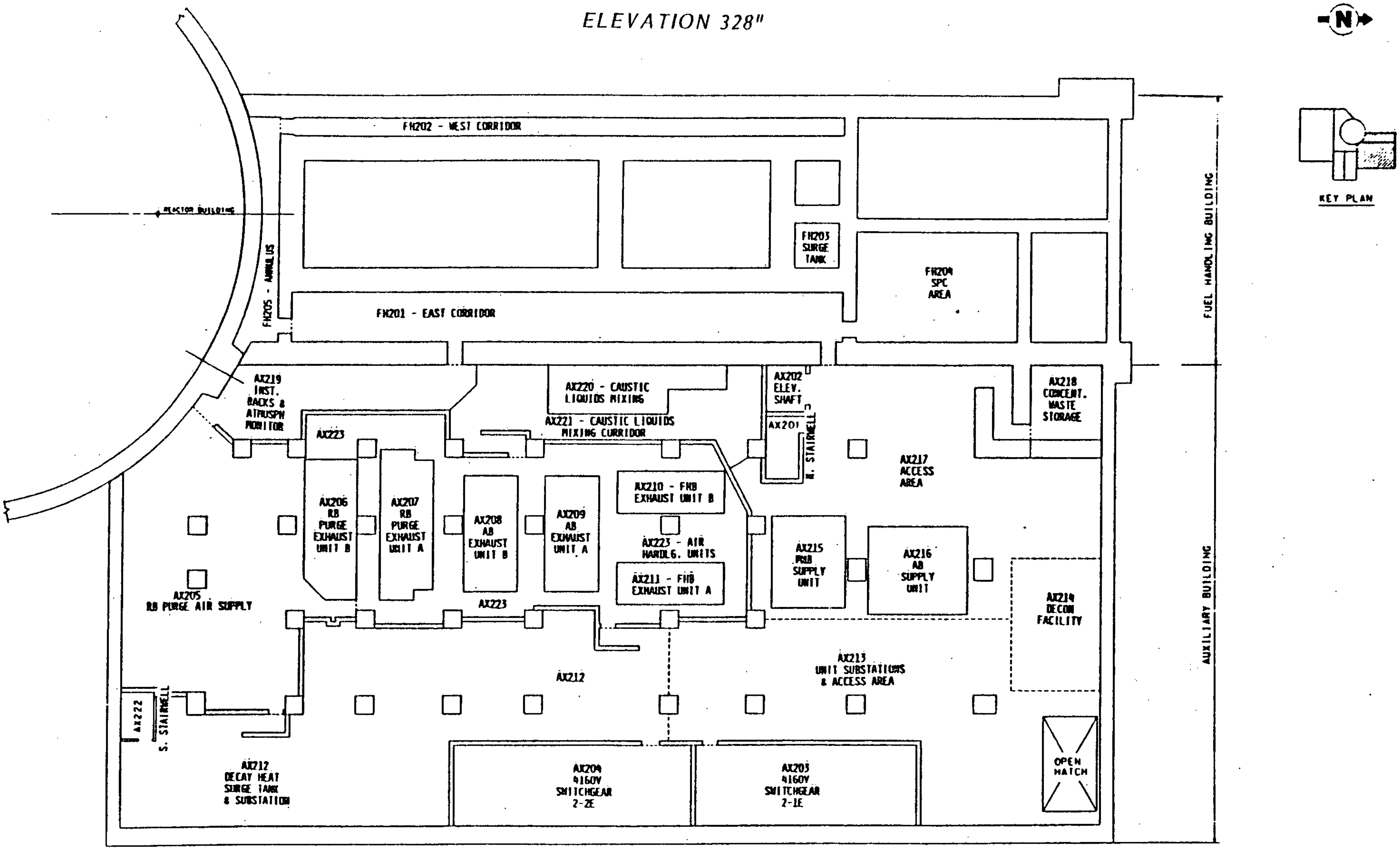


FIGURE 4

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 347'6"

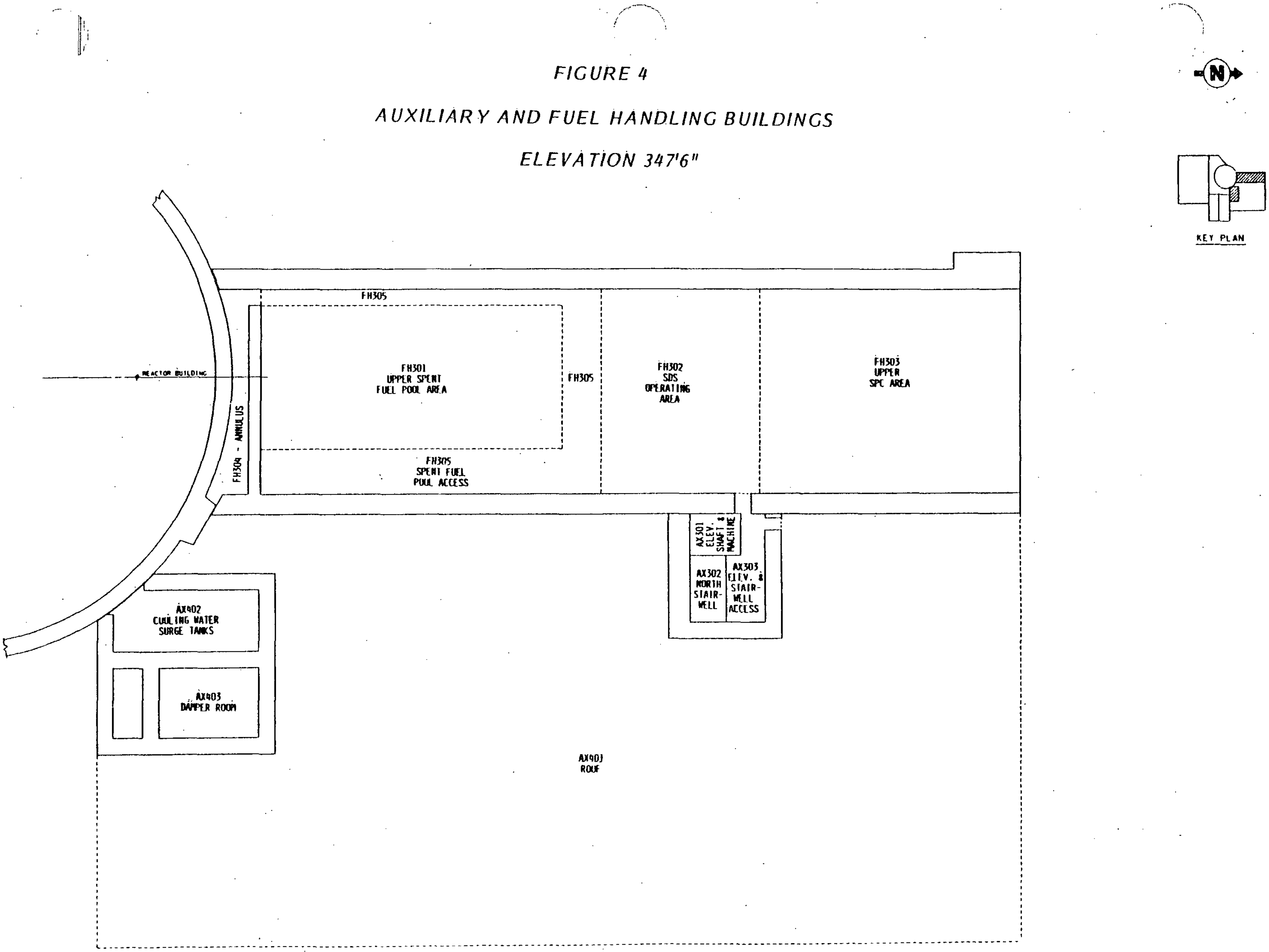


FIGURE 5
MAKEUP AND PURIFICATION SYSTEM
FLOWPATH: FIRST SECTION, 305'-0" EL AFHB

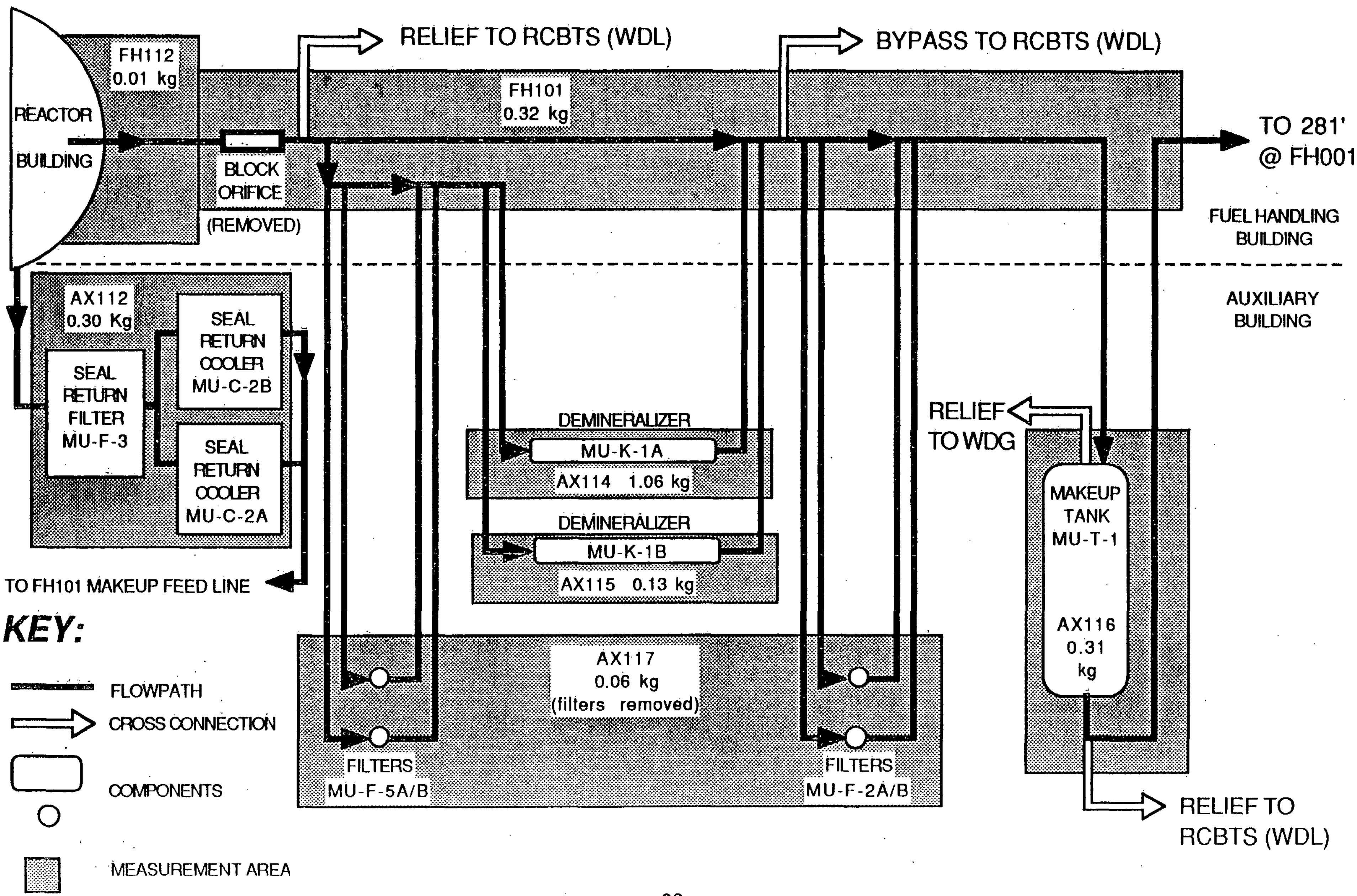


FIGURE 6
MAKEUP AND PURIFICATION SYSTEM
FLOWPATH: SECOND SECTION, 281'-6" EL AFHB

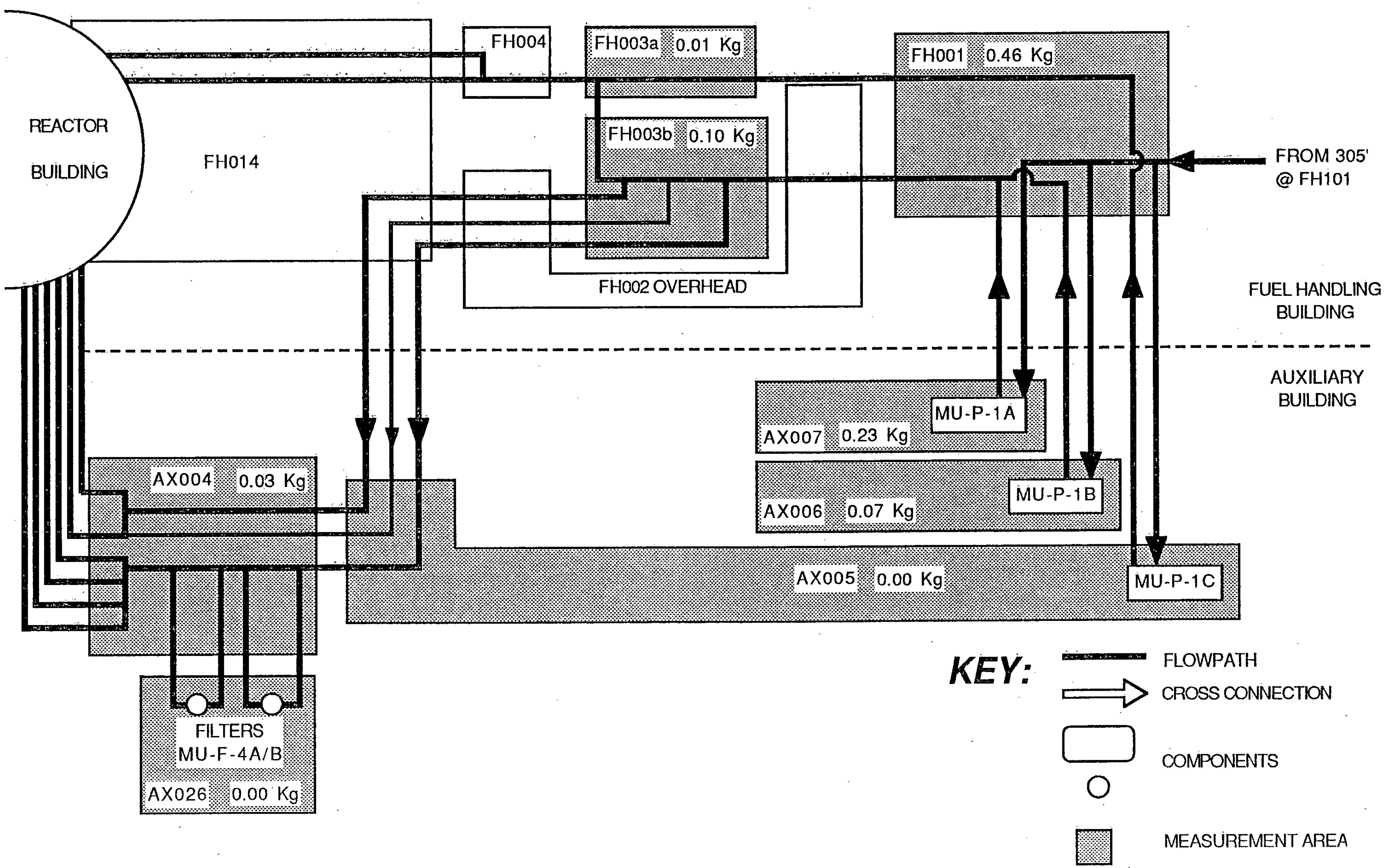


FIGURE 7
LIQUID RADWASTE DISPOSAL SYSTEM
AUXILIARY BUILDING - ELEVATION 258' 6"

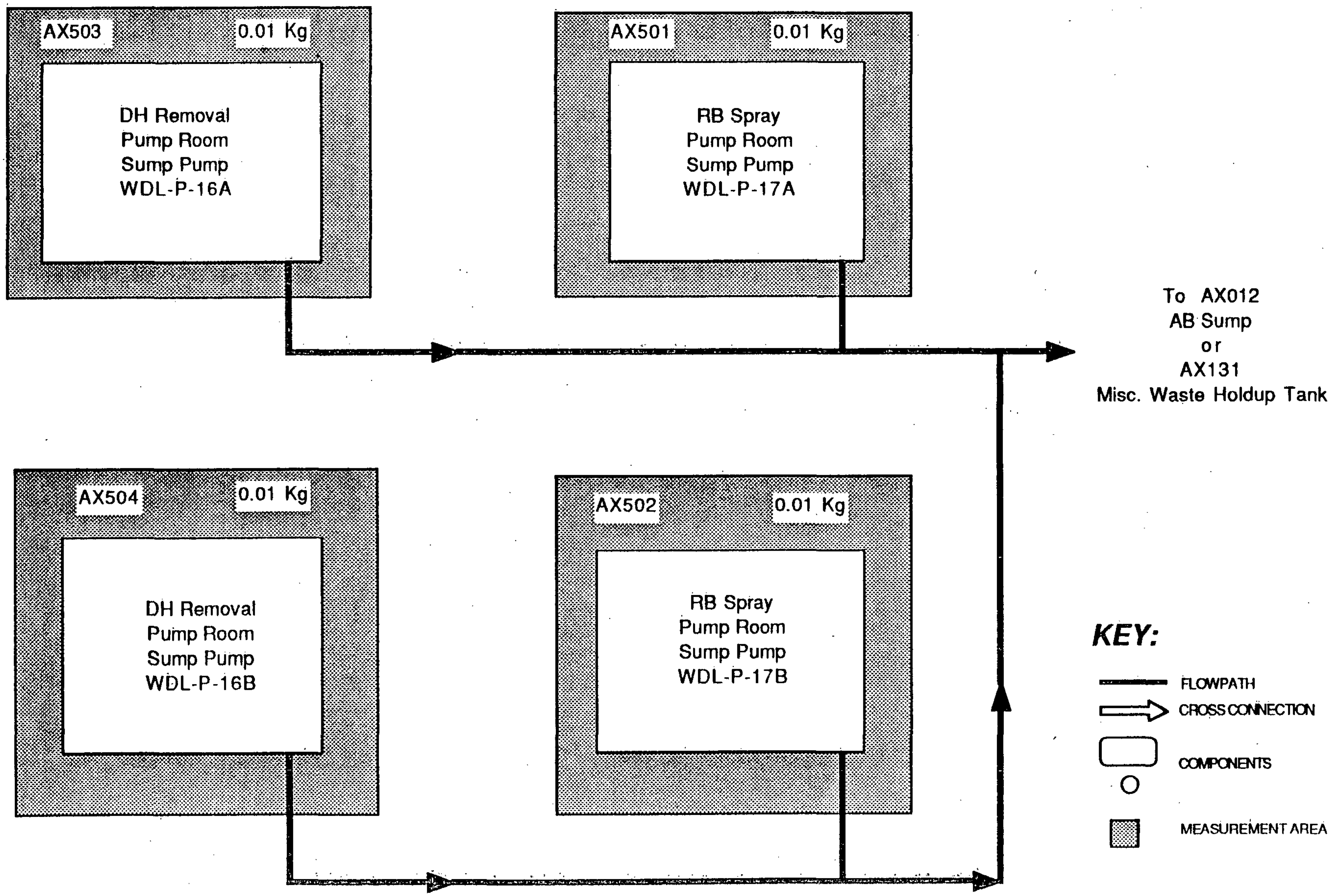


FIGURE 8
LIQUID RADWASTE DISPOSAL SYSTEM
FUEL HANDLING BUILDING - ELEVATION 280' 6"

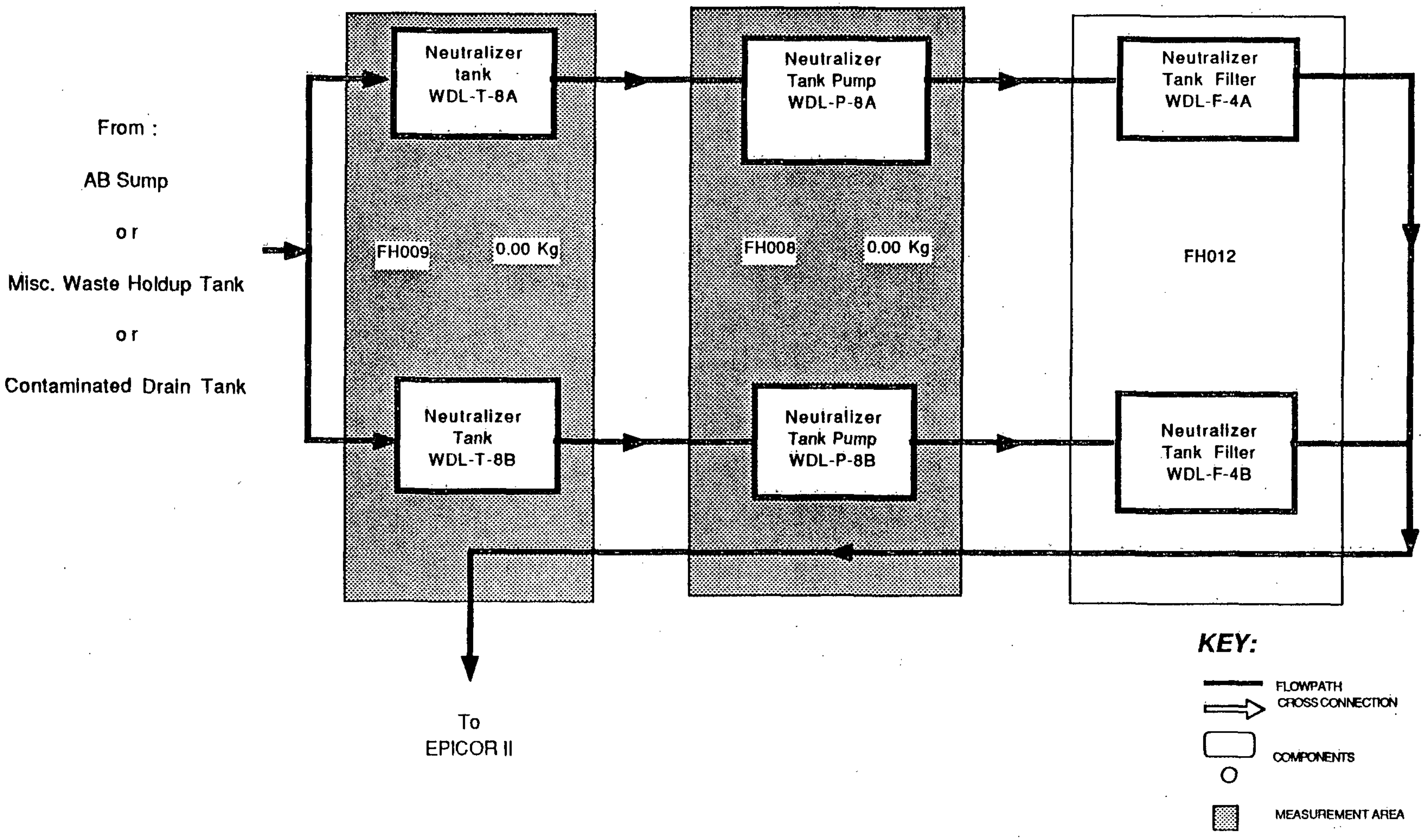
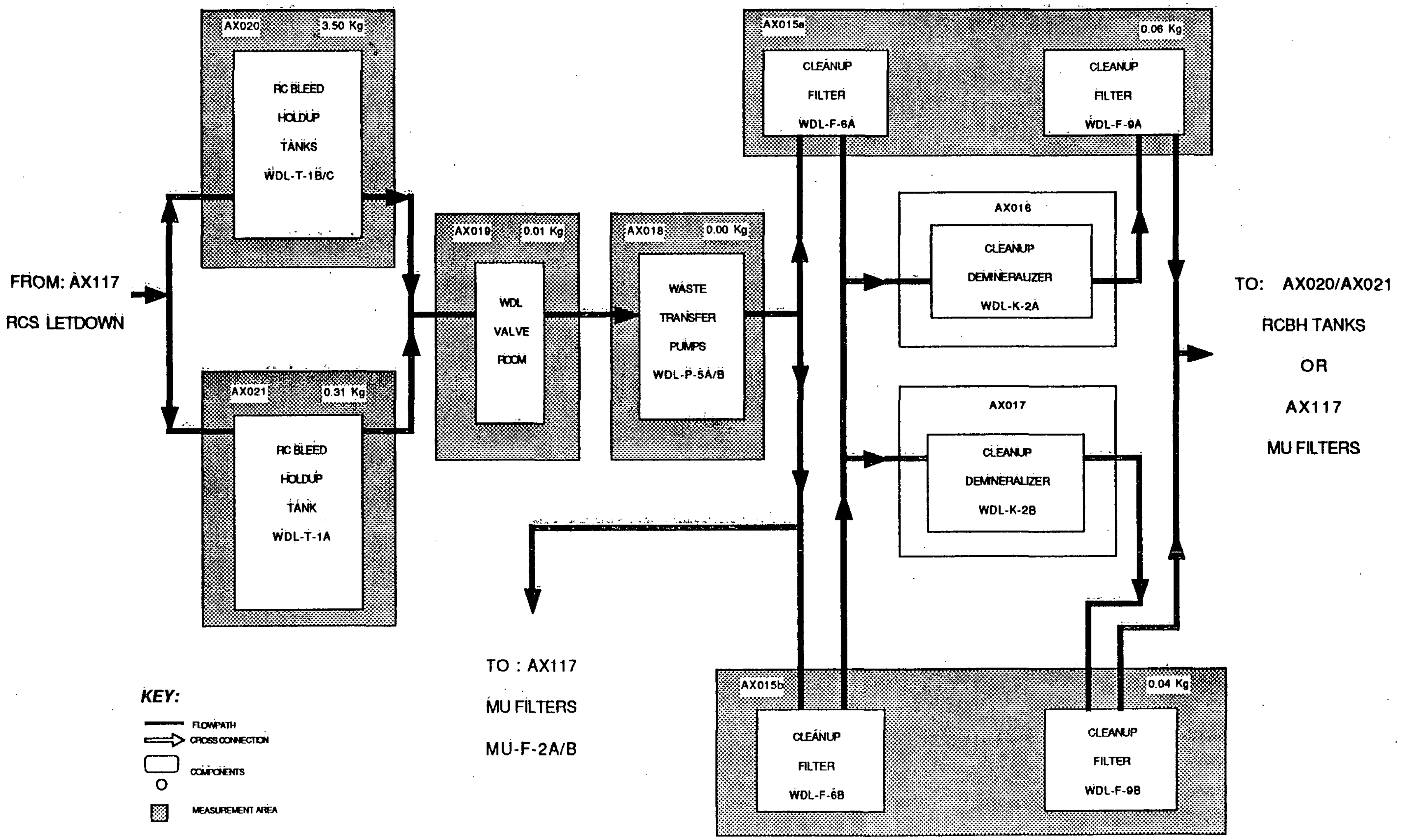


FIGURE 9
LIQUID RADWASTE DISPOSAL SYSTEM - CLEANUP SUBSYSTEM
AUXILIARY BUILDING - ELEVATION 280' 6"



F... JRE 10
LIQUID RADWASTE DISPOSAL SYSTEM - RC EVAPORATOR AND AB SUMP SUBSYSTEMS
AUXILIARY BUILDING - ELEVATION 280' 6"

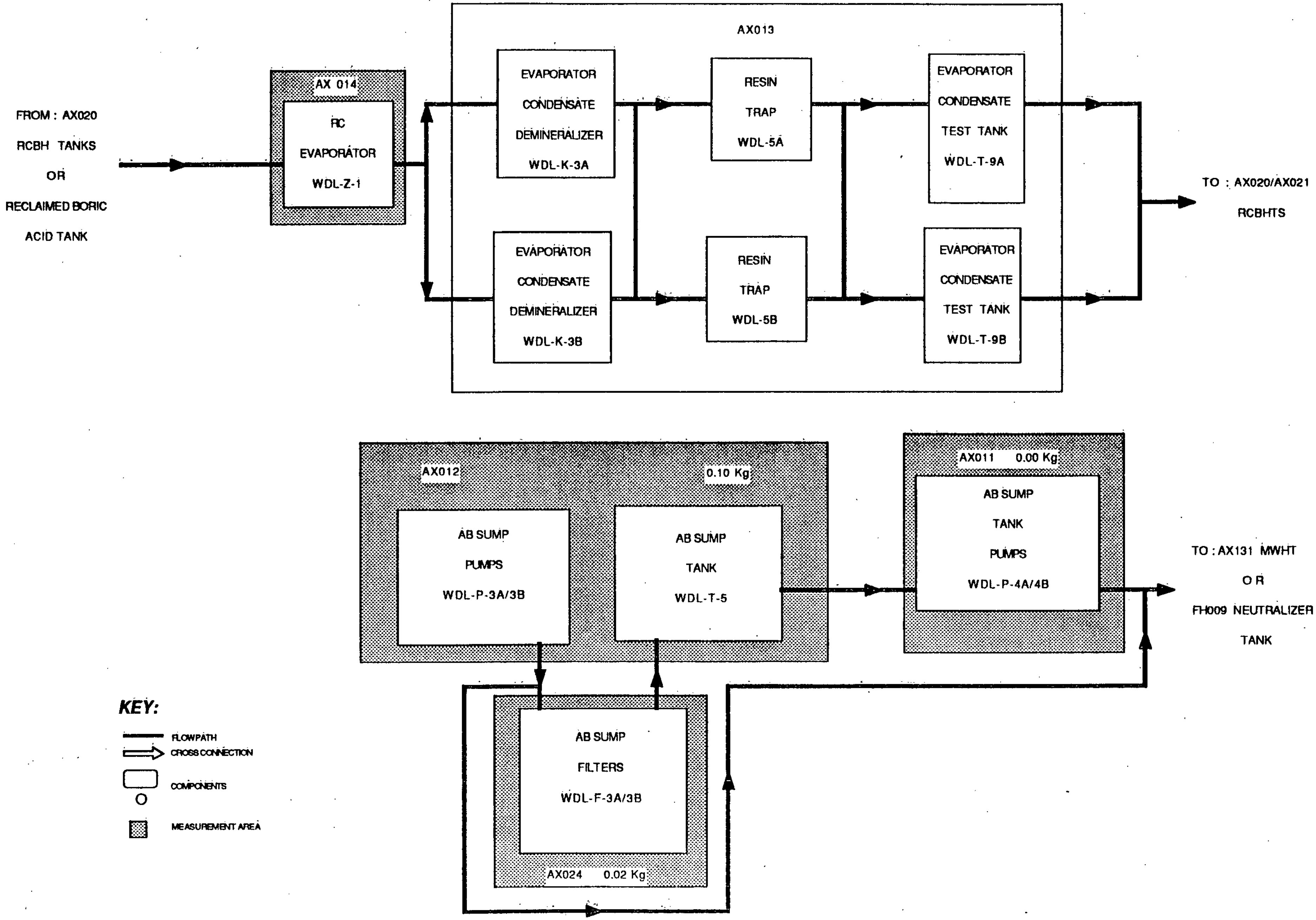


FIGURE 11
LIQUID RADWASTE DISPOSAL SYSTEM - MISCELLANEOUS WASTE
AUXILIARY BUILDING - ELEVATION 305'

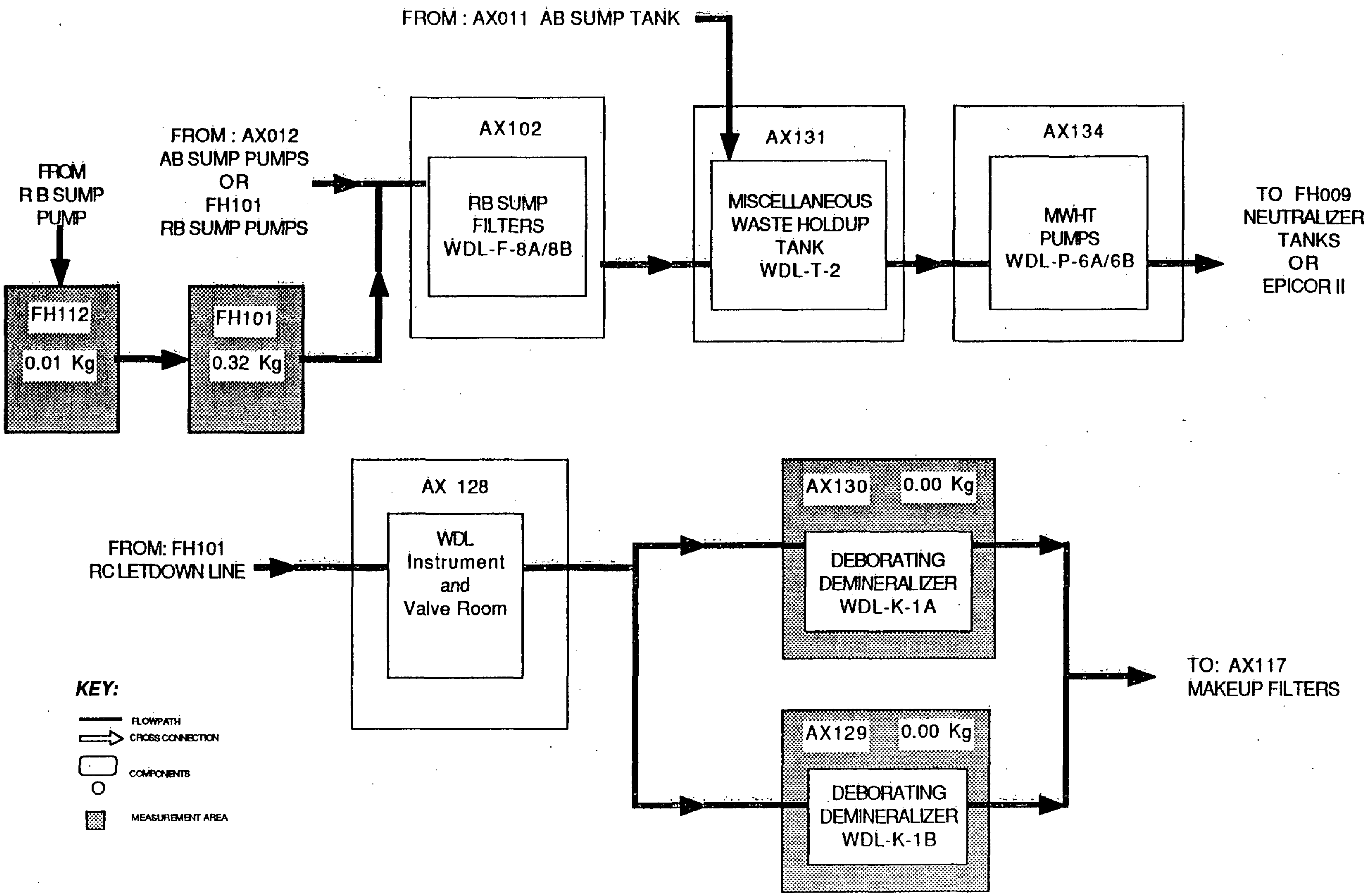


FIGURE 12
SOLID RADWASTE DISPOSAL SYSTEM -CONCENTRATED WASTE SUBSYSTEM FLOWPATH - AFHB - ELEVATIONS 281'6",305' & 328'

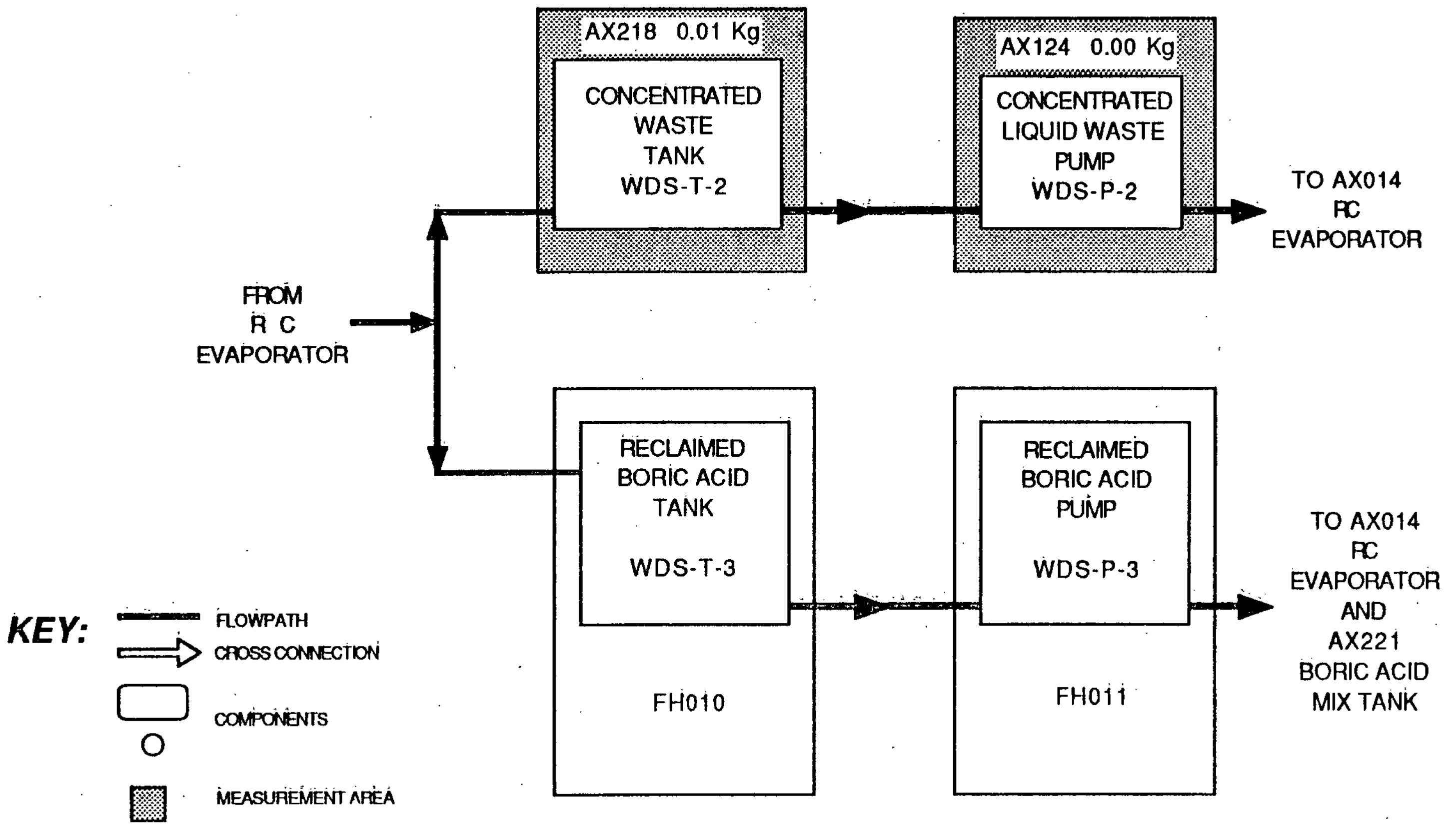


FIG E 13
SOLID RADWASTE DISPOSAL SYSTEM - RESIN WASTE SUBSYSTEM FLOWPATH - AUXILIARY BUILDING - ELEVATION 281' 6"

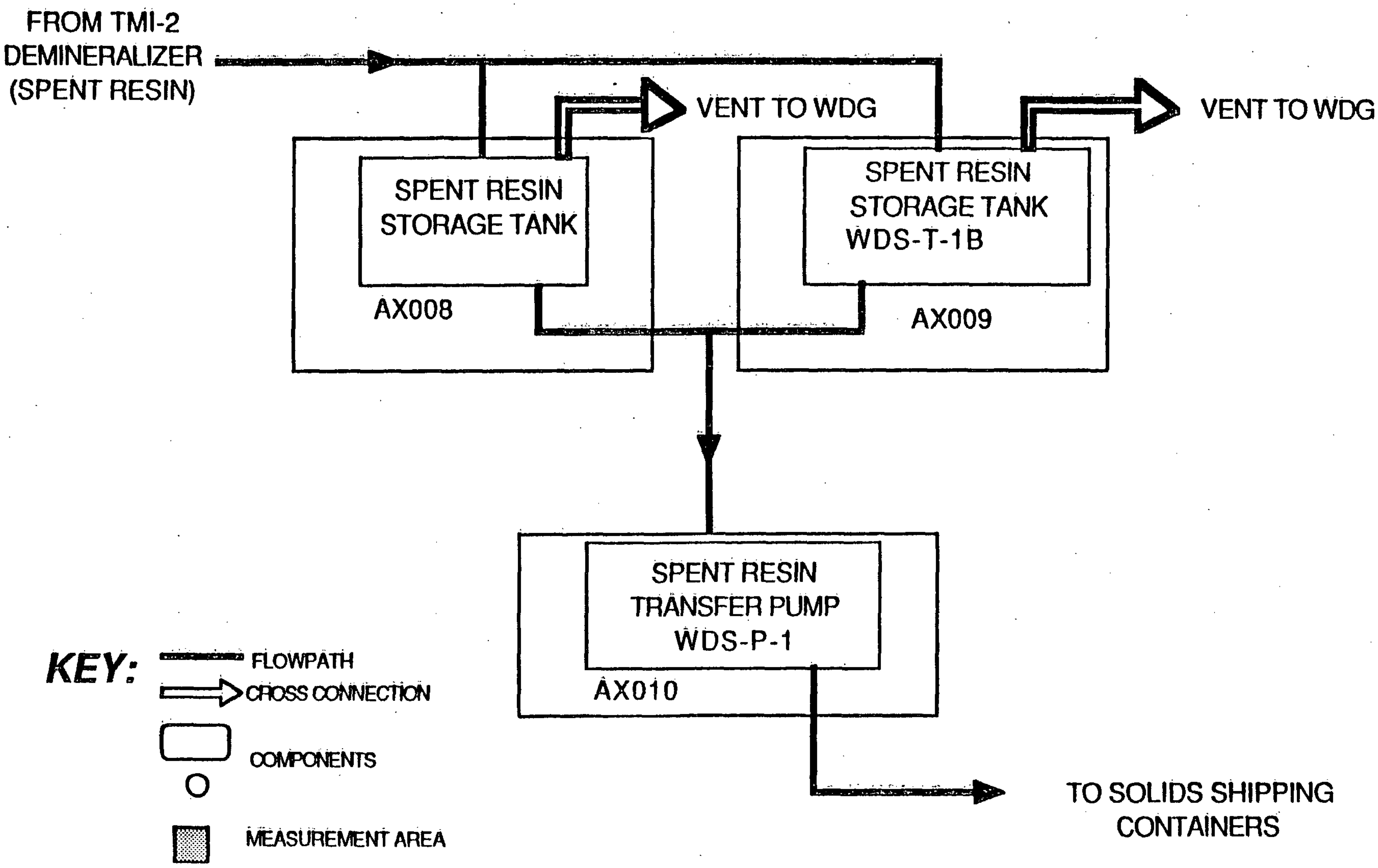


FIGURE 14
DWC FUEL TRANSFER CANAL/SPENT FUEL POOL CLEANUP SYSTEM
FUEL HANDLING BUILDING - ELEVATION 347'6"

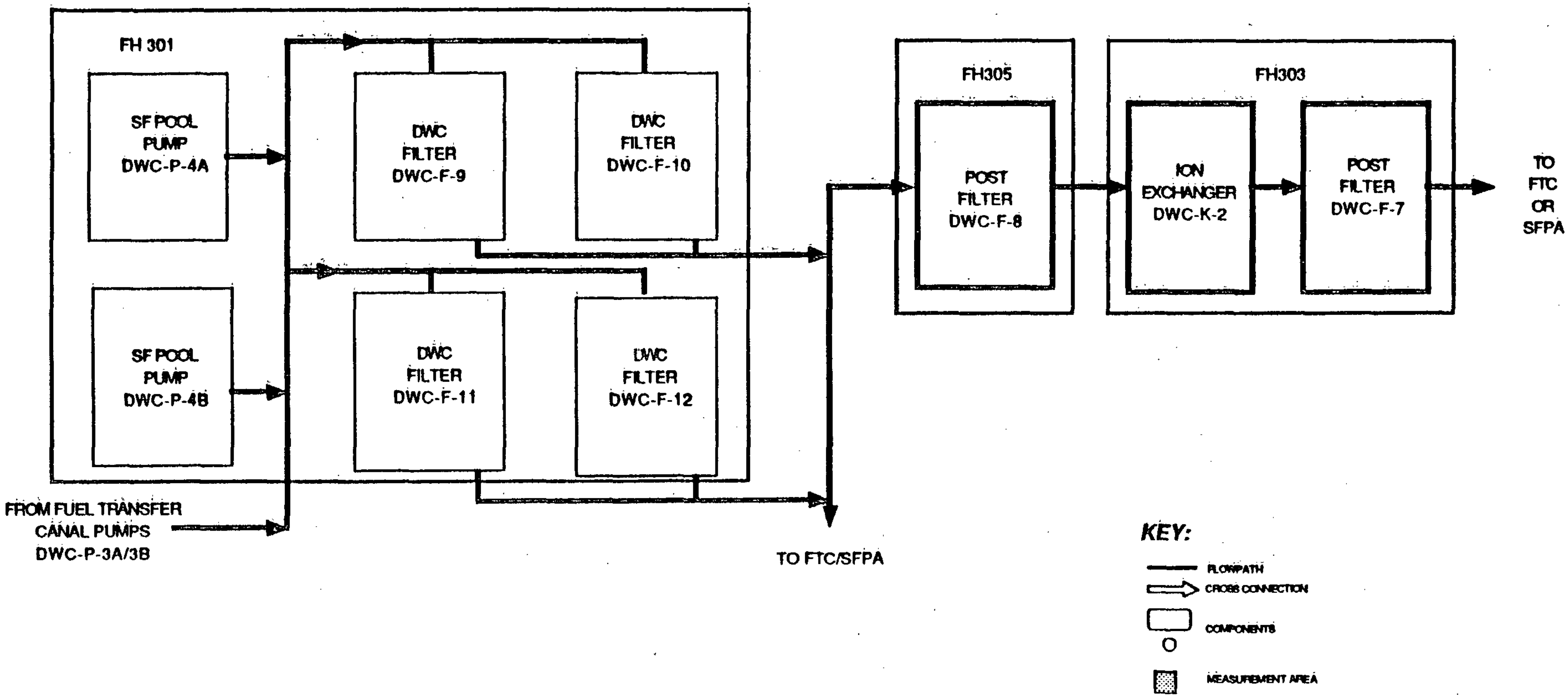


FIGURE 15
DWC REACTOR VESSEL CLEANUP SYSTEM
FUEL HANDLING BUILDING - ELEVATION 347' 6"

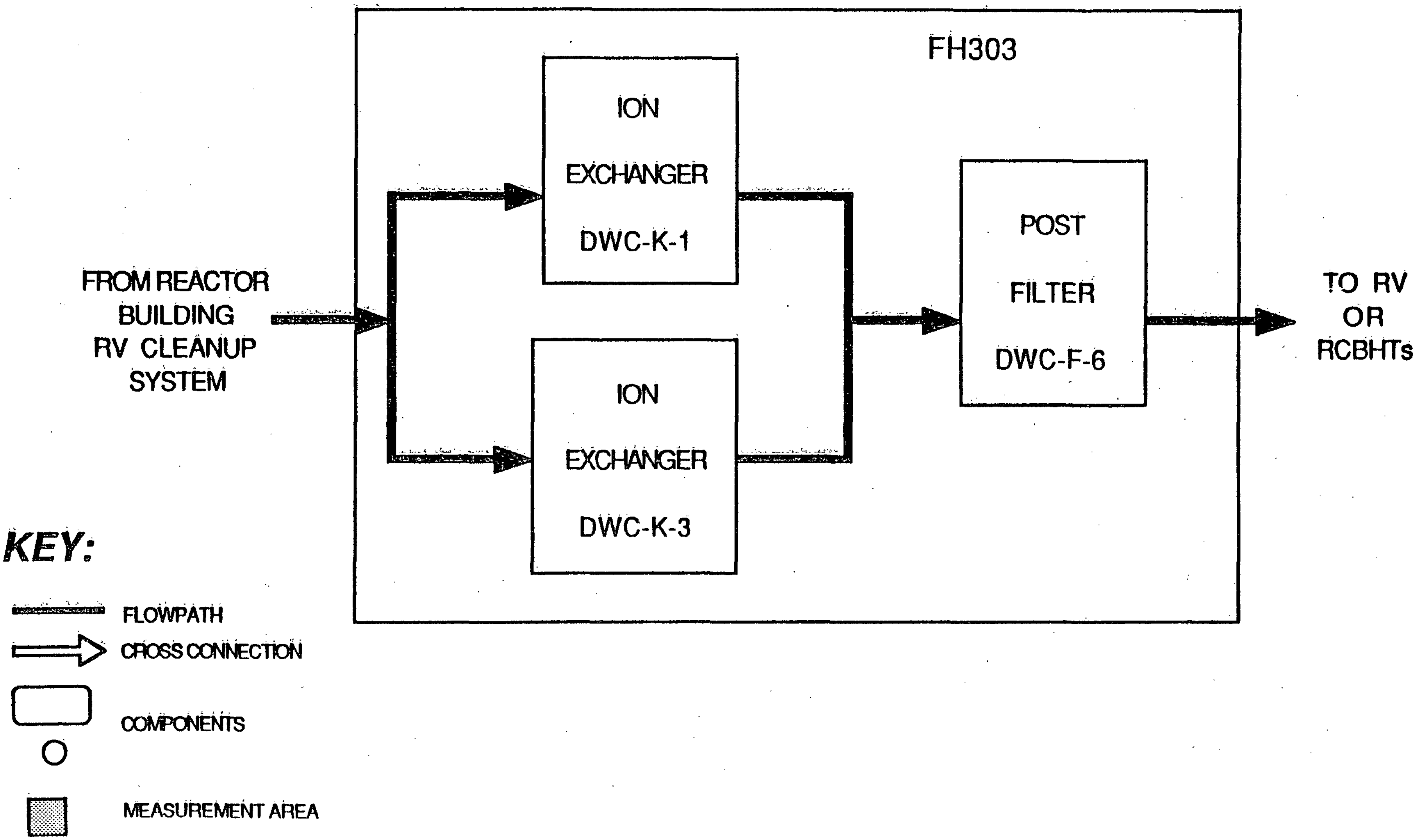
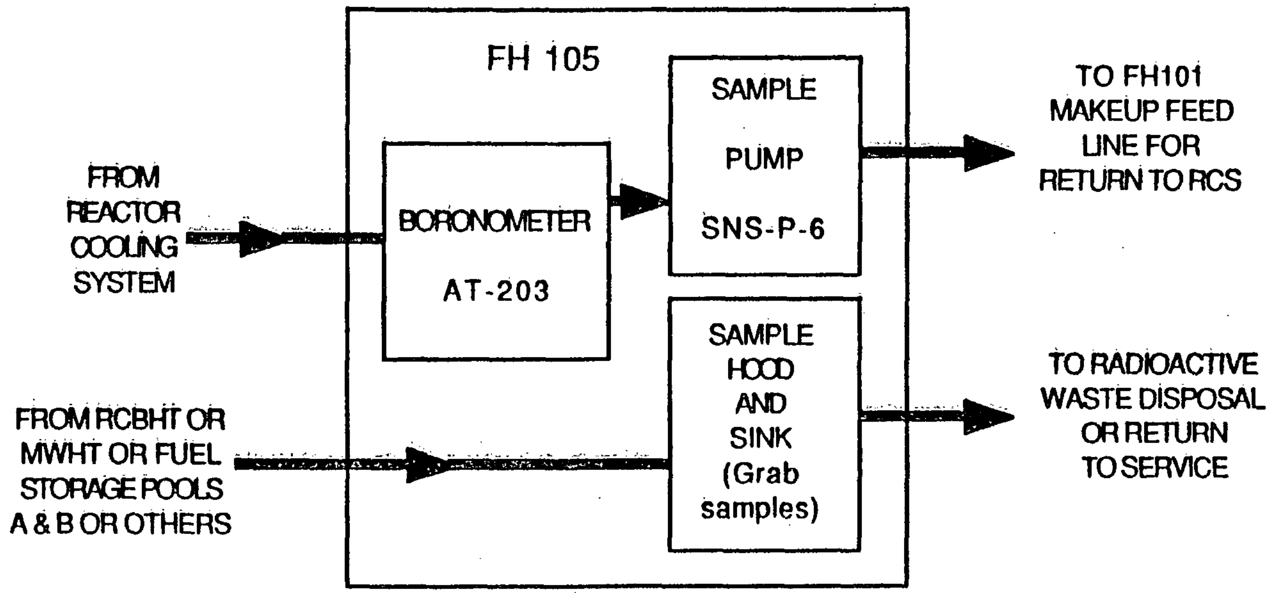


FIGURE 16
NUCLEAR SAMPLING SYSTEM
FUEL HANDLING BUILDING - ELEVATION 305'



KEY:

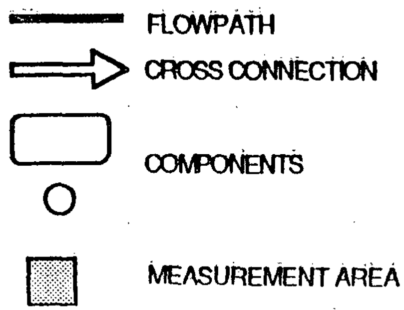
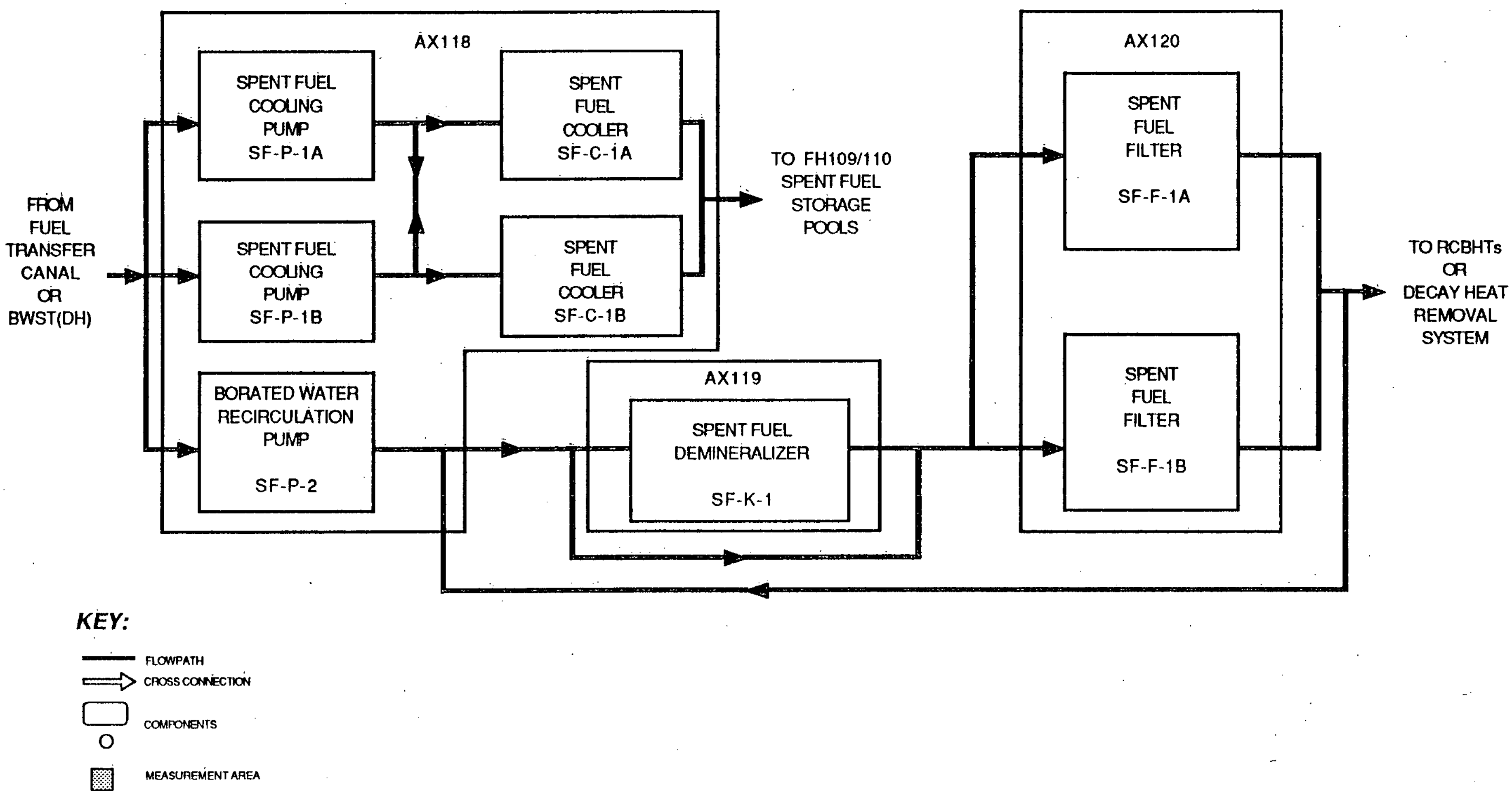


FIGURE 17
SPENT FUEL COOLING SYSTEM
AUXILIARY BUILDING - ELEVATION 305'



NOTE: AX118 PIPING ALSO USED DURING RECOVERY AS A FLOWPATH FOR THE DEFUELING WATER CLEANUP SYSTEM.

FIGURE 18
SUBMERGED DEMINERALIZER SYSTEM
FUEL HANDLING BUILDING - ELEVATION 305'

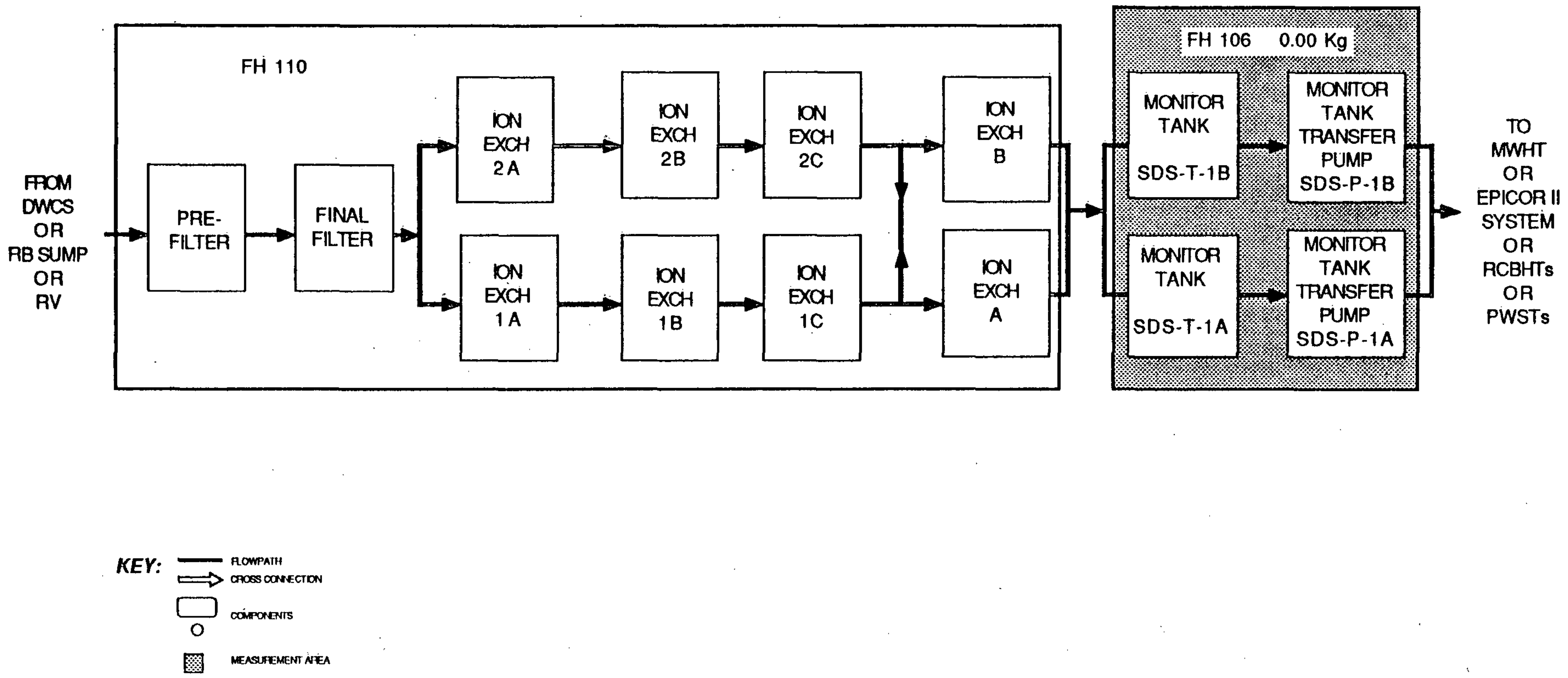
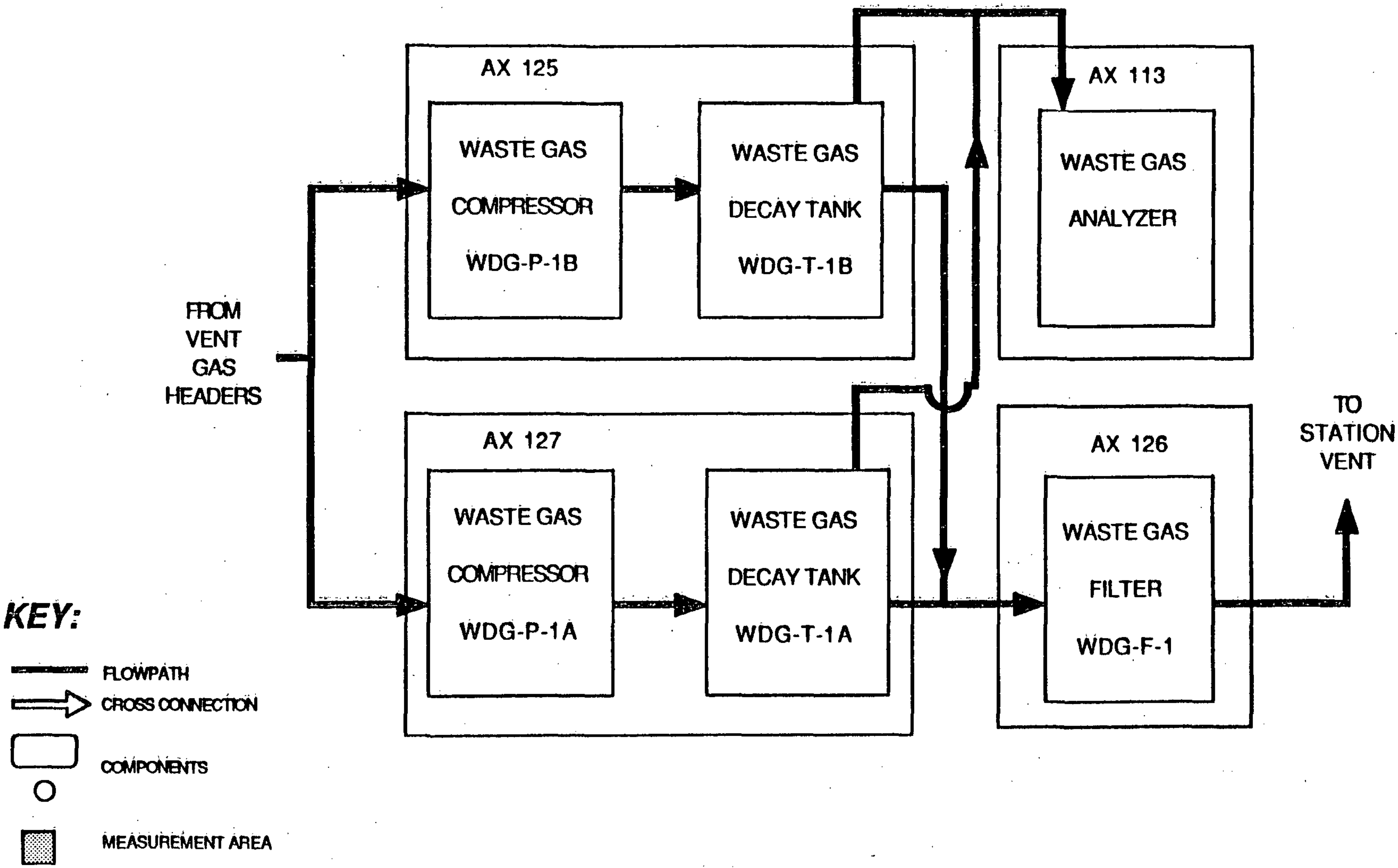


FIGURE 19
GAS RADWASTE DISPOSAL SYSTEM
AUXILIARY BUILDING - ELEVATION 305'



APPENDIX A

MAKEUP AND PURIFICATION SYSTEM

1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel (UO_2) remaining in the Makeup and Purification (MU&P) System contained in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are illustrated in Figures 5 and 6 which shows the primary flowpaths between areas containing MU&P components and piping in the AFHB. All statistical uncertainties are expressed as ± one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the MU&P system, and the part played during the 1979 TMI-2 Accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area in the AFHB containing MU&P system components and piping. The majority of areas were measured using gamma spectroscopy featuring either shielded sodium iodide scintillation or high purity germanium detectors connected to multi-channel analyzers. Since the presence of Ce-144(Pr-144) was not always identifiable, a MDL calculation was prepared when necessary.

Section 4, "Analysis", explains how the estimate of record of the amount of fuel in the MU&P system was determined and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the MU&P system and supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analyses performed.

2.0 BACKGROUND

Prior to the accident, the Makeup and Purification (MU&P) System was designed to control the Reactor Coolant System (RCS) water inventory as well as maintaining the water quality and chemistry within specifications. The system also served to:

a. provide seal injection water to the reactor coolant pumps
b. provide a means to separate and vent gases from the coolant
c. add makeup water to the Core Flood tanks
d. provide High Pressure Injection (HPI) water to the RCS in the event of a Loss-of-Coolant Accident (LOCA).

The MU&P System is interconnected to the following auxiliary systems in order to perform its functions:

a. Chemical Addition System (CA)
b. Radioactive Waste Disposal Systems (WDS, WDL, WDG)
c. Core Flood System (CF)
d. Decay Heat Removal System (DH)
e. Nitrogen Gas System (NM)
f. Demineralized Water System (DW)

During normal plant operation, reactor coolant flow was confined to a closed loop from the reactor core through the steam generators and back to the reactor. In order to maintain water quality, it was necessary to pass a portion of the flow (usually 35 gallons per minute) through a series of filters and demineralizers to remove contaminants. The water was then directed to the makeup tank (MU-T-1). The makeup tank served as a reservoir for coolant letdown from the RCS and provided a water source for the makeup pumps which return the water to the RCS. The makeup tank was also used to separate radioactive gases from the coolant. These gases were collected in the gas space of the MU tank and vented to the WDG System. The safety function of the MU&P System was to provide a high pressure flow of additional coolant to the core from the Borated Water Storage Tank in the event of a rupture in the RCS system piping.

During the TMI-2 accident, while the reactor coolant was being letdown from the RCS, fuel particles were distributed throughout the MU&P System. Part of the fuel deposited in the MU&P System crossed over into the interconnected systems during the defueling and recovery operations performed from 1979 through 1989. Additional fuel was relocated during the TMI-2 cleanup. Directional radiation surveys indicated that fuel and/or fission products were dispersed throughout much of the primary piping system as finely divided particles and/or as plating on surfaces. In general, the MU&P System piping and components have been flushed with processed water. In addition, the letdown block orifice and the Makeup filters have been removed. Several portions of the MU&P piping and components have been physically isolated and drained. A major portion of the remaining fuel deposited in the Makeup and Purification System is concentrated in large volume containers such as tanks and demineralizers and is mixed with sediment and demineralizer resins.

3.0 METHODS

Standard gamma spectroscopy methods utilizing NaI (Tl) and HPGe detectors were employed to measure the remaining fuel (UO_2) in 12 of the 18 areas occupied by the Makeup and Purification System. Ce-144(Pr-144) was selected as a tracer for fuel because of its reasonable half-life (284 days), relative high abundance, chemical similarity to reactor fuel which results in a low escape rate coefficient from reactor fuel, and its 2.19 MeV gamma-ray that is readily identifiable on a multi-channel analyzer (MCA). The ratios of Ce-144-and Eu-154-to-fuel (UO_2) were developed using sample data (Reference 6). Details of the specific methods used in each area are described in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report. When a Ce-144(Eu-154) peak was not identifiable, a MDL calculation was performed in accordance with Reference 14. The gross counts in the region of interest (ROI) for Ce-144 (Eu-154), converted to an MDL value, was divided by the measured detector efficiency, the calculated photon fluence rate per kilogram of fuel for the geometry and count time under consideration.

One (1) area, AX005, utilized a series of exposure rate measurements made with tungsten-shielded directional probes to quantify the amount of residual fuel. The remaining two (2) measured areas, AX114 and AX115, were initially measured using gamma-ray spectrometry (Reference 15), but their residual fuel values were revised using improved sample data obtained in 1983.

Three (3) other areas, FH002, FH004 and FH014, containing MU&P valves and piping were not formally measured but were analyzed based on their service and measured fuel deposits in similar areas. Details of this determination are discussed in Section 4.0, Analysis, of this appendix.

4.0 ANALYSIS

The MU&P System contains 0.4% of the fuel remaining at TMI-2. The MU&P System occupies portions of eighteen (18) areas in the Auxiliary and Fuel

Handling Buildings (AFHB), which are illustrated in Figures 5 and 6. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan, Reference 1. Fifteen (15) of these areas have been measured for fuel content, and the results are summarized in Table 4. Four (4) of the measured areas (AX006, AX007, AX026 and AX112) resulted in MDL values. Details of the methods and assumptions used to calculate the remaining fuel for each area can be found in the engineering calculations listed in Table 4.

The residual fuel determination for the MU&P Demineralizers "A" and "B" (AX114 and AX115) was based on resin/fuel samples taken in March and April 1983. These samples were believed to be more representative of the demineralizer radionuclide contents than previous samples. These samples were analyzed by Oak Ridge National Laboratory (ORNL), and the results (Reference 10) were used to determine the residual fuel values shown in Engineering Calculation 4249-3211-90-069, Revision 0 (Reference 16). The final fuel value for A Demineralizer was 1.06 Kg of UO_2 and for B Demineralizer was 0.13 Kg of UO_2.

Most of the resins have been removed from the 'A' demineralizer and some resins have been removed from the 'B' demineralizer. Flush water does not flow well out of the 'A' demineralizer and it is assumed that the resin outlet piping is filled with a hard, packed resin/fuel mixture. The 'B' demineralizer still contains a hard, agglomerated resin/fuel mixture in the lower portion of the tank.

Since both MU&P Demineralizers were measured for fuel (UO_2) content while full of resins, any fuel laden resins transferred out of the demineralizers should have reduced the quantity of fuel remaining in the MU&P Demineralizers. The measured residual fuel value for each demineralizer could be reduced to correct for any fuel transferred to the

Spent Resin Storage Tanks. However, since the amount of fuel transferred is unknown and radiation levels in both areas, AX114 and AX115, are still very high, it is assumed that most of the fuel originally measured is still in the piping at TMI-2. Some of the resin/fuel still remains in the 'A' and 'B' demineralizers (AX114 and AX115) and it is assumed that the balance of the fuel is deposited in the pipe lines between the demineralizers and the spent resin storage tanks and pump (AX008, AX009 and AX010). Since the measured residual fuel values for each MU&P Demineralizer are reliable data by themselves, it is assumed for purposes of SNM accountability that much of the fuel originally deposited in the MU&P Demineralizers is still remaining at TMI-2. Therefore, any fuel transferred from the demineralizers was already included in the residual fuel values established for the demineralizers and no additional fuel amounts are included for the Spent Resin Storage Tanks (AX008 and AX009) and the Spent Resin Storage Tank Pump (AX010).

Both of the MU&P demineralizers had been previously measured in 1982 using a collimated Si(Li) Compton recoil gamma-ray spectrometer (A Demineralizer) and a Be (Υ,n) detector (B Demineralizer) collecting gamma-ray flux data above 1.667 MeV. Even though the measured values were greater than the values obtained using the ORNL sample data, the measured values support the conclusion drawn in this PDSR. Details of the Be (Υ,n) and Si(Li) measurements can be found in Reference 15. The ORNL analyzed sample data was selected over the Si(Li) and Be (Υ,n) data for use in this PDSR because the sample analysis utilized a direct measurement of Uranium. The Si(Li) and Be (Υ,n) measurements relied on an analog ratio which inherently results in a larger uncertainty.

The three (3) remaining areas that were not formally measured and are the Makeup Valve Access Corridor (FH002), the Westinghouse (W) Valve Room (FH004), and the 281' Elevation Annulus (FH014). FH002 is an enclosed corridor between the Makeup Suction and Makeup Discharge Valve Rooms (FH001 and FH003a/b), which has HPI, pressurizer level control, and reactor coolant pump seal injection lines running through the overheads above its concrete slab ceiling. All of these lines originate in the assayed area of FH003a/b, have been flushed several times since the accident, and are assumed to contain very little fuel. These lines also pass through the other unmeasured areas; with the two HPI lines running through both FH004 and FH014, and the remaining lines running through FH014 into AX004, which is an assayed area.

Since these unmeasured areas contain pipes and valves which were exposed to similar core debris and are physically similar in layout and design to those contained in the 305' elevation Makeup Valve Room (FH101), a conservative appraisal of the unmeasured segments can be made using the FH101 value of 0.320 kg of UO_2. First, the amount and size of valves and pipes in FH101 is significantly greater than the total amount contained in the three unmeasured areas. Second, FH101 contains letdown piping that carried primary coolant which had not been filtered prior to entering FH101.

It is very unlikely that the three (3) areas (FH002, FH004 and FH014) would contain exactly the same amount of fuel as FH101. It is more likely that the amount of fuel in FH101 is an upper bound value for the amount of fuel in the three (3) areas. This argument is supported by the most recent general radiation survey data for each of the four (4) areas. FH101 survey data indicates an average general area exposure rate of 470 mR/hr while FH002, FH004 and FH014 indicate an average general exposure rate of 43 mR/hr. FH101 contains all of the letdown pathways which is where most of the core debris was initially transported during the accident while the other three (3) areas contain makeup pathways which generally contains much less residual fuel. In view of these arguments, it is assumed that the total quantity of residual fuel in FH002, FH004 and FH014 is 0.16 ± 0.08 kg of UO_2.

Approximately 10% of the process piping was assumed to be embedded in concrete walls and floors, and was not included in the fuel determination for the measured areas. Using the balance of the residual fuel (approximately 1.74 kilograms) deposited outside the makeup tank and demineralizers as a basis, an additional 10%, or 0.17 kilograms of UO_2, is assigned to the embedded valves and piping of the MU System.

This results in a fuel estimate of 2.81 Kg UO_2 ± 27% plus 0.60 Kg UO_2 (MDL value) for the entire Makeup and Purification System. The range of this estimate extends from 2.05 Kg to 4.17 Kg of UO_2. See Appendix F for the statistical determination of the range of values. After draindown of the RCS, including the reactor vessel, the MU&P System will be isolated from the RCS by administrative controls to insure that the containment isolation valves are maintained in the closed position.

5.0 CONCLUSION

The estimate of record of the amount of UO_2 remaining in the MU&P System is 3.4 kg (at one sigma). Based on the data shown in Table 4 and the data reduction discussion outlined in Appendix F, the range of the estimate of record extends from 2.05 Kg to 4.17 Kg UO_2. The MDL portion of the upper range value is 0.60 Kg UO_2.

This estimate of record is derived from existing measurement and sample data, and analyzed values shown for each area listed in Table 4. The MU&P System is expected to remain static since most of the areas have been drained and isolated, and the remaining areas will only be subjected to low flow rates necessary for final draindown. Additional measurements of the MU&P system are not considered to be warranted based on ALARA considerations and the small quantities of fuel measured to date. After draindown of the RCS, the MU&P System will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

APPENDIX B

LIQUID RADWASTE DISPOSAL SYSTEM

1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel remaining in the Liquid Radwaste Disposal (WDL) System contained in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are illustrated in Figures 7, 8, 9, 10, and 11 which show the primary flowpaths between areas containing WDL components and process piping in the AFHB. All statistical uncertainties are expressed as ± one (1) sigma limits (defined as one (1) standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the WDL System, and the part played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area in the AFHB containing WDL system components and process piping. The majority of areas were assayed using gamma spectroscopy featuring either shielded sodium iodide scintillation or high purity germanium detectors connected to multichannel analyzers. Since the presence of Ce-144(Pr-144) was not always identifiable, MDL calculations were prepared when necessary.

Section 4, "Analysis", explains how the estimate of record of fuel in the WDL System was determined and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the WDL System and supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analysis performed.

2.0 BACKGROUND

Prior to the accident, the Liquid Radwaste Disposal (WDL) System consisted of process equipment and instrumentation necessary to collect, process, monitor and recycle or dispose of radioactive liquid wastes. All potentially radioactive liquids generated in Unit 2 were collected and processed through the liquid radwaste system prior to recycling or discharge. The liquid radioactive waste was processed on a batch basis to permit optimum control of releases. Aerated drain wastes collected in the reactor drain system, e.g. containment and auxiliary building (AB) drains and laboratory drains, and demineralizer regeneration solutions were segregated based on their origin and activity, and processed through the miscellaneous waste system. The miscellaneous waste system consisted of hold-up tanks, an evaporator, demineralizers, filters, and waste test tanks. Prior to the accident, liquid wastes from normally nonradioactive areas were collected separately and were discharged without treatment into the yard drainage system. The discharge was preceded by routine sampling for radioactivity. If the value exceeded a predetermined level, the liquid waste was processed through the miscellaneous waste system.

The WDL system provided the following functions for the Reactor Coolant (RC) and other liquids:

a. Boron removal from Reactor Coolant for reactivity control
b. Collection of all liquid wastes
c. Hold-up storage of radioactive liquids to allow decay

d. Process radioactive liquids by ion exchange to remove corrosion and fission products so that its radioactivity level will be as low as practicable

e. Redundant sampling and monitoring of the liquids before discharging

f. Controlled discharge of the liquid waste

g. Recovery of boric acid by evaporation

h. Volume reduction of the radioactive liquid waste by evaporation

i. Stripping of radioactive gases

j. Venting of gases to Waste Gas System

k. Transfer of concentrated wastes for disposal

l. Positive control of all liquid releases

m. Filtering and chemical treatment of liquid releases

During normal plant operation, prior to the 1979 accident, the concentration of boron in the reactor coolant was reduced as the fuel was consumed with core burn-up. Initially the concentration was reduced by simultaneously bleeding highly borated reactor coolant and feeding non-borated coolant into the reactor loops. The highly borated coolant was stored in one of three (3) RC Bleed Holdup Tanks (RCBHT). The non-borated water was fed into the suction of the Makeup (MU) Pumps where it was injected into the reactor loops. Feed and bleed operations were controlled from the Feed and Bleed Panel located in the Control Room.

The RCBHTs collected liquid from the following sources:

a. Reactor Coolant Letdown Bleed

b. Reactor Coolant Drain Header

c. Reactor Coolant Drain Tank

d. Letdown Line Relief Valve Drain

e. Reactor Coolant Evaporator Distillate

f. Makeup Tank Drain

g. Core Flood Tank Drain

h. Spent Fuel Cooling System Purification Loop

i. Evaporator Condensate Test Tanks

Before removal from the RCBHTs, the contents were recirculated to obtain a representative sample for determining disposition of the liquid.

The RC Evaporator was used to concentrate non-volatile radioactive liquid and boric acid. The concentrated liquid was transferred either to the Concentrated Waste Tank (CWT) for disposal or to the Reclaimed Boric Acid Tank for reuse. The volatile and non-condensable gases were removed by a gas stripper and directed to the Radwaste Gas System. The distillate was either pumped through the Evaporator Condensate Demineralizers to the Evaporator Condensate Test Tanks, or it was returned to a RCBHT for storage.

The miscellaneous liquid disposal portion of the WDL System was originally designed to collect, handle, sample and dispose of miscellaneous liquid waste generated throughout the plant. The system tanks and pumps were designed to store the average amount of liquids used or processed in one (1) week. The system was designed to minimize the release of radioactivity to the environment. The discharge paths were monitored for radioactivity and were automatically shut off if radioactivity limits were approached. The miscellaneous liquids disposal portion of the WDL System received liquids from the Service Building, the Auxiliary Building, the Fuel Handling Building and the Reactor Building Sump. These liquids were processed to chemically neutralize and/or reduce the radioactivity level prior to discharge.

The Auxiliary Building (AB) waste system collected liquid from tank drains, reliefs, vents, filter drains, flush line drains, and all the AFHB floor drains. The liquid wastes were pumped into the AB sump tank. After hold-up, recirculation and sampling, the liquids were normally pumped to the Neutralizer Tanks. After any necessary chemical treatment in the Neutralizer Tank and dependent on the radioactivity level, the liquids were directed to the plant discharge line or storage tanks for long term storage. In summary, all waste liquids were treated so that the solution was neutral or slightly basic.

During the TMI-2 accident, the reactor coolant was letdown from the Reactor Coolant System (RCS) and fuel particles were distributed into many areas of the WDL System. Some of these fuel particles were further distributed into other interconnected systems during the defueling and cleanup operations from 1979 through 1989. A major portion of the fuel remaining in the WDL system is concentrated in large volume containers such as tanks and demineralizers, and is mixed with sediment or demineralizer resins. Other fuel particles are deposited in low velocity areas and dead-ended piping not accessible to flushing action. The primary purposes of the WDL System during the recovery phase was to store water to be used as makeup to the RCS, process stored water through SDS or EPICOR II, and collect water that was letdown from the RCS.

3.0 METHODS

Standard gamma spectroscopy methods utilizing NaI (Tl) HPGe detectors were employed to assay the remaining fuel (UO_2) in seventeen (17) of the twenty-nine (29) areas occupied by the WDL System. Cerium-144(Pr-144) was selected as a tracer for the reactor fuel because of its reasonable half-life (284 days), relative high abundance, low escape rate coefficient from reactor fuel and its 2.19 MeV gamma-ray that is readily identifiable on a multichannel analyzer (MCA). The HPGe system was also sometimes able to detect Eu-154 which was used if it was detectable and Ce-144(Pr-144) was not. Ce-144-and Eu-154-to-fuel (UO_2) ratios were developed using sample data (Reference 6). Details of the specific methods used in each area are described in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report. When a Ce-144(Eu-154) peak was not identifiable, a MDL calculation was performed in accordance with Reference 14. The gross counts in the ROI for Ce-144(Eu-154), converted to an MDL value, was divided by the measured detector efficiency, the calculated photon fluence rate per kilogram of fuel, and the count time.

One (1) area, AX024 utilized a series of gross gamma exposure rate measurements in conjunction with a tungsten-shielded directional probe to quantify the amount of remaining residual fuel. Two (2) other areas, AX129 and AX130 were sampled, and the samples were assayed for fuel content. The residual fuel content in the remaining nine (9) areas was not measured but instead has been determined by analysis of measured fuel deposits in areas with similar flow origin and water processing history. Details of these estimations are discussed in Section 4, Analysis, of this appendix.

4.0 ANALYSIS

The WDL System contains 0.5% of the fuel remaining at TMI-2. The WDL system occupies portions of twenty-nine (29) areas in the Auxiliary and Fuel Handling Buildings (AFHB), which are shown in Figure 7, 8, 9, 10, and 11 except for AX101. The Radwaste Disposal Panel (AX101) is not connected to any of the processing pathways. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan. Twenty (20) of these areas have been assayed for residual fuel, and the results are summarized in Table 5. Seven (7) of the measured areas (AX014, AX015a, AX015b, AX016, AX017, AX018, and AX019) resulted in MDL values. Details of the methods and assumptions used to calculate the remaining fuel for each area can be found in the engineering calculations listed in Table 5. Since the RC evaporator (AX014) was assayed by Science Applications, Incorporated, their report (Reference 17) is listed as a reference instead of a calculation.

The initial methodology used to estimate the residual fuel (UO_2) in the cleanup filters (AX015A,B) and the cleanup demineralizers (AX016 and AX017) estimated a total of 0.317 kg of UO_2 present in all four (4) areas (Reference 18). A review of the data and methods used in the initial calculation indicate that a new approach to data reduction would provide improved and more accurate fuel estimates in the areas of concern. The revised estimates of the residual fuel (UO_2) in the pre-filter cleanup filters (WDL-F-6A and 6B) indicated that AX015A contains 0.055 kg of UO_2 and AX015B contains 0.044 kg of UO_2. Both of these values were MDL values. No measurements of the two (2) cleanup demineralizers (AX016 and AX017) were performed and since fuel particles would not be expected to pass through the cleanup pre-filters; the cleanup demineralizers were estimated to contain no fuel or negligible quantities of fuel (UO_2). Similarly, the after-filters (WDL-F-9A and 9B) were estimated to contain no fuel or negligible quantities of fuel. No additional measurements were justified based on the small quantities of fuel estimated and ALARA considerations. A more detailed discussion of the latest methodology is contained in Reference 19.

The remaining seven 7 areas are the Evaporator Condensate Test Tanks and Pumps (AX013), the Radwaste Disposal Control Panel Area (AX101), the RB Sump Pump Filters (AX102), the Instrument and Valve Room (AX128), the Miscellaneous Waste Holdup Tank (AX131), the Miscellaneous Waste Holdup Tank Pump (AX134) and the Neutralizer Tank Filters (FH012). The Evaporator Condensate Test Tanks (AX013) normally collected distillate from the Reactor Coolant Evaporator via one of the Evaporator Condensate

Demineralizers. During the recovery phase for TMI-2, the Evaporate Condensate Test Tanks were used for storage of decontamination water used to clean the walls and floors of contaminated areas in the Reactor, Auxiliary and Fuel Handling Buildings. Prior to storage in the tanks, the water was processed through the EPICOR II System to decontaminate the water by filtration and ion exchange. Recent (1987) water samples and area survey maps show very low radioactivity in AX013. The Monitor Tanks (FH016) have also served to store decontaminated water which was processed through filters and ion exchangers. Recent assay measurements performed in FH106 determined its residual fuel content as 0.003 kg. The Monitor Tanks are the same size, 12,000 gallons each, as the Evaporator Condensate Test Tanks and since both sets of tanks were used in the same capacity during the cleanup phase, it was reasonable to assume they contained equivalent amounts of fuel. Even if the Evaporator Condensate Test Tanks contained double that amount measured in the Monitor Tanks, the resulting difference of 0.003 kg was insignificant in comparison to the uncertainty in the total fuel remaining at TMI-2.

The Radwaste Disposal Control Panel Area (AX101) contained a local Control Panel for the liquid, gaseous and solid radwaste disposal systems. This panel displayed the position of valves, flow rates of mediums being processed, tank levels, tank temperature, tank pressure, and differential pressure indication for all demineralizers and filters. This local control Panel contained only electrically generated signals and consequently was completely isolated from all processing pathways and radioactive materials. Since there was no piping of any kind connected to the local control panel, it was concluded that there was no residual fuel in the Radwaste Disposal Control Panel area.

The Reactor Building (RB) Sump Pump Filter Area (AX102) contained the filters used to filter influent to the Miscellaneous Waste Holdup Tank

(AX131). The RB Sump Pump Filters were used during the TMI-2 accident to filter the water pumped from the flooded RB basement to the Auxiliary Building (AB). Post accident sampling of the sludge in the RB basement found it to contain small quantities of fuel. Therefore, some fuel may have been transferred from the RB basement and deposited in AX102 during the accident. Since the TMI-2 accident, there has been no transfers from the RB basement to the AB sump via the RB sump filters. The RB sump filter elements installed during the accident were removed during 1980 and disposed of as radioactive waste. Subsequent to the accident, the RB sump filters were used routinely to filter water transferred from the AB sump to the Miscellaneous Waste Holdup Tank (MWHT), WDL-T-2. From 1980 to the last change on June 3, 1988, there were over 30 filter change outs of the RB sump filters. Since June 1988, the water processed through the RB sump filters came from the AB sump. Due to the numerous filter change outs, AX102 probably does not contain more fuel than was determined for the AB Sump which was 0.10 kg UO_2.

The Miscellaneous Waste Holdup Tank (AX131) and the Miscellaneous Waste Holdup Tank (MWHT) Pumps (AX134) are the next components in the processing sequence following the RB sump filters. The next component in the processing sequence is the MWHT pumps which were protected from fuel deposits by the filters and the settling aspects of a large volume tank, MWHT. Considering the processing sequence, it was judged that the total fuel deposits in the RB Sump Pump Filters, the MWHT and the MWHT Pumps could not exceed the fuel measured in the AB Sump (AX012). The AB Sump was a relatively large volume, 7600 gallons, and acts as a settling volume for influent particulates. Based on this settling effect and the numerous filter changes since the accident, it was deduced that there cannot be more fuel deposits downstream from the AB Sump than were actually deposited in the AB Sump itself.

Spectroscopy measurements taken as part of the SNM Accountability Program determined the residual fuel (UO_2) remaining in the AB Sump was 0.10 kg (± 0.10). Therefore, it was judged for this PDSR that the total fuel remaining in the RB Sump Pump Filter area (AX102), the Miscellaneous Waste Holdup Tank (AX131), and the Miscellaneous Waste Holdup Tank Pump (AX134) was 0.10 kg (± 0.10). It is unlikely that transfer of 100% of the fuel could be achieved by simply pumping the AB Sump; the 0.10 kg of fuel appraisal is considered a bounding value.

The Instrument and Valve Room (AX128) contained liquid radwaste processing lines, vent headers for the Radwaste Disposal - Gas System and miscellaneous piping and valves for both systems. The gaseous radwaste disposal system was assumed to contain very little if any fuel deposits because vent gas provided very little driving force to distribute fuel throughout the system. In addition, most venting pathways start out vertically and with very little driving force; transport of significant fuel particulates was highly unlikely. The greater contributor to fuel deposits in AX128 was more likely in the liquid radwaste disposal processing piping which was no longer in use. All of the lines have been flushed and are relatively free of fuel deposits except possibly in the dead-ended piping where flushing was not effective. AX128 is similar to the WDL Valve Room (AX019) but on a much smaller scale. Spectroscopy measurements performed in AX019 resulted in a MDL fuel value of 0.01 kg, and there were many more areas to trap fuel deposits in AX019. Based on the similarity of both areas and that both areas were exposed to similar processing flow conditions during and following the accident, it was concluded that the fuel (UO_2) deposit in the Instrument and Valve Room (AX128) was bounded by the 0.01 kg fuel quantity determined for AX019.

The last unmeasured area in the Liquid Radwaste Disposal System was the Neutralizer Tank Filter Area (FH012). The Neutralizer System was used during the cleanup phase as a batch tank to receive effluent from the MWHT, the AB Sump and the Contaminated Drain Tank, and processed its contents through the EPICOR II System for filtration and purification via ion exchangers. Although originally intended to be used to chemically treat waste liquid, the Neutralizer System had not been used in that manner since the TMI-2 accident. The Neutralizer Tank Filters (WDL-F4A/B) were changed out frequently during their use and were last changed out on June 1, 1990. The spectroscopy measurement performed in the Neutralizer Tank Room (FH009) determined the total residual fuel (UO_2) was 0.003 kg. Since the Neutralizer Tank was located upstream from the Neutralizer Tank Filters, and the Neutralizer Tanks acted as a settling volume, and the filters elements were frequently changed out, it was deduced that the fuel deposits in the Neutralizer Tank Filter Area were bounded by the 0.003 kg value for FH009.

Approximately 10% of the processing piping was assumed to be embedded in concrete walls and floors and was never included in the fuel appraisal for the measured areas. Using the balance of the residual fuel (approximately 0.36 kilograms) deposited outside the RC Bleed Holdup Tanks as a basis, an additional 10% or 0.04 kilograms of fuel was assigned to the shielded valves and embedded piping of the WDL system.

This results in a fuel estimate of of 4.13 Kg UO_2 ± 71% plus 0.11 Kg UO_2 (MDL value) for the entire Liquid Radwaste Disposal System. The range of this estimate extends from 1.20 Kg to 7.17 Kg UO_2. See Appendix F for the statistical determination of the range of values. Since the RCBT measurements were completed, the potential for fuel mobility has been greatly reduced. Water transfer has been limited, and the tank tends to act as a settling area because of the low water velocity inside the tank.

5.0 CONCLUSION

The best estimate of record of the amount of UO_2 remaining in the WDL System is 4.2 kg (at one sigma). Based on the data shown in Table 5 and the data reduction discussion outlined in Appendix F, the range of the estimate of record extends from 1.20 kg to 7.17 kg UO_2. The MDL portion of the upper range is 0.11 kg UO_2.

This estimate of record is derived from the existing measurement and sample analysis data, and the appraisal values shown for each unmeasured area listed in Table 5. The uncertainty or range is based on the statistical data combination method described in Appendix F. The WDL System is expected to remain static since many of the areas have been drained and isolated, and the remaining areas have been subjected to water transfer on a regular basis. Consequently, all of the fuel that would move has already been relocated to a tank, filter or other low velocity location. Additional measurements of the WDL System are not considered to be justifiable based on ALARA considerations and the small quantities of fuel (UO_2) measured to date. After draindown of the RCS, the WDL System will be isolated from the RCS by maintaining the containment isolation valves in a closed position.

APPENDIX C

SOLID RADWASTE DISPOSAL SYSTEM

1.0 INTRODUCTION

This appendix presents the analysis of the amount of UO_2 remaining in the Solid Radwaste Disposal (WDS) System contained in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are illustrated in Figures 12 and 13 which show the primary flowpaths between areas containing WDS components and process piping in the AFHB. All statistical uncertainties are expressed as $\pm$ one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the WDS System, and the part played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area in the AFHB containing WDS System components and process piping. One area was assayed using gross gamma exposure rate measurements, and a second area was assayed using the standard gamma spectroscopy utilizing a HPGe detector connected to a multichannel analyzer.

Section 4, "Analysis", explains how the estimate of record of fuel in the WDS System was determined and discusses supporting data, assumptions made and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the WDS System and the supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analysis performed.

2.0 BACKGROUND

Prior to the accident, the function of the WDS System was to store and transfer bead type resins, concentrated liquid wastes and reclaimed boric acid. The WDS System was composed of two (2) independent subsystems, the resin waste subsystem and the concentrated liquid waste subsystem.

The resin waste subsystem was designed to process bead type ion exchange resins from the demineralizers. The resins exchange ions of radioactive impurities and chemically bind the ions to the resin. Additionally, the resin filters out insoluble radioactive material and activated insoluble corrosion and fission products. The activated corrosion products were formed by corrosion of metalic components and flaking of corroded surfaces which were activated while in transit through the RCS. Fission products formed during fissioning of the uranium fuel entered the reactor coolant by destruction of most of the fuel rod cladding (TMI-2 accident). The corrosion products are insignificant compared with the activity made available through destroyed fuel rod cladding.

Each Spent Resin Storage Tank (WDL-T-1A/B) was designed with a capacity of 516 cubic feet. As designed, the two tanks combined had a nominal two year resin retention capacity. The resin waste subsystem consisted of the Spent Resin Storage Tanks (SRST's), WDS-T-1A and WDS-T-1B, and the Spent Resin Transfer Pump (SRTP), WDS-P-1. The SRTP took suction from either tank and could either recirculate or discharge the contents to storage for off-site disposal.

During the cleanup phase of the Cleanup Program at TMI-2, the resin waste subsystem was used to collect and process spent resins from all the TMI-2 demineralizers and sludge from the RB basement. The greatest quantity of radioactive resins came from the 'A' MU&P demineralizer and was eventually shipped off-site for disposal.

The concentrated liquid waste subsystem was designed to handle liquid wastes which appeared in two basic forms:

a. Concentrated radioactive boric acid solution
b. Miscellaneous radioactive liquid wastes.

The activity in both of these liquids resulted from the same sources that challenged the solid system. The concentrated liquid waste subsystem consisted of the Concentrated Waste Tank, WDS-T-2, and the Reclaimed Boric Acid Tank, WDS-T-3. Each tank had its own recirculation/discharge pump and could receive liquids from or have its contents pumped to the Reactor Coolant Evaporator or any other storage tank. The liquid in the Reclaimed Boric Acid Tank was normally pumped to the Boric Acid Mix Tank for reuse. Wastes stored in the Concentrated Waste Tank or the Reclaimed Boric Acid Tank could be reprocessed through the Reactor Coolant Evaporator for further concentration. Normally the wastes were stored to allow radioactivity to decay to a lower level for processing, packaging and subsequent off-site disposal. Due to the intensely radioactive nature of this charge, the WDS System was designed to be controlled remotely. The wastes were sampled to determine their radioactivity and chemical makeup. All operations were batch type and were initiated and terminated by operator action.

During the Cleanup Program at TMI-2, the concentrated liquid waste subsystem was used mostly for storage and solidification of radioactive wastes.

3.0 METHODS

Only two (2) areas containing WDS System components and piping were formally measured to determine the remaining fuel (UO_2). One (1) area, AX218, which contained the Concentrated Waste Tank, was measured utilizing a HPGe detector. Cerium-144(Pr-144) was used a tracer for fuel because of its reasonable half-life (284 days), relative high abundance, low escape rate coefficient from fuel and its 2.19 MeV gamma-ray that was readily identifiable on a multichannel analyzer (MCA). A Ce-144-to-fuel (UO_2) ratio was previously developed using sample data (Reference 6).

The second area, AX124, which contained the Concentrated Liquid Waste Pump, was measured by using a series of gross gamma exposure rate measurements made with a tungsten-shielded directional probe. Details of the specific methods used for both areas are described in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report. The remaining five (5) unmeasured areas were not formally assayed but were appraised based on their service history and measured fuel deposits in similar areas which were exposed to similar service conditions. Details of these appraisals are discussed in Section 4, Analysis, of this appendix.

4.0 ANALYSIS

The WDS System contains less than 0.1% of the fuel (UO_2) remaining at TMI-2. The WDS System occupies portions of seven (7) areas in the Auxiliary and Fuel Handling Building (AFHB), which are illustrated in Figures 12 and 13. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan (Reference

1). Two (2) of these areas have been measured for residual fuel and the results are summarized in Table 6. The five (5) remaining areas are the two (2) Spent Resin Storage Tanks (WDL-T-1A/B), the Spent Resin Transfer Pump (WDS-P-1), the Reclaimed Boric Acid Tank (WDS-T-3), and the Reclaimed Boric Acid Pump (WDS-P-3). The Spent Resin Storage Tank (SRST) areas, AX008 and AX009, and the Spent Resin Storage Tank Pump area, AX010, were never formally surveyed to determine their fuel content. The piping, tanks, and pump in AX008, AX009 and AX010 were not in the makeup, letdown or waste disposal liquid flowpaths at the time of the accident. However, as a result of resin transfer operations, they probably contain small quantities of fuel. Since the TMI-2 accident, the SRST's were used to collect spent resins from all the TMI-2 demineralizers and were most recently used to collect the resins from the MU&P Demineralizers during 1988. Both SRST's were emptied and flushed early in 1989 and were partially refilled with water for shielding purposes. Their contents were estimated to each contain less than one cubic foot of resin and approximately 600 gallons of water. Since the measured residual fuel values for each MU&P Demineralizer are reliable data, it is assumed for purposes of SNM accountability that any fuel transferred to the SRSTs and dedicated transfer pump has already been accounted for in the MU&P Demineralizer data and no additional quantity is required for the SRSTs and transfer pump. This argument holds true for all the demineralizers at TMI-2. All possible fuel in the SRSTs and transfer pump had to come from a demineralizer (by design) and all demineralizers containing fuel were measured while full of resins; therefore, all possible remaining fuel in the SRSTs and transfer pump has already been included in the residual fuel values established for the various demineralizers. No additional fuel amounts are included in the Summary of Solids Radwaste Disposal System SNM Inventory, Table 6.

The only other unmeasured areas in the WDS System were the Reclaimed Boric

Acid Tank (RBAT), FH010, and the Reclaimed Boric Acid (RBA) Pump, FH011. The RBAT and RBA Pump were never used for their designed function but were used to store accident generated water for a relatively short period. This tank and pump were utilized much less frequently than the Neutralizer Tank, FH009, and Neutralizer Tank Pump, FH008, which were also used to store accident generated water. The Neutralizer Tank and Pump were assayed for UO_2 using gamma spectroscopy and the results were 0.003 kg for the tank and 0.001 kg for the pump. Based on the similarity of service and the short duration of use, it was assumed that the assayed contents for the Neutralizer Tank and Pump were bounding values for the RBAT (FH010) and the RBA Pump (FH011). FH010 and FH011 were assigned residual fuel values of 0.003 kg and 0.001 kg, respectively. Both of these values round off to 0.00 Kg UO_2.

Approximately 10% of the process piping is assumed to be embedded in concrete walls and floors and was never included in the fuel quantities for the measured areas. Using the total of the determined fuel values for each area in the WDS System as a basis, an additional 10% or 0.00 kilogram of fuel was assigned to the shielded piping of the WDS System.

This results in a fuel estimate of record of 0.01 (rounded to 0.0) kilograms of UO_2 for the entire Solid Waste Disposal System. The range of this estimate extends from 0.00 Kg to 0.02 Kg UO_2. See Appendix F for the statistical determination of the range of values. It is possible that additional attempts will be made to remove resins from the 'B' Makeup and Purification Demineralizer. In any event, once the accident generated water has been evaporated, the potential for mobility is greatly reduced because only small batches of water will be processed during the post-defueling storage period.

5.0 CONCLUSION

The estimate of record of the amount of UO_2 remaining in the WDS System is 0.0 kg (rounded from 0.01 Kg).

This estimate of record is derived from the existing measurements and analyzed data summarized in Table 6. The uncertainty or range is based on the statistical data combination method described in Appendix F. The WDS System is expected to remain static since most of the areas have been drained and isolated, and the remaining areas will not be exposed to flow rates necessary to transport fuel in its present location. It is assumed that all fuel which would normally relocate due to water transport has already been relocated to a tank, filter or other low velocity location. Additional measurements of the WDS System are not considered to be justified based on ALARA considerations and the small quantities of fuel measured to date. After draindown of the RCS, the WDS System will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

APPENDIX D

BALANCE OF SYSTEMS CONTAINING SNM INVENTORY

1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel remaining in selected support systems which with a single exception, contain insignificant quantities of fuel; the Spent Fuel Pool A (FH109) is estimated to contain 3.8 kg of UO_2. The boundaries of this analysis are illustrated in Figures 14, 15, 16, 17, 18 and 19 which show the primary flowpaths between areas containing system components and the process piping in the AFHB. This appendix addresses the following support systems:

Defueling Water Cleanup System (DWCS)
Gas Radwaste Disposal System (WDG)
Nuclear Sampling System (SNS)
Spent Fuel Cooling System (SFC)
Submerged Demineralizer System (SDS)

All statistical uncertainties are expressed as ± one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the individual support systems and the part each system played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area measured in the AFHB. Only three (3) areas out of the total of seventeen (17) areas addressed in this appendix were formally measured using a series of gross gamma directional measurements. The balance of the areas were estimated based on their exposure and service history.

Section 4, "Analysis", explains how the estimate of record of fuel in those five (5) systems addressed in this appendix was determined and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the five (5) systems and the supporting rationale leading to the conclusion that the estimate is reasonable based on the available data and analysis performed.

2.0 BACKGROUND

Three of the support systems (DWCS, SDS and SNS) were constructed after the accident as part of the TMI-2 Cleanup Program. The other two (2) systems (WDG and SFC) were originally installed when TMI-2 was constructed.

The original primary function of the WDG System was to collect potentially radioactive gas from components and tanks in the plant, compress and deliver this gas to the Waste Gas Decay Tank, store the gas for decay, and recycle or release the gas through the unit vent at a controlled rate within the limits of 10CFR20. The WDG System continued to serve the same function during the cleanup program.

The SFC System was designed to remove the decay heat generated by the spent fuel stored within the fuel storage pools and to purify the water in the transfer canal and fuel storage pools. Due to the short operation of

TMI-2, no spent fuel was stored. The fuel pools were converted and used to store fuel canisters and ion exchange vessels during the cleanup period. Portions of the SFC System were modified during the cleanup program and used for process piping in the DWC System to remove radioactive material from the reactor vessel and fuel storage pools. The DWC System was used primarily to remove organic carbon, radioactive ions, and particulate matter from the fuel transfer canal (FTC), spent fuel pool (SFP) "A", and the reactor vessel (RV). The majority of the particulate matter was removed by processing the water through 0.5 or 16 micron filter canisters. Removal of the radioactive ions (i.e., soluble fission products) and/or organic carbon was accomplished by processing a portion of the filter effluent through 4x4 liners containing zeolite or charcoal materials that were similar to those used for EPICOR II.

The DWC System was composed of two (2) major subsystems, the RV cleanup system and the FTC/SFP cleanup system. Only a small portion of the RV cleanup system was located in Fuel Handling Building and that portion, the ion exchangers, was used on a very infrequent basis. The entire FTC/SFP cleanup system was located in the Fuel Handling Building, elevation 347'. The primary function of the two (2) subsystems was to remove suspended solids down to the nominal 0.5 micron rating to maintain the clarity of the water. Sample analysis data of RCS particulates indicated that approximately 1% of the sample was 0.5 microns or less (Reference 20). Since a filter-aid material was used to improve the filtering efficiency of the main filters, it was estimated that substantially less than 1% of the particulates passed through the main filters for possible deposit down stream from the filters. Most of the effluent from the main filters was returned to its source, RV or FTC/SFP "A". Up to 15 GPM was directed to the ion exchangers to remove fission products and/or organic carbon. Two (2) additional filters were installed on each side of the ion exchangers to remove any particulates greater than 0.45 microns that might possibly have entered the ion exchanger loop. All of the filter canisters have been removed from the TMI site.

The SDS System was a temporary liquid radwaste processing system installed in the TMI-2 Spent Fuel Pool "B" and the area immediately adjacent to the spent fuel pool. The system was designed to remove the fission products contained in the Reactor Containment Building and sump, the RCS and other liquid systems such as WDL, etc. by the process of ion exchange. The SDS System consisted of a liquid waste processing system, an off gas system, a monitoring and sampling system, and a solid waste handling system. The liquid waste processing system decontaminated the RCS water by filtration and demineralization. The off gas system collected, filtered and adsorbed radioactive gases produced during processing, sample dewatering, and spent SDS liner venting. The solid waste handling system also provided for moving, dewatering, storing and loading of filter and demineralizer vessels into the shipping cask. SDS performed the notable task of decontaminating the Reactor Building Sump water (625,000 gallons of water generated during the accident), the RCS (90,000 gallons), the FTC and water accumulated in the RCBT's or MWHT in combination with EPICOR II. The SDS System has been permanently shutdown although final flushing/draining has not yet been accomplished.

The SNS System was installed after the 1979 TMI-2 accident to provide a means to obtain representative liquid and gas samples from selected points containing post-accident waste. The sampling system provided a location for remote sampling of the sources of radioactive water within TMI-2. This eliminated the necessity of running highly radioactive sample water to the TMI-1 Sample Room and made the sampling operation independent of TMI-1 operations. The SNS System provided a means for recirculation and purging of the sample lines, backflushing with demineralized water to reduce activity levels, and transferring the sampling waste back to the storage system. The temporary sample hood for TMI-2 was located in the north end of the FHB at elevation 305'. This location permitted connection to the existing sampling lines running through the FHB. Samples from the individual subsystems could be collected by in-line sample containers or taken as grab samples. In addition, a boronometer was provided to monitor the boron concentration in the RCS. A separate loop, with its own pump and tank, was provided for calibration of the boronometer. The sample hood and room were ventilated by a filtered ventilation system which withdrew air from the hood/room and exhausted it to the Auxiliary Building ventilation system. A sample sink was provided to collect spillage from all operations conducted in the hood and also collected the liquid used to flush the sample lines from the hood isolation valves to the sample collection point. All sample waste was transferred to either the MWHT (WDL-T-2) or the RCHTs 'B' or 'C' (WDL-T-1B or 1C).

3.0 METHODS

Only three (3) of the seventeen (17) areas containing SNS, SDS, WDG, DWCS and SFC System components and piping were formally measured to assay the residual fuel. The three (3) areas, FH105, FH106 and FH109, were measured using a series of gross gamma exposure rate measurements in conjunction with a tungsten-shielded directional probe. Details of the specific procedures used for all three (3) areas is contained in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report and in each of the engineering calculations listed in Table 7. The remaining fourteen (14) unmeasured areas were analyzed based on their service history and their similarity to other areas that were measured. Details of these appraisals are discussed in Section 4, Analysis, of this appendix.

4.0 ANALYSIS

The results of the three (3) measured areas, FH105, FH106 and FH109, are shown in Table 7. Over 99% of the remaining fuel in the seventeen (17) areas addressed in this appendix is contained in FH109, Spent Fuel Pool A (SFPA). Most of this fuel was transported to SFPA from the RV as debris adherent to the outside of the fuel bearing canisters. The SFPA was used to store and to flush the outsides of fuel canisters prior to loading into the shipping casks. The balance of the Spent Fuel Cooling (SFC) System areas (AX118, AX119, AX120 and FH111) contain an insignificant quantity of fuel because the system was never exposed to the RCS primary cooling water and was not operated during the cleanup program at TMI-2. The only SFC System areas that were utilized during the cleanup program were the Spent Fuel Pool A (FH109), Spent Fuel Pool B (FH110), and a portion of the fill/drain piping for the FTC and pools.

SFPA was measured using gamma spectroscopy, and the results indicate that 3.8 kg of fuel (UO_2) was still deposited in SFPA. The other pool, SFPB, was used to house SDS vessels and remained clean through the entire recovery program. The water in the SFPB has been processed through EPICOR II and/or SDS and subsequently was not recontaminated. SDS vessels installed in SFPB were stored in secondary containment enclosures to collect any leakage from the connections to the vessels. Pool water was continuously drawn through these enclosures and passed through separate ion exchangers. This design prevented the pool water from becoming contaminated. The leakage was monitored through the established SDS sampling system. A residual fuel value of 0.003 kg (rounded to 0.00) was arbitrarily assigned to SFPB (FH110) to cover any potential film deposits that might remain in the connecting hoses which are stored in the FH110. FH111 is another relatively small storage pool which communicated with

FH110 but did not house equipment processing radioactive water or possible fuel. Since FH111 is clean and was not used to store containers which contained core debris, it was concluded that there was no residual fuel in FH111. FH302 is an area directly above the SFPB where SDS vessels were located and contained the supporting equipment for operating the SDS System. Numerous hoses and instrumentation are located in FH302, and the hoses probably have developed thin film deposits of fuel. A residual value of 0.001 kg UO_2 (rounded to 0.00) was assigned to FH302. The other areas in the Spent Fuel Cooling System were the Spent Fuel Cooler Area (AX118), the Spent Fuel Demineralizer Area (AX119), and the Spent Fuel Filter Area (AX120). They were never operated during or after the accident and therefore, were not internally exposed to core debris. AX118 served a second function during the cleanup program which was to store and package radioactive waste material. AX118 was periodically contaminated and decontaminated to prevent high level buildup.

The Gaseous Radwaste Disposal (WDG) System occupies a major portion of the following areas: AX113, AX125, AX126 and AX127. These four (4) areas contained the Waste Gas Filter (WDG-F-1), two (2) Waste Gas Decay Tanks (WDG-T-1A/1B), and the Waste Gas Analyzer. The waste gas was filtered prior to entering the Waste Gas Analyzer and it was assumed that no fuel is in the analyzer itself. On the other hand, there is some chance that a small amount of fuel was blown into AX125, AX126 or AX127. On this basis, a residual fuel value of 0.001 kg UO_2 (rounded to 0.00) was assigned to each of the three areas.

The only areas remaining (FH205, FH303 and FH305) were utilized for portions of the Defueling Water Cleanup System (DWCS). As a consequence, it was assumed that a thin film of fuel residue was retained in the hoses and pipes used to process the RCS and FTC/SFPA water. An arbitrary value of 0.002 kg UO_2 was assigned to FH205 and 0.001 kg UO_2 was assigned to

FH303 and FH305. These values were in line with the sample line values measured in FH106.

This results in a fuel estimate of record of 3.8 kilograms of UO_2 for all of the systems listed in Table 7. The range of this estimate extends from 0.30 Kg to 5.09 Kg UO_2. See Appendix F for the statistical determination of the range of values. After draindown of the RCS, including the reactor vessel, the Gaseous Radwaste Disposal System, the Spent Fuel Cooling System, the temporary Nuclear Sampling System, the Defueling Water Cleanup System, and the Submerged Demineralizer System will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

5.0 CONCLUSION

The estimate of record of the amount of UO_2 remaining in the Balance of Systems Containing SNM Inventory, listed in Table 7, is 3.8 kg. See Appendix F for details of statistical data combination.

This estimate of record was derived from the measurement and analysis results summarized in Table 7. These areas are expected to remain static since most of the areas have been drained and isolated, and the remaining areas have been subjected to water flows on a regular basis. Consequently, all of the fuel that would move has already been relocated to a tank, filter or other low-velocity location. Additional measurements of those areas listed in Table 7 are not considered to be justifiable based on ALARA considerations and the small quantities of fuel (UO_2) measured to date. After draindown of the RCS, the areas will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

APPENDIX E

AFHB AREAS THAT CONTAIN NO SNM

1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel remaining in a miscellaneous group of areas which could contain quantities of fuel if their normal function had been misused. The following is a list of those miscellaneous areas.

Area	Description	Primary System
AX001	RB Emergency Booster Pumps	RB Emergency Cooling (RR)
AX109	NS Coolers and Pumps	Nuclear Service Closed Cooling Water (NS)
AX110	Intermediate Coolers	Interm. Closed Cooling Water (IC)
AX111	Interm. Cooling Pumps & Filters	Interm. Closed Cooling Water (IC)
AX212	DH Surge Tank	DH Closed Cooling Water (DC)
AX214	Decon Facility	Temporary Cleaning Facility
AX219	Atmospheric Monitors	Health Physics (HP)
AX220	Caustic Liquid Mixing Tanks	Chemical Addition (CA)
AX221	Caustic Liquid Mixing Tanks	Chemical Addition (CA)
AX402	Cooling Water Surge Tanks	IC, RB and NS Systems
FH005	MDH Vault Pumps	Misc. Decay Heat Removal (MDH)
FH006	DH Service Coolers	Decay Heat Closed Cooling Water (DH)
FH007	Access Area - North	Fuel Handling Building
FH103	Sample Room	Temporary Nuclear Sampling (SNS)
FH107	Trash Compactor	Temporary Storage Area

FH108	Truck Bay	Fuel Canister Loading
FH203	Off-gas Separator & Standpipe	Submerged Demin. System (SDS)
FH204	Tanks & Charging Pumps	Standby Pressure Control System
FH301	Fuel Handling Bridge	Fuel Handling System

All statistical uncertainties are expressed as ± one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background," describes the original design functions of all of the primary systems and the part each system played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods," is not applicable to this appendix because no areas were formally assayed; they were analyzed for fuel based on their vulnerability during the accident and service during the TMI-2 Cleanup Program.

Section 4, "Analysis," explains how the estimate of record of fuel for those areas addressed in this appendix was determined and discusses assumptions made and conclusions drawn.

Section 5, "Conclusion," presents the estimate of record and uncertainty for the amount of fuel remaining in these areas and the supporting rationale leading to the conclusion that the estimate is reasonable based on the analysis performed.

2.0 BACKGROUND

Six (6) of the support systems, SNS, SDS, SPC, MDH, temporary cleaning facility, and a modified fuel handling bridge, were constructed after the accident and installed as part of the TMI-2 Cleanup Program. The other six (6) systems, CA, DC, HP, IC, NS and RR, were originally installed when TMI-2 was constructed. A short discussion of the original function of each system follows.

The Chemical Addition (CA) System was designed to dissolve chemicals in demineralized water. Some of the chemicals mixed were boric acid, sulfuric acid, sodium thiosulfate, lithium hydroxide, and hydrazine. The only chemical used during the TMI-2 Cleanup Program was the boric acid to maintain the RCS boron level to protect against criticality.

The original function of the Decay Heat Closed Cooling Water (DC) System was to supply demineralized cooling water to the DH Removal Coolers and Pumps which supplied emergency cooling to the RCS and the core. The DC System remained operable to provide cooling of the DH Removal System pumps, motors and coolers in the event it ever became necessary to operate the DH Removal System.

The original function of the Intermediate Closed Cooling Water (IC) System was to provide cooling water to various equipment located inside the Reactor Building. Some of the equipment supplied cooling water were the letdown coolers, the reactor coolant pumps, the control rod drive mechanism, and the SG hot drain coolers. Since none of this equipment operated after the accident, there was no need to operate the IC System during the cleanup program. The system has remained shutdown and isolated for the last eight years.

The original function of the Nuclear Service Closed Cooling Water (NS) System was to provide cooling water to the following equipment: instrument air compressors, RC pumps and motors, waste gas compressors, spent fuel pools, RC evaporator, RB spray pumps, MU pumps, NS pumps and motors, and the RB emergency cooling booster pumps. Since most of this equipment was operating or available to operate in any emergency, the NS System continued to operate during all of the cleanup program.

The original function of the RB Emergency Cooling (RR) System was to provide cooling water for all nuclear-related and fuel handling requirements. The RR System also provided emergency cooling to the RB cooling coils via the RB Emergency Cooling Booster Pumps. No emergency cooling has been required since the TMI-2 accident.

The original objective of the Health Physics (HP) Program was to minimize the radiation exposure of station personnel and visitors. During the TMI-2 Cleanup Program, it was even more important because TMI-2 was severely contaminated during the 1979 accident. Initially, the airborne radioactivity monitoring system (atmospheric monitors) was comprised of nine (9) fixed channels located in AX219 and two (2) mobile channels, each consisting of a particulate, iodine and gas measured monitor in series. During the cleanup program, additional mobile monitors were utilized as necessary to prevent worker overexposure.

The Mini Decay Heat Removal (MDH) System was installed after the 1979 accident to provide a flow path for make-up water from the Borated Water Storage Tank (BWST) to the RCS. The MDH System was never operated but was held in abeyance for emergency situations requiring additional cooling of the RCS.

The Submerged Demineralizer System (SDS) was a temporary liquid radwaste processing system installed in the 328' and 347' elevations of the Fuel Handling Building. The system was designed to remove the fission products contained in the RB sump and other liquid radwaste systems by the process of ion exchange. SDS was used to decontaminate the RB sump water (625,000 gallons), the RCS (90,000 gallons), and the FTC in combination with EPICOR

II. The SDS has been disassembled and all of the ion exchange and filter vessels have been removed from the FHB.

The Temporary Nuclear Sampling (SNS) System was installed after the 1979 TMI-2 accident to collect liquid and gas samples from selected points. The SNS System provided a location for remote sampling of the sources of radioactive water within TMI-2. This eliminated the necessity of running highly radioactive sample water to the TMI-1 Sample Room and made the sampling function independent of TMI-1.

A temporary cleaning facility was constructed at the north end of the Auxiliary Building (AX214) to decon tools and radiation monitors. The facility was used to clean equipment that was slightly contaminated and remained relatively clean throughout its use.

The Fuel Handling Bridge had been removed from the FHB shortly after the accident. The Fuel Handling Bridge was modified to transport fuel canisters prior to reinstallation. A completely new canister handling trolley was added to the existing Fuel Handling Bridge. The new trolley included the canister transfer shield, grapple, grapple guiding tool, hoists, cable and hose reels, and a load cell with a digital readout. Shielding for the fuel canisters was provided by a fixed, shielded mast (i.e., canister transfer shield) attached to the trolley. A nine (9) foot high shield collar was located on the lower end of the canister transfer shield.

The Standby Pressure Control (SPC) System was a recovery system used primarily as a source of borated water for makeup to the RCS. The SPC System was normally isolated from the RCS except when it was injecting borated water into the RCS. During defueling operations with the reactor vessel head removed, the SPC System was used as a source of makeup water to the RCS or a supply of flush water to the DWCS Systems. The SPC System components were located in the new fuel storage cell (FH204) on the 331' elevation of the FHB.

3.0 METHODS

All nineteen (19) areas addressed in this appendix were analyzed based on their vulnerability during the TMI-2 accident and their service history during the TMI-2 Cleanup Program. Since none of the areas were assumed to contain fuel, no formal assay measurements were performed. Details of the individual analyses are discussed in Section 4, Analysis, of this appendix.

4.0 ANALYSIS

Since no areas were formally measured and it was assumed that no fuel was located in any of the areas, this section is a discussion of the history of each area and the reasons why it has been concluded that they do not contain fuel.

AX001 (RB Emergency Cooling Booster Pumps), AX109 (Nuclear Service Coolers and Pumps), AX110 (Intermediate Coolers), AX111 (Intermediate Cooling Pumps and Filters), AX212 (DH Surge Tank), AX402 (IC and NS Cooling Water Surge Tanks), and FH006 (DH Service Coolers) all contain components which are part of secondary cooling systems.

All of these components are separated from radioactivity by heat exchangers or cooling tubes which give positive protection against migrating fuel. Area survey data support the supposition there is no fuel in any of these areas, even though some of these areas contain various degrees of contamination.

AX220 and AX221 are areas set aside for mixing chemicals prior to injection into various process systems. The areas were designed to facilitate mixing of boric acid, sulfuric acid, sodium thiosulfate, lithium hydroxide, and hydrazine. During the cleanup program, only boric acid was mixed to keep the RCS water borated to approximately 5,000 ppm while defueling the reactor vessel. Both areas were slightly contaminated but were never exposed to fuel. All injection lines to fuel bearing systems were protected by double valve isolation and were checked every 24 hours.

AX219 contains the Atmospheric Radiation Monitors for the Station Vent, RB Purge and Vent, and the Hydrogen Purge. All of these monitors received contamination from time to time and had to be decontaminated; they were never directly exposed to fuel.

FH005 contained two MDH pumps which were never placed in service; operation was limited to testing of the system. These pumps were designed to inject borated water into the RCS but were never used. Their flow pathway remained isolated during the entire cleanup program and was never exposed to fuel.

FH203 contained the off-gas standpipe and separator of the SDS which were used intermittently to process vent gas from the ion exchanger vessels. The gas flow was filtered prior to being exhausted to the plant vent stack. These filters prevented any particulates from entering the off-gas standpipe and separator and prevented fuel from reaching FH203.

FH103 contains portions of the Nuclear Sampling System. It is possible that some samples contained fuel. The entire sample area was cleaned and decontaminated periodically to remove any fuel that may have collected in the area. Upon completion of analyses, all samples were transferred to either the MWHT or the RCBHT 'B' or 'C'. Therefore, any fuel that might have been located in FH103 would have eventually found its way to either the MWHT or the RCBHT 'B' or 'C'.

AX214 was a decon facility installed during the cleanup program to decon hand tools and radiation monitors. The facility was used to clean slightly contaminated equipment and remained relatively clean throughout its use. All contaminated cleaning materials were disposed through proper channels, and there was no chance of fuel remaining in the facility. Area surveys were performed periodically and showed very low radiation exposure rates, less than 0.2 mR/hr, which supports the conclusion that no fuel was deposited in AX214.

FH204 contained the SPC charging pumps and tanks which were used to supply borated water to the RCS and other flushing operations. The system was used intermittently to supply water to the DWCS Systems. When not supplying water at high pressure, the SPC System was isolated from the RCS by double valve isolation which was verified every 24 hours. Since this was a new system maintained at a positive pressure when not isolated, there was no chance that fuel could have entered FH204.

FH301 contained the fuel handling bridge which was used to move fuel canisters in the SFPA. Since the canisters were contaminated with fuel when they entered the SFPA, it was possible that fuel was deposited in the canister transfer shield. The transfer shield was stored in the SFPA, and any residual fuel remaining was measured as part of the total amount determined for the fuel pool (FH109).

All of these above areas have been evaluated for fuel content, and it has been concluded that it is improbable any remaining fuel not already analyzed and accounted for elsewhere, exists in any of those areas.

5.0 CONCLUSION

The estimate of record of the amount of UO_2 remaining in the nineteen (19) areas listed in Section 1.0, Introduction, is 0.0 kg. This estimate of record is based on the discussion outlined in Section 4.0, Analysis. Additional measurements of any of the areas discussed in this appendix are not considered to be justifiable based on ALARA considerations and the absence of measured fuel (UO_2) to date. After draindown of the RCS, the areas will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

APPENDIX F

CONSTRUCTION OF OVERALL ERROR

The total UO_2 deposited within the AFHB is expressed as a sum. The total fuel value, termed the estimate for record, is the sum of all quantities measured or deduced within the Auxiliary and Fuel Handling Building areas. However problems arise in the expression of the overall uncertainty for differently derived individual uncertainties.

Three cases exist for the present. Most of the uncertainty is expressed as the usual bi-directional one sigma uncertainty. Due to unmeasureably small quantities, several results were expressed as minimum detectable quantities at the 95% confidence interval. Finally one area has an unsymmetrical uncertainty expressed as one (1) sigma. The purpose of this appendix is to detail the method used to combine the three cases into an overall uncertainty.

The method employed here was borrowed from suggestions made by Dr. Tingey, Idaho State University as follows:

Consider a set of counting data x_i, $i = 1, 2, \ldots k$, which are intended to estimate fuel quantities Z_i, $i = 1, 2, \ldots k$. Assume the variance of Z_i, σ_i^2 is known.

Consider another set y_j, $j = 1, 2, \ldots \ell$ for which the fuel quantity say θ_j can only be less than some minimum detectable level m_j and greater than or equal to zero.

Consider

$$W = \sum_{i=1}^{k} x_i + \sum_{j=1}^{\ell} y_j \qquad (1)$$

The expected fuel value of W, say N, is

$$N = \sum_{i=1}^{k} Z_i + \sum_{j=1}^{\ell} \theta_j \qquad (2)$$

$$N + \left[\sum_{i=1}^{k} \sigma_i^2\right]^{1/2} \geq \mathbb{R} \geq N - \left[\sum_{i=1}^{k} \sigma_i^2\right]^{1/2} - \sum_{j=1}^{\ell} \theta_j \qquad (3)$$

Note that the limits are not symmetrical about

$$N$$

Also note that when combining these limits with others determined similarly, a partitioning over all measurements being combined should be made in accordance with the above and equation (3) applied to the total set, that is we cannot simply combine the upper limits or lower limits statistically.

where: x_i are the measured counting data

Z_i are the expected fuel values corresponding to the measured counting data

σ_i^2 is the variance of Z_i

y_j are the MDL counting data

θ_j are the fuel values which are known to lie between zero and θ_j

W is the sum of x_i's and y_j's

N is the expected fuel value of W

R is the range of the expected fuel values

UO_2 in Areas with Symmetrical Uncertainty

As described in the body of the report, fuel is deposited in three systems. They are the Makeup and Purification, MU&P, the Liquid Radwaste Disposal, WDL, and the Solid Radwaste Disposal, WDS, Systems. Results for the individual areas comprising the MU&P and the WDL systems are presented in Tables F1, and F2. Results for the Solid Waste Disposal System were taken from the body of the report since only normal symmetrical uncertainty was found.

Summary results are presented in two parts. Total symmetrical uncertainty is presented in Table F3, and the combined overall uncertainty is discussed later.

The first step is to define the total fuel quantity for values measured above background with symmetrical uncertainties. The combined error was defined, in quadrature, as the square root of the sum of squares of the individual uncertainties for each area. The MU&P and WDL system totals are also shown on Tables F1 and F2.

Total UO_2 in All Areas

Table F3 provides the total fuel quantity and uncertainty for all increments with symmetrical uncertainty. Two additional components must be included to derive the grand total and overall uncertainty. The first is the simple sum of all fuel values derived from quantities too small to be measured above background, MDL's. The last is the fuel value for FH109 of 3.80 kg with an associated asymmetric uncertainty of +34% and -92%.

The grand total, also the fuel quantity of record, was calculated by equation 2 as:

$$\text{Total} = 6.95 + 0.71 + 3.80 = 11.46 \text{ kg } UO_2$$

The overall range of values was calculated by a form of equation 3 modified to include the asymmetric FH109 uncertainties, the negative term on the left and the positive on the right as follows:

$$\text{low value} = (6.95 - 3.03) + 3.80*(1-.92) = 4.22 \text{ kg } UO_2$$

and:

$$\text{high value} = (6.95 + 3.03) + 3.80*(1+.34) + 0.71 = 15.78 \text{ kg } UO_2$$

The estimate for record is 11.5 kg UO_2 with an uncertainty of +38% and - 63%.

TABLE F-1

FUEL (UO_2) AND UNCERTAINTY CALCULATIONS FOR THE MU&P SYSTEM

Area	Fuel Quantity	Uncertainty	σ	σ^2
AX004	0.03 Kg	± 58%	0.0174	0.000303
AX005	0.00 Kg	-	-	-
AX114	1.06 Kg	± 44%	0.4664	0.217529
AX115	0.13 Kg	± 25%	0.0325	0.001056
AX116	0.31 Kg	± 95%	0.2945	0.086730
AX117	0.06 Kg	± 68%	0.0408	0.001665
FH001	0.46 Kg	± 85%	0.3910	0.152881
FH003a	0.01 Kg	± 100%	0.0100	0.000100
FH003b	0.10 Kg	± 100%	0.1000	0.010000
FH002 FH004 FH014	0.16 Kg	± 50%	0.0800	0.006400
FH101	0.32 Kg	± 71%	0.2272	0.051620
Misc.	0.17 Kg	± 95%	0.1615	0.026082
Totals	2.81 Kg			0.555104
			Square Root =	0.7450
			Percentage =	26.51 or 27%

The sum of the MDL values is 0.60 Kg.

The MU&P System total is 2.81 Kg ± 27% + 0.60 Kg.

TABLE F-2

FUEL (UO_2) AND UNCERTAINTY CALCULATIONS FOR THE WDL SYSTEM

Area	Fuel Quantity	Uncertainty	σ	σ^2
AX012	0.10 Kg	± 104%	0.1040	0.010816
AX020	3.50 Kg	± 83%	2.9050	8.439025
AX021	0.31 Kg	± 46%	0.1426	0.020335
AX024	0.02 Kg	± 36%	0.0072	0.000052
AX102, AX131, AX134	0.10 Kg	± 104%	0.1040	0.010816
AX128	0.01 Kg	± 95%	0.0095	0.000090
AX501	0.01 Kg	± 76%	0.0076	0.000058
AX502	0.01 Kg	± 59%	0.0059	0.000035
AX503	0.01 Kg	± 100%	0.0100	0.000100
AX504	0.01 Kg	± 92%	0.0092	0.000085
FH112	0.01 Kg	± 55%	0.0055	0.000030
Misc	0.04 Kg	± 95%	0.0380	0.001444
Totals	4.13 Kg			8.482886

Square Root = 2.9125
Percentage = 70.52 or 71%

The sum of the MDL values is 0.11 Kg.

The WDL System total is 4.13 Kg ± 71% + 0.11 Kg.

TABLE F-3

FUEL SUMMARY FOR AUXILIARY AND FUEL HANDLING BUILDING

Symmetrical Data:

System	Fuel Quantity	Uncertainty	σ	σ^2
MU&P	2.81 Kg	± 27%	0.7587	0.575626
WDL	4.13 Kg	± 71%	2.9323	8.598383
WDS	0.01 Kg	± 57%	0.0057	0.000032
Totals	6.95 Kg			9.174041
			Square Root =	3.0289
			Percentage =	43.58 or 44%

Asymmetrical Data:

Two (2) other components must be included to determine the total quantity of fuel remaining in the AFHB. Those are the asymmetric term and the MDL term. The asymmetric value for FH109 is 3.80 Kg UO_2 +34%, -92%. The MDL value is 0.71 Kg UO_2.