Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:
(717) 948-8400

September 22, 1989
4410-89-L-0097/0356P

US Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

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

Dear Sirs:

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 and describes the methodology utilized to determine the quantity of SNM in each case. The attached PDSRs transmit the post-defueling survey results for the Pressurizer, the Reactor Building basement, and a revision of the Letdown Cooler Room survey results originally provided in the September 30, 1988, submittal.

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

Sincerely,

M B Roche

M. B. Roche
Director, TMI-2

JJB/emf

A001
1/1

Attachments

cc: W. T. Russell - Regional Administrator, Region I
J. F. Stolz - Director, Plant Directorate I-4
L. H. Thonus - Project Manager, TMI Site
F. I. Young - Senior Resident Inspector, TMI

TMI-2 POST-DEFUELING SURVEY REPORT
FOR
THE PRESSURIZER

SUMMARY

The estimate of record of the amount of uranium dioxide (UO_2) remaining in the TMI-2 pressurizer after June 1988 defueling is 0.3 kg $\pm$ 52% (at one sigma, defined as one standard deviation), distributed as follows:

Gravel-like and fine material	0.267 kg
Film	0.001 kg
Total	$\sim$ 0.3 kg

A video-examination of the debris material at the bottom of the pressurizer was used to determine the volume of remaining residual core materials. A sample of this material was removed from the pressurizer for analysis. Neutron interrogation and gamma spectrometry methods were used to calculate the amount of UO_2 in the sample by direct comparisons with a uranium standard and standard Ce-144 and Eu-154 sources. The total amount of residual UO_2 remaining in the pressurizer was calculated by scaling the sample for the total solids observed.

The quantity of UO_2 remaining in the pressurizer is $< 0.04\%$ of the maximum allowable residual UO_2 inventory for the entire TMI-2 facility in Mode 2 (Reference 1).

TMI-2 POST-DEFUELING SURVEY REPORT
FOR
THE PRESSURIZER

1.0 INTRODUCTION

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

Section 2, "Background", describes the physical attributes, location, and intended functions of the pressurizer. Its relationship to the accident and subsequent cleanup activities is also discussed.

Section 3, "Methods", discusses the two methods used based on radioisotope fuel analogs and the more direct assay by neutron interrogation.

Section 4, "Analysis", explains how the estimate of record of the amount of fuel in the pressurizer was calculated, and discusses supporting data, assumptions made, and the assigned uncertainty.

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

2.0 BACKGROUND

The TMI-2 pressurizer was intended to establish and maintain reactor coolant system (RCS) pressure within prescribed limits, and to provide a steam surge chamber and a water reserve to accommodate RCS density changes during operation. It is located in the south end of the 'A'

D-ring next to the once-through steam generator 'A' (OTSG 'A'). The bottom of its lower head is at approximately elevation 312', with the top of its upper head at approximately elevation 354'.

The pressurizer is an approximately 13.0 m vertical, cylindrical vessel with upper and lower hemispherical heads. It has an inside diameter of approximately 2.1 m in its cylindrical portion, with a wall thickness of approximately 0.2 m. The upper and lower heads have a radius of approximately 1.1 m and a minimum wall thickness of approximately 0.1 m. The upper head has three nozzles which connect to the relief system, and one approximately 0.1 m schedule 120 nozzle which connects to the pressurizer spray line. It has an approximately 0.03 m schedule 160 vent nozzle and an approximately 0.4 m inside diameter manway opening. The center of the lower head has an approximately 0.3 m schedule 140 nozzle which connects to the surge line. A surge diffuser is connected to this nozzle, extending approximately 0.3 m into the lower head. The surge line nozzle connects the bottom of the pressurizer to the 'A' RCS loop hot leg. Vent safety and relief valve lines are attached to the top of the pressurizer and connect to the reactor coolant drain tank (RCDT) through a drain line. Three removable 480-V AC electric heaters are located at the bottom of the pressurizer, approximately 1.5 m above the base. They are designed to replace heat lost during normal operation and to raise the pressure to normal operating pressure (approximately 14.4 MPa) during RCS heatup. The heaters also function to return the system pressure to normal following transients. The pressurizer spray provides an introduction of relatively cool water from the RCS 'A' loop cold leg to pressurizer steam space, which condenses some of the steam to reduce RCS pressure.

The boundaries of this report encompass the pressurizer, the electric heaters, and the internal ladders of the pressurizer. The location of the pressurizer in relation to other RCS components is shown in Figure 1. Figure 2 illustrates the position of the relief, block, and safety valves in relation to the pressurizer. Figure 3 illustrates the pressurizer as defined for this report, which excludes all of the pressurizer's incoming and outgoing pipes and valves.

During the TMI-2 accident, the pressurizer water level was raised and lowered to thermally control the reactor pressure. Core material was relocated to the pressurizer, most probably as a result of this process. This postulate is supported by the existence of a trace of core material in the RCDT and in the Reactor Building basement.

The general method used to quantify residual UO_2 in the pressurizer was to estimate the volume of residual core debris after defueling and to obtain and analyze a sample of the material. The results of the sample analysis were used to infer the total UO_2 content of the remaining material. Two independent analytical methods were used due to the relatively large errors associated with the methods. The analytical methods are based on radioisotope fuel analogs and the more direct assay by neutron interrogation.

Radiological Environment

Various valves and joints located on the top of the pressurizer leaked intensely radioactive RCS coolant during the accident. As a result, the general exposure rates near the surfaces of the pressurizer are between 0.5 and 3 R/hr, with a frequent value of approximately 1 R/hr. The 0.2 to 0.3 R/hr exposure rate inside the pressurizer is significantly less due to lower film activity on the stainless steel liner. The external carbon steel surfaces are more chemically active and able to sequester more RCS activity in a tightly adherent film normally termed fixed activity.

A loosely adherent film also exists leading to easy transfer of contamination to personnel and equipment. Typical levels of approximately 1 $\mu Ci/cm^2$ (2.0 E+06 dpm/cm^2) on and near the pressurizer require substantial radiation protection controls that further complicate detailed assay programs.

As a result of this hostile environment, it is not considered ALARA to perform additional measurements to reduce the uncertainties below the values quoted herein.

3.0 METHODS

Sample Collection

In March of 1988, technicians used Mini-Rover to collect approximately 1 liter of core debris from the pressurizer. From this amount, a 100 g sample was taken consisting of one large consolidated mass, three smaller masses, and two scoops of fine material.

Defueling

In June of 1988, a robotic submersible device called Mini-Rover was used to remove most of the debris (13.5 liters) at the bottom of the pressurizer. Calculations showed that an estimated volume of 0.9 liters $\pm$ 35% of sediment remains in the pressurizer (Reference 3). Due to the inaccessible, underwater location of the core debris, the estimated volume was calculated by examination of videotapes recorded via the Mini-Rover camera system after the June 1988 defueling of the pressurizer. Almost all of this amount was at the bottom of the pressurizer. Since the surge line nozzle is not flush with the bottom of the pressurizer, the sediment collected around this nozzle (Figure 3). Very small quantities of sediment were also deposited on heater assemblies and in shallow pockets at the base of each heater bundle. The material at the bottom of the pressurizer looked very light and was moved easily by Mini-Rover thrusters. From observations of videotape, it was determined that the small crust-like material that formed on the heater bundles was due to corrosion and therefore discounted.

Pressurizer dimensions and geometry, and the depth of the sediment (gauged by comparison to parts of the standpipe diffuser nozzle and the scoop that was installed on the Mini-Rover for sediment collection) were used to calculate the volume of the sediment. The average depth of sediment around the pipe was estimated to be 1.6 cm, tapering to 1.0 cm (Reference 3).

Gamma Spectroscopy

A gamma spectroscopy analysis was performed on the 100 g sample by using a calibrated HPGe spectroscopy system (Reference 4). Gamma spectroscopy was used because it is a passive, non-destructive method of assaying fuel. Gamma radiation associated with the decay of fission products can be used to indicate the presence of fuel. The signal strength of a specific energy peak in a gamma spectrum corresponding to a radioisotope that is chemically similar to fuel can be used to determine the amount of fuel. A measurement of that isotope activity should be directly proportional to the amount of fuel present, which is expressed as UO_2. Fuel sample examinations and code calculations at TMI-2 support the use of Ce-144 and Eu-154 as fuel tracers (Reference 5). Ce-144 and Eu-154 have been shown to be an acceptable analog for transuranic material. The oxides of uranium, cerium, and europium have similiar water solubility, therefore the Ce-144 and Eu-154 in the reactor fuel matrices were transported through the reactor coolant system during the accident.

The fuel value corresponding to the film layer on the interior surfaces of the pressurizer was determined using the total inside surface area of the pressurizer, the pressurizer ladders, the heater bundles, and the average fuel value determined by measurements on the pressurizer manway cover (Reference 6).

Active Neutron Interrogation

In April and May of 1989, an antimony-beryllium (Sb-Be) photo-neutron (γ, n) source was used to conduct a series of active neutron interrogation measurements.

The interrogating neutrons are generated by inserting a Sb-124 source into the Be cylinder where neutrons are produced via Be (γ, n) reaction. These neutrons are thermalized by a polyethylene $[(CH_2)_n]$ cylinder which covers the Be cylinder. Thermalized neutrons from this source system were impinged on the normal UO_3 standard or TMI-2 residual fuel samples to initiate fission in U-235 contained in UO_3 or

samples. Fast fission neutrons emitted during the fission process were detected by a He-4 detector that is sensitive to fast neutrons. A schematic diagram of the top and side view of this system are shown in Figures 4 and 5, and the details of the work can be found in Reference 7.

The validity of the standard to unknown comparison method depends on small or similiar neutron absorption for both. This was determined to be the case in Reference 8.

4.0 ANALYSIS

Gamma Spectroscopy Analysis of Samples

The isotopic activity recorded during the spectroscopy analysis of the 100 g sample taken in March of 1988 is shown in Table 1. Measurements of the Eu-154 activity were uncertain by approximately $\pm$ 8.7%, whereas the uncertainty of the Ce-144 based activity was $\pm$ 41.3%. The errors quoted are based on counting efficiency, calibration source activity, and geometric uncertainties.

Using Ce-144/fuel and Eu-154/fuel ratios (82.8 µCi/g $\pm$ 53% and 40.4 µCi/g $\pm$ 86% UO_2, respectively, obtained from more than 100 in-vessel and ex-vessel samples as of April 4, 1988), the amount of UO_2 in the sample was calculated to be 0.0051 kg $\pm$ 67% for the Ce-144/fuel ratio and 0.007 kg $\pm$ 86% for the Eu-154/fuel ratio (Reference 3). The uncertainties quoted are $\pm$ one sigma values of the respective data set and not of the means of the sets.

Active Neutron Interrogation Method

The active neutron interrogation method required three measurements:

1. Background for the system was determined while the Sb-Be source was active, in absence of any samples or a normal UO_3 standard.

2. A normal UO_3 standard (0.5 kg UO_3) was placed in the measurement location (see Figures 4 and 5) and measurements were repeated.

3. The pressurizer sample was placed in the same location as the two above.

Results of background measurements were subtracted from the results of samples and UO_3 standard measurements, and then background-subtracted results of samples and UO_3 were compared to one another.

Since the U-235 content of the UO_3 standard (0.711% by weight) and average U-235 enrichment of TMI-2 fuel (2.5% by weight) are known, residual fuel content of samples can be estimated by ratioing the results of samples and standard measurements. These series of neutron measurements resulted in a UO_2 estimate of 0.0097 kg $\pm$ 48% in the pressurizer sample. The U-235 content of both the standard and the sample allowed lower comparisons to be made.

The weighted average of the Ce-144, Eu-154, and neutron based determinations, resulted in the final UO_2 sample value of 0.0067 kg $\pm$ 38% for the 0.1 kg sample (Reference 3). The fractional UO_2 content of the sample was therefore 0.067 kg/kg.

Final Analysis

The average bulk density of 4.4 g/cm^3 (Reference 3) corresponding to TMI-2 core debris was adopted for the remaining material within the pressurizer. The amount of UO_2 in the pressurizer, excluding the fuel value corresponding to the film layer on the interior surfaces of the pressurizer, is estimated to be approximately 0.267 kg of UO_2 as follows:

Volume x Density of material x Weight fraction of reactor fuel in the sediment = Amount of UO_2 in pressurizer (Reference 3).
908 cm^3 x 4.4 g/cm^3 x 0.067 = 0.267 kg.

Using the total inside surface area of the pressurizer (8.12 E+5 cm^2), pressurizer ladders (1.32 E+4 cm^2), and heater bundles (4.27 E+5 cm^2) (Reference 3), and using the average UO_2 value (8.2 E-7 g/cm^2 ± 51%) determined by measurements on the pressurizer manway cover (Reference 4), the fuel value corresponding to the film layer on the interior surfaces of the pressurizer is estimated to be 0.001 kg ± 51% UO_2 (Reference 3).

The uncertainty is partly determined from a volume estimation of the residual debris at the bottom of the pressurizer. Five engineers examined the videotape of the pressurizer and estimated the amount of material remaining. The estimate of record is the average of these estimates. The volume uncertainty is derived from the range of these estimates (Reference 3) and is ± 35%.

Analysis Conclusion

The estimate of record of 0.3 kg has an overall uncertainty of ± 52% (Reference 3), which was calculated as the counting, and volume uncertainties taken in quadrature.

The goal of the current defueling program is to remove more than 99% of the original core inventory of approximately 94,000 kg. In that context, the 0.3 kg quantity of UO_2 remaining in the pressurizer is $<0.04\%$ of the maximum allowable residual UO_2 inventory for the entire TMI-2 facility in Mode 2 (Reference 1).

5.0 CONCLUSION

Using all available information, the estimate of record of the amount of fuel in the pressurizer after June 1988 defueling is 0.3 kg ± 52% (at one sigma).

The pressurizer, as it currently exists, is considered to be isolated. For an extended period of time in the past, approximately 400 liters/minute of water was continually added to the top of the pressurizer and discharged from the bottom. The purpose of this flow was

to clarify the RCS water in the pressurizer. Video-examinations during that period showed that there was no significant change in the quantity of UO_2 due to that flow. Therefore, the amount of UO_2 remaining in the pressurizer is considered to be stable, and is not anticipated to change during subsequent system draindown at the completion of defueling, as long as the flow rate is approximately 400 liters/minute or less. If the draindown rate is significantly greater, it may be necessary to reinspect the pressurizer and associated surge line to ensure that material did not relocate.

REFERENCES

1. R. E. Rogan. Residual Fuel Basis. GPU Nuclear Internal Memorandum 4410-88-M-0306/4400-88-078. Middletown, PA: GPU Nuclear Corporation. 24 August 1988.

2. J.S. Schork. TMI-2 Core and Special Nuclear Accountability Program. Administrative Procedure 4000-ADM-4420.02, Rev. 3-01. Middletown, PA: GPU Nuclear Corporation. 10 October 1988.

3. M.H. Haghighi. Calculation of the Pressurizer Fuel Value. 4550-3220-88-016, Rev. 2. Middletown, PA: GPU Nuclear Corporation. 19 July 1989.

4. R.E. Lancaster. Calculation of the Pressurizer Sample. 4550-3223-88-011, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 7 April 1988.

5. P.J. Babel. Ce-144, Eu-154 and Eu-155 as Tracers for Fuel Debris. TB-86-41, Rev. 2. Middletown, PA: GPU Nuclear Corporation. 26 February 1988.

6. J. Greenborg. Deposition of Fuel on the Inside Surfaces of the RCS. TB-86-37, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 4 September 1986.

7. M. H. Haghighi. Fuel Estimate of OTSG and Pressurizer by Active Neutron Method. 4800-3221-89-037, Rev. 1. Middletown, PA: GPU Nuclear Corporation. 19 July 1989.

8. M. H. Haghighi. Comparison of UO_3 Standard and Pressurizer Sample Total Absorption Cross-Section. 4800-3321-89-079, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 25 August 1989.

TABLE 1

RESULTS OF SAMPLE FROM THE PRESSURIZER*

Isotope	Activity (μCi)
Cs-137	36,653 ± 5.2%
Cs-134	490 ± 6.8%
Co-60	5,006 ± 5.7%
Sb-125	5,235 ± 6.3%
Ce-144	420 ± 41.3%
Eu-154	284 ± 8.7%
Ru-106	788 ± 8.6%

* References 3 and 4

FIGURE 1
TMI-2 PRESSURIZER
(IN RELATION TO OTHER RCS COMPONENTS)

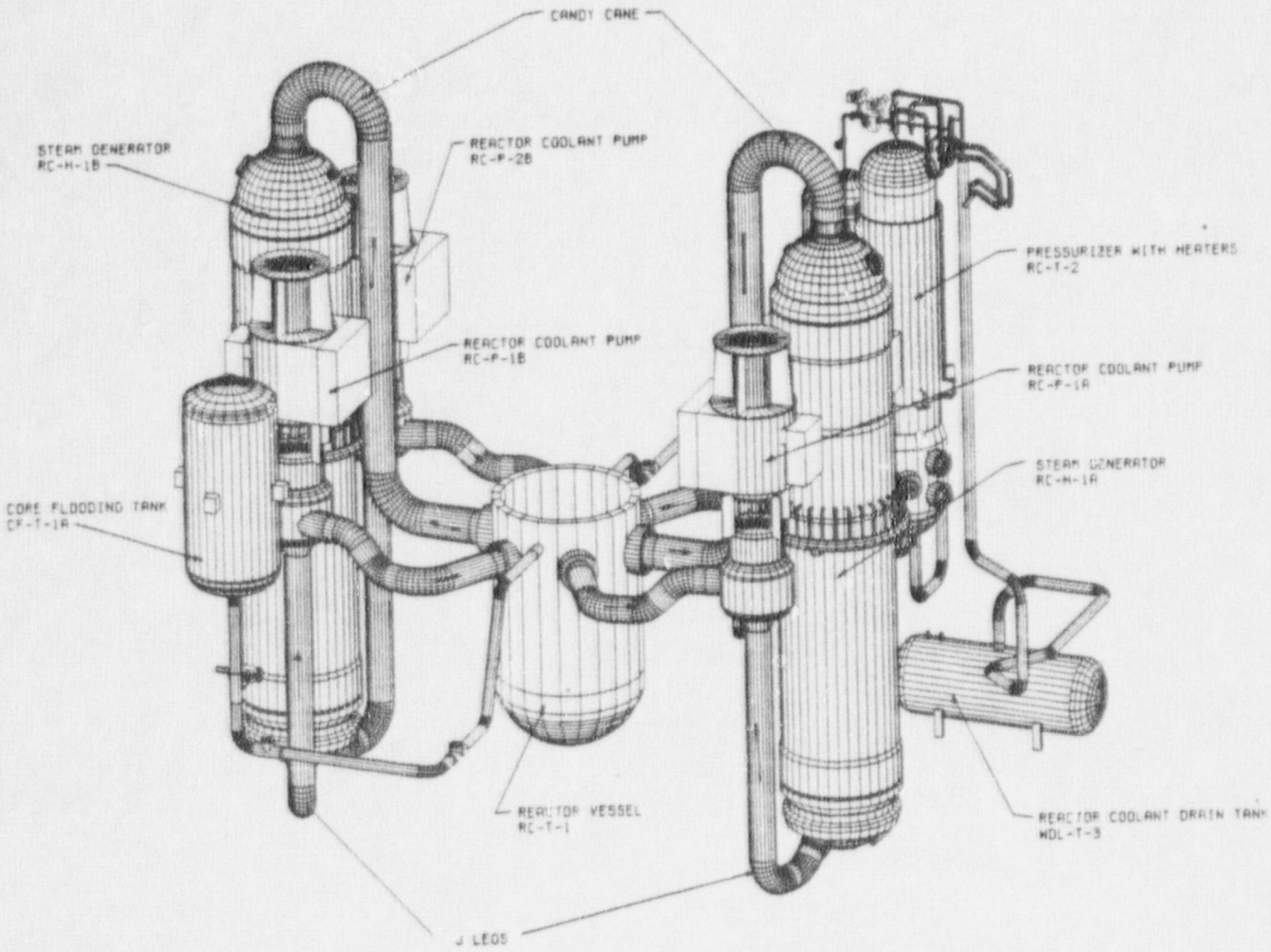


FIGURE 2
TMI-2 PRESSURIZER
(SHOWING VALVE LOCATIONS)

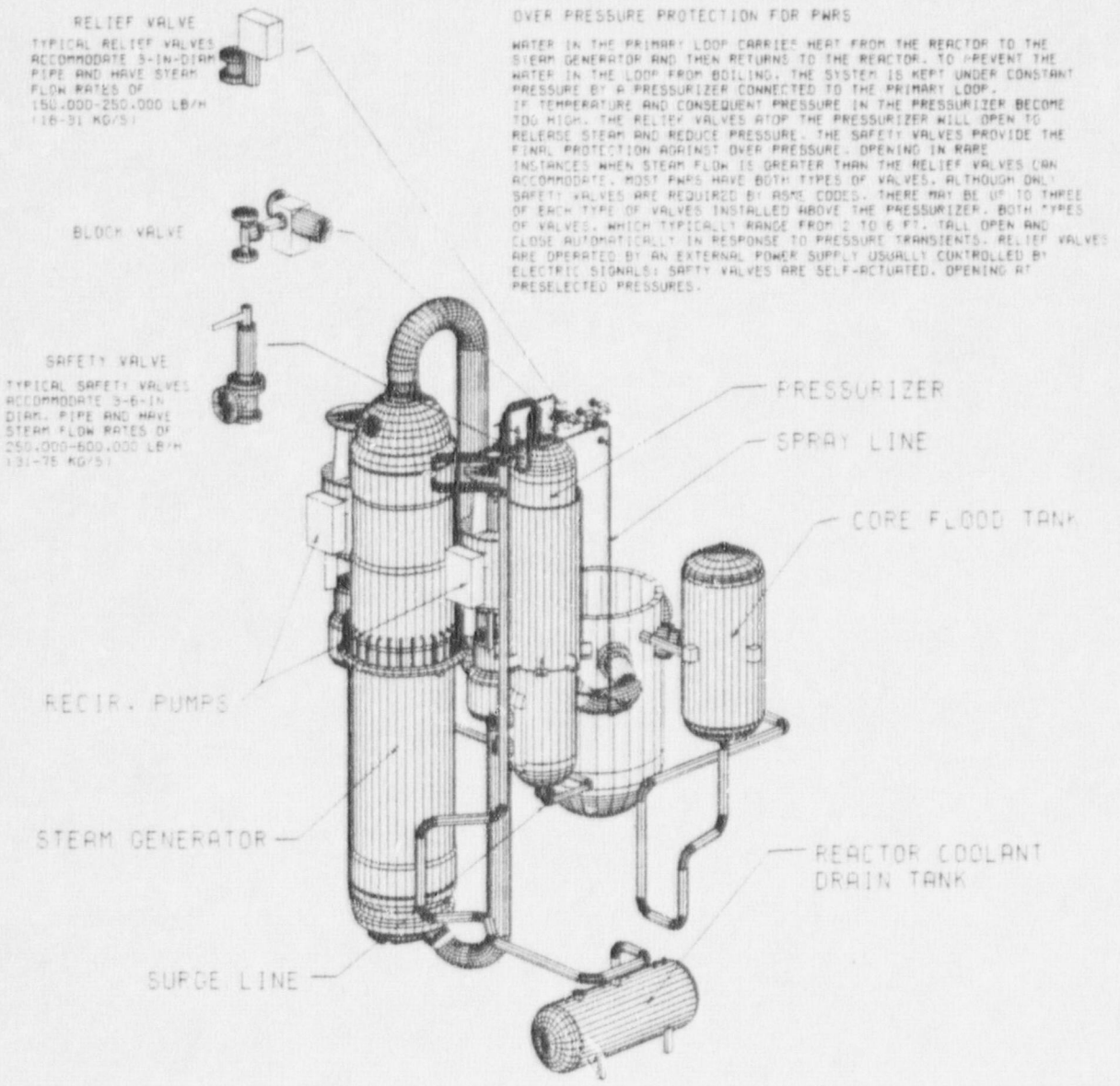


FIGURE 3
TMI-2 PRESSURIZER GEOMETRY AND DIMENSIONS

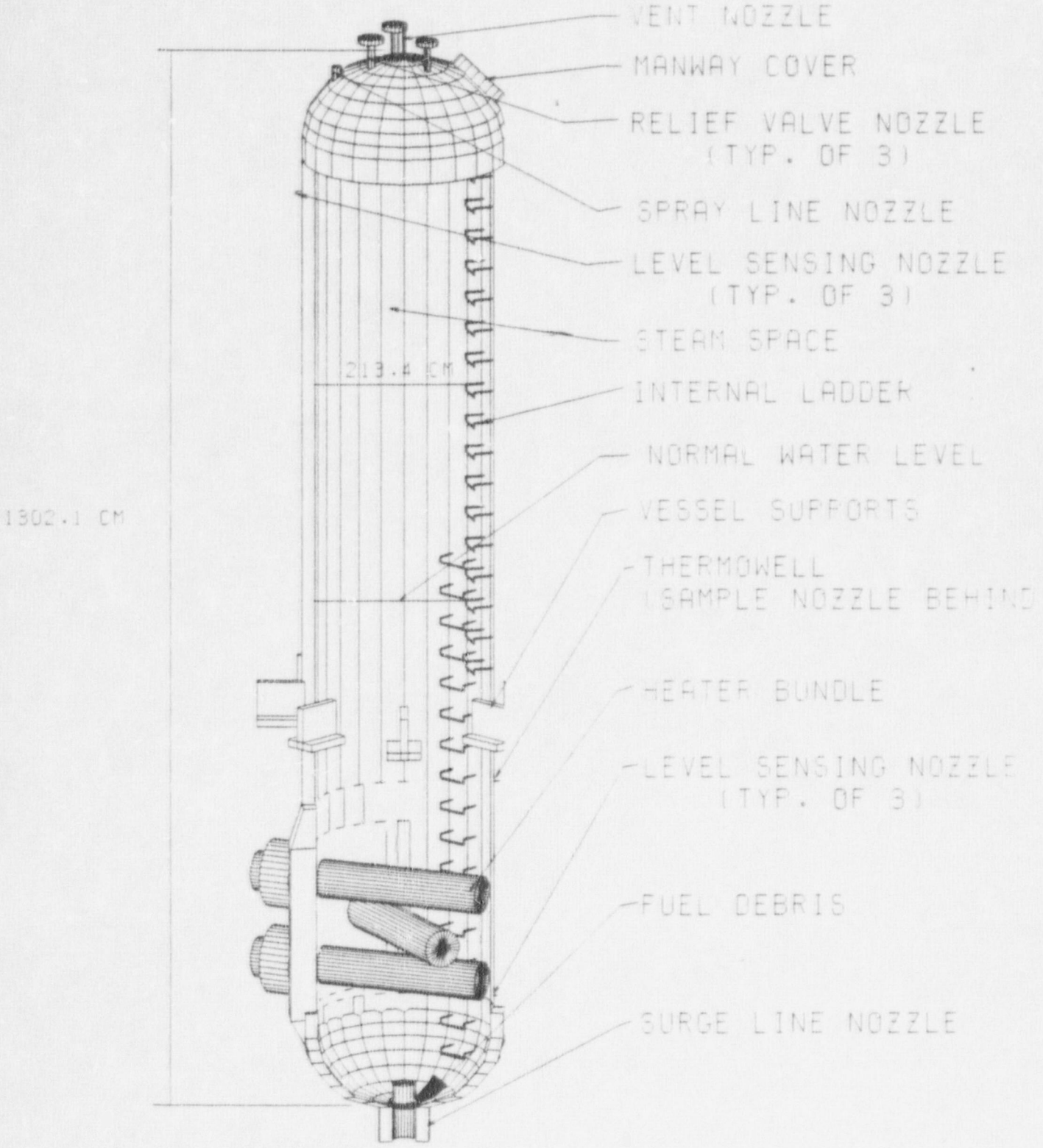
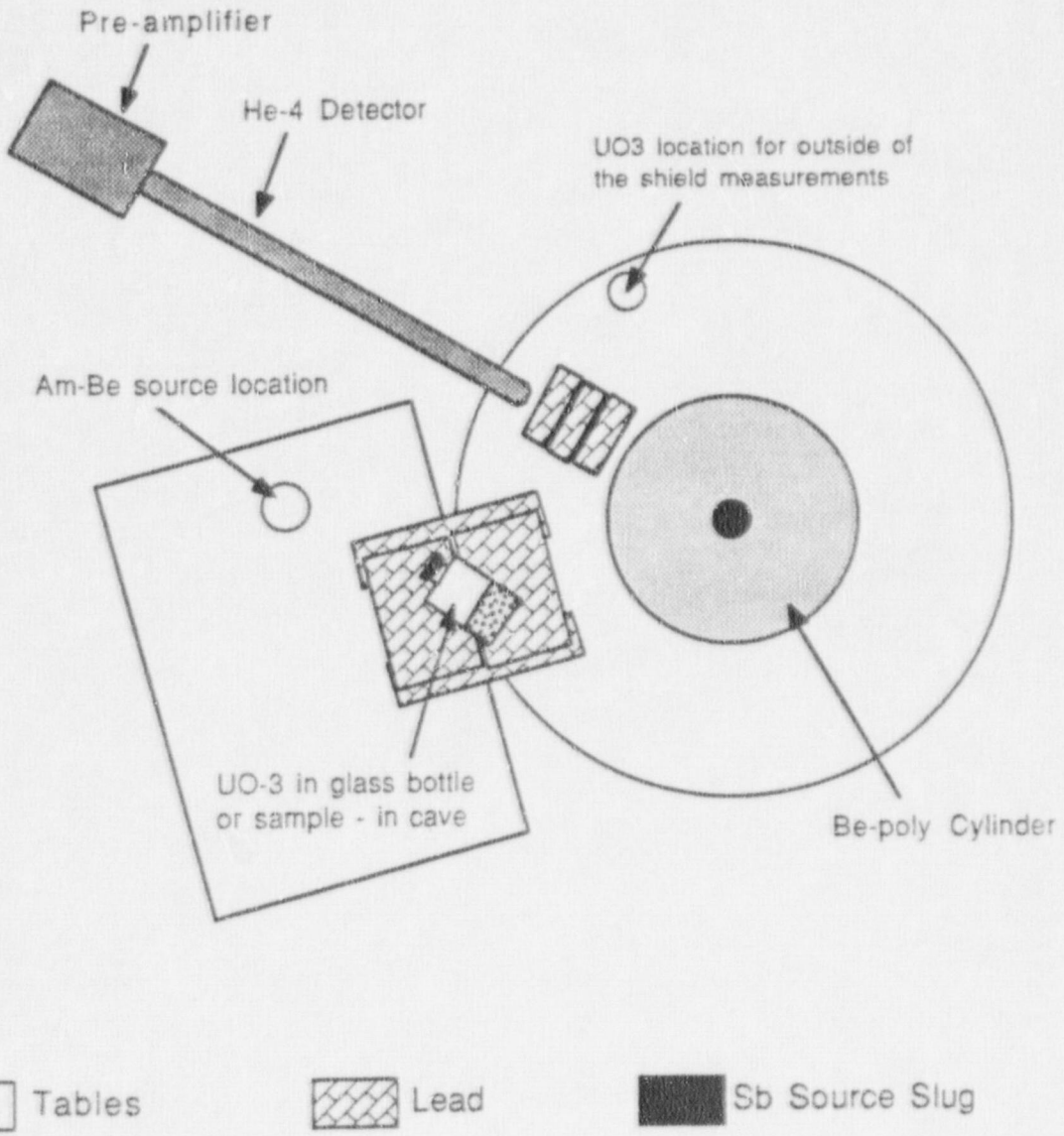


FIGURE 4

NEUTRON INTERROGATION SYSTEM IN AX130
(Top View)

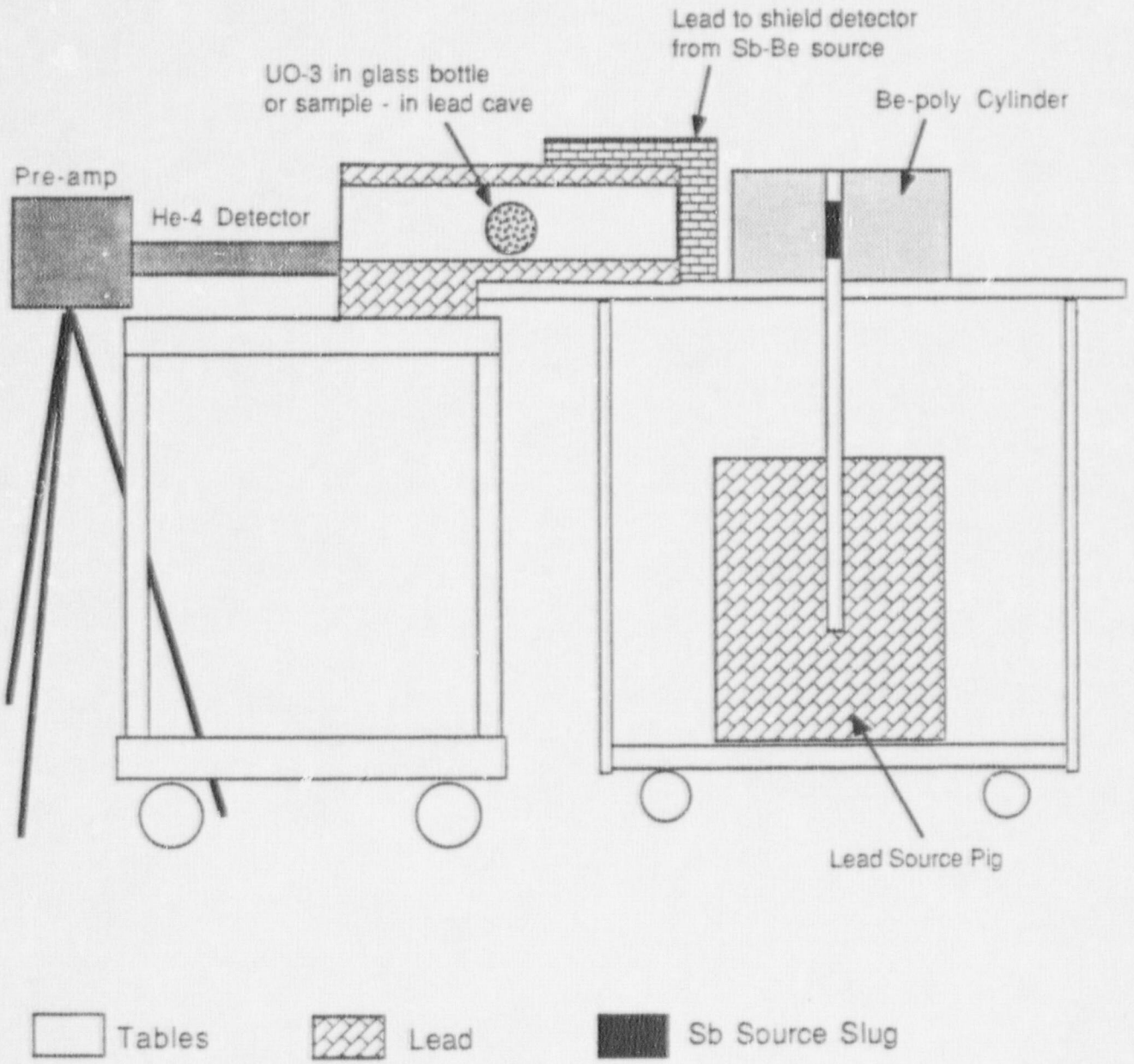


* Drawing is not to scale.

FIGURE 5

NEUTRON INTERROGATION SYSTEM IN AX130

(Side View)



* Drawing is not to scale.

TMI-2 POST-DEFUELING SURVEY REPORT
FOR
THE REACTOR BUILDING BASEMENT

SUMMARY

The estimate of record of the amount of uranium dioxide (UO_2) remaining in the Three Mile Island Unit 2 (TMI-2) Reactor Building (RB) basement is 1.3 kg $\pm$ 54% (at one sigma, defined as one standard deviation).

The distribution of the fuel is as follows:

Basement Sediment	1.3 kg $\pm$ 54%
Basement Water	< 0.0003 kg
Basement Walls	< 0.0001 kg
Total Estimate	1.3 kg $\pm$ 54%

The amount of UO_2 remaining in the RB basement sediment was developed using remote sampling and gamma spectroscopy.

The amount of UO_2 remaining in the RB basement water was estimated by the boildown of a sample and the analysis of the gross alpha emitter content by counting a planchet of the evaporator solids in an alpha scintillation counter.

The amount of UO_2 remaining in the RB basement walls was conservatively estimated because none of the core samples taken of the walls contained measurable Ce-144 or Eu-154, or inferred UO_2.

The 1.3 kg quantity of UO_2 remaining in the RB basement is < 0.2% of the maximum allowable residual UO_2 inventory for the entire TMI-2 facility in Mode 2 (Reference 1).

TMI-2 POST-DEFUELING SURVEY REPORT
FOR
THE REACTOR BUILDING BASEMENT

1.0 INTRODUCTION

This report presents the analysis of the Three Mile Island Unit 2 (TMI-2) Reactor Building (RB) basement uranium dioxide (UO_2) inventory. It is one in a series of reports generated to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 2). All statistical uncertainties are expressed as $\pm$ one sigma limits (defined as one standard deviation).

Section 2.0, "Background", describes the physical attributes of the RB basement and its relationship to the accident and subsequent cleanup activities. The boundaries of this report are also discussed.

Section 3.0, "Methods", describes how existing sample data were used to produce the estimate of record. Where gamma spectroscopy data were available, these data were used in lieu of calculated sample estimates. This is considered an overestimate, since much of the sediment volume estimation was done prior to desludging.

Section 4.0, "Analysis", explains how the estimate of record of fuel in the RB basement was calculated based on the sample analysis data which were collated for each source category (sediment, water, and walls).

Section 5.0, "Conclusion", states the estimate of record and uncertainty for the amount of UO_2 remaining in the RB basement, and supporting rationale leading to the conclusion that the estimate is reasonable based upon the available data and analysis performed.

2.0 BACKGROUND

The RB basement consists of the space between the floors of elevation 282'-6" and elevation 305' of the Reactor Building. Although the RB basement houses several major components, this PDSR does not include the amount of fuel retained in the various system components below the 305' elevation. These components will be discussed in other PDSRs. This PDSR does, however, include the entire floor area of the 282'-6" level, including sediment, water, and walls. The interior walls and floors are constructed primarily of poured reinforced concrete. The RB containment is poured reinforced concrete with a carbon steel liner.

During the TMI-2 accident, reactor coolant was discharged from the reactor coolant system (RCS) into the RB basement via the pressurizer pilot-operated relief valve (PORV) and the reactor coolant drain tank (RCDT). The discharged reactor coolant increased pressure in the RCDT causing its rupture. This vented the RCDT to the RB basement. The reactor coolant that was discharged into the RCDT and RB basement contained reactor fuel fines from the damaged reactor core. The locations of the RCDT and the discharge are shown in Figure 1.

The reactor coolant that was discharged into the RB sump became mixed with sediment-bearing river water, building spray water, and additional leakage from the RCS. The RB basement remained flooded with this water for approximately two years. During this time, the reactor fuel fines settled out onto the RB basement floor (Figure 2).

Radiological Environme

The Reactor Building basement was the repository of most of the RCS water released as a result of the March 1979 accident. The accident water remained until removed during the period starting September 1981 and ending March 1982. During this period, a fraction of the soluble fission products that contaminated the water were absorbed into porous surfaces. As a consequence, present area radiation levels vary from a low of 1-2 R/hr to a high of 50 to 100 R/hr. The general area exposure rate tends to be near 10 R/hr.

Significant loose surface contamination existed (>1 $\mu Ci/cm^2$) and contaminated water (1-3 $\mu Ci/ml$) was present to a depth of 15 to 20 cm on the RB basement floor as of September 1989. The methodology covered by this report is considered to be consistent with keeping radiation exposure as low as reasonably achievable (ALARA) due to the extremely hostile nature of the basement environment.

3.0 METHODS

The estimate of record of the amount of UO_2 remaining in the RB basement was determined by sample analysis of the RB basement sediment, the remaining RB water, and the walls of the RB basement and gamma spectroscopy data. Samples have been taken and analyzed for uranium content since 1979. The fuel analysis data were collated for each source category (sediment, water, and walls), and the data were then applied to the appropriate area and source category to arrive at the estimate of record.

For sediment analysis, remote sampling and gamma spectroscopy were performed. Initial remote samples (from 1979 to 1983) were acquired by lowering sample acquisition devices into the RB basement at various locations. These initial samples were suspect due to an insufficient mass of solid materials. When a robot became available, it was possible to obtain larger samples. Leachability, trace metal, radiochemical, particle size, and morphology analyses were performed (Reference 5). Gamma spectroscopy was used in the RCDT discharge area and in other locations that were identified as the most probable locations for reactor fuel. Gamma spectroscopy had the advantage of being able to remotely examine large areas of the RB basement.

In three locations in the basement (see Figure 3), gamma spectroscopy was performed with sodium iodide detectors (References 3 and 4). The sodium iodide detectors were lowered into the basement and were used to measure the 2190 KeV gamma ray signature of the praseodymium 144 daughter of cerium 144. Using these data, the fuel amounts in those areas were determined by back-calculating using a cerium-to-fuel ratio

(Reference 6). The results of these spectral measurements provide a measure of the total fuel content integrated over the field of view of the detector and have been used when appropriate.

For the water samples, most of the analyses involved the boildown of a sample of basement water and the analysis of the gross alpha emissions by counting a planchet of the evaporated solids in an alpha scintillation counter.

Gamma spectroscopy measurements of the RB basement concrete core did not show the presence of Ce-144, which is the analog for reactor fuel. The fuel content of the walls are assumed to be due to intrusion and plateout from the basement water. Since the quantity of dissolved UO_2 in the water is low, it can be assumed that the walls also contain minimal reactor fuel.

4.0 ANALYSIS

The following sections describe the analyses used for each of the major source categories.

4.1 RB Basement Sediment

The UO_2 content for the sediment has been estimated using two samples and one spectroscopy scan taken in the Reactor Building basement. Although many samples have been taken in the RB basement from August 28, 1979 to present (see Figure 3), most are suspect due to insufficient volumes to provide a representative sample. The largest of the samples was 0.7 g. A sample taken in the RCDT (see Figure 3) did not represent the basement floor and was rejected. Two samples taken from the impingement area (see Figures 1 and 2) had enough mass (10 to 20 g) to be used in the calculation. The uranium to sediment ratio was determined by averaging the results of the samples. The average of the analysis results was 3.7×10^{-6} g UO_2 per gram of sample (Reference 7).

The standard deviation associated with the sediment samples was approximately ± 81%. This UO_2/sediment ratio was used in all areas of the basement except the RCDT discharge area. In that area, the results of gamma-ray spectroscopy were used.

For the gamma-ray spectroscopy scans, a total of three measurements were made. Two measurements were eliminated due to the non-observance of a Ce-144 peak and a high minimum detectable level (MDL) value caused by high water levels in the basement. The gamma spectroscopy scan of the RCDT discharge area was used to characterize this area. The result was an estimate of 1.2 kg ± 58%. The uncertainty in the measurement is based on the counting error and the Ce-144 to reactor fuel ratio.

The amount of sediment, previously estimated from video-inspections of the floor area (Reference 7), was reduced by the quantity of sediment removed by the desludging operations (Reference 8) to derive an estimate of the remaining sediment volume. The sediment mass is given in Reference 7 and summarized in Table 1 of this report.

The next step was to multiply the total grams of sediment in each area by the UO_2/sediment ratio and add the quantity of reactor fuel for the RCDT discharge area. Table 2 shows the estimated final area tallies. The result was an estimate of 0.05 kg ± 80% of UO_2 in the sediment that was located outside of the gamma scanned area. To this amount was added the 1.2 kg ± 58% of UO_2 identified from gamma scan conducted at the RCDT discharge area. Addition of these terms yields a total of 1.3 kg ± 54% of UO_2.

Desludging operations to remove the sediment removed approximately 0.2 kg of UO_2 and 4890 kg of sediment, leaving 9710 kg of sediment in the basement (Reference 7). Therefore, the quantity of reactor fuel on the floor was 1.1 kg ± 64%. The sediment represents the major source term in the RB basement, since the dense insoluble fuel fragments and fines are assumed to have remained with the sediment.

This estimate does not account for UO_2 that has been deposited in the drainage system. However, most portions of the drainage system have been flushed by waste water from the gross decon experiment and other work via the floor drain risers from the upper elevations. Assuming that a portion of the drainage piping not subjected to this flow has approximately 0.3 cm of sediment distributed along its bottom, there was 7.1×10^3 g of sediment in the piping (Reference 8), equating to 0.03 g of UO_2 using the impingement sediment fuel concentration of 3.7 E-6 grams of UO_2 per gram of sediment. The drain contribution to the sediment term was, therefore, insignificant.

In addition to the sediment in the drain system, reactor fuel may have been deposited in part of the drain system due to the decontamination of reactor defueling tools at the 347' elevation in the tool decontamination facility. The quantity of reactor fuel was estimated from the number of tools decontaminated (approximately 500), the surface activity of the tools (30 µg UO_2/cm^2 of micron size particles), and an average surface area. It was assumed that all surface activity on the tools was washed into the sump. Larger fuel particles may have been transferred to the building drain piping that was connected to the tool decontamination sink. This drain will be assayed for fuel and reported upon separately. The result of this analysis is that there is, conservatively, 0.2 kg of fuel in the floor drain system/RB sump from tool washing (Reference 8). Therefore, the estimate of record of the amount of UO_2 in the RB basement from the sediment and the drain system is 1.3 kg $\pm$ 54%.

The amount of fuel contained in the RB basement is expected to remain static, since it is not anticipated that further reactor coolant will be deposited on the floor. This estimate also assumes that future desludging operations will not remove additional material.

4.2 RB Basement Water

The contribution of UO_2 from the water presently in the RB basement is insignificant in comparison to the sediment estimate. To determine this estimate, the results of 17 basement water samples taken in 1988 have been retrieved from the GPU Nuclear Laboratory Data System (NLDS) database for use in this estimate. Initially, the samples were not analyzed for the purpose of quantifying the UO_2 content. However, the boildown residues were counted for 20 minutes in a SAC-4 alpha scintillation counter to obtain a gross alpha analysis. Based on the average water level in the RB basement, as reported by Plant Operations on the Plan-of-the-Day printouts from May 1 to May 24, 1988, there were approximately 15 cm of standing water in the RB basement, equating to approximately 1.5×10^8 ml of liquid. At an alpha activity of $< 5.3 \times 10^{-6}$ µCi/ml (the mean MDL activity), there were $< 7.88 \times 10^2$ µCi (Reference 10) of the calculated alpha activity in the basement water. Using the ratios of alpha-emitting isotopes present in the reactor fuel and the alpha activity of the water, < 0.3 g of UO_2 was estimated to be in the water, which indicates that negligible fuel quantities were present in the RB basement water.

The estimate of record of the combined total for the sediment and water therefore remains to be 1.3 kg $\pm$ 54% of UO_2.

4.3 RB Basement Walls

Gamma spectroscopy measurements of the concrete cores of the RB basement concrete did not show the presence of Ce-144, which is the analog for reactor fuel. The fuel content of the walls was assumed to be due to intrusion or plateout from the basement water. As shown above, the assay of the RB basement water yields negligible UO_2 content. Since the quantity in the water is low, it can be assumed that the walls also contain minimal fuel due to the low probability that particulate fuel would seep into the concrete.

The quantity of UO_2 in the concrete was calculated from the gross alpha activity in the RB basement water and the volume of water that diffused into the concrete. From numerous concrete core bores of the various concrete walls and the floor, it is known that the radioactive material penetrated approximately 5 mm into the concrete walls and floors and completely through the block wall. The volume fraction of water absorbed into the concrete block and poured concrete is 0.21 and 0.05, respectively. The total amount of affected concrete is 6.3×10^6 cm^3 of concrete block and 8.8×10^6 cm^3 of poured concrete. The activity is therefore the product of the volume of affected concrete, the fraction of absorbed water, and the gross alpha activity. The gross alpha activity is related to the quantity of UO_2 as described in section 4.2 of this report. If it is assumed that the walls hold a quantity of UO_2 equal to the maximum concentration in the water, the result is an additional 0.003 g of UO_2.

The estimate of record of the combined total for the sediment, water, and walls therefore remains to be 1.3 kg $\pm$ 54% of UO_2.

The goal of the current defueling program is to remove more than 99% of the original core inventory of approximately 94,000 kg. In that context, the 1.3 kg quantity of UO_2 remaining in the RB basement is $<0.2\%$ of the maximum allowable residual UO_2 inventory for the entire TMI-2 facility in Mode 2 (Reference 1).

5.0 CONCLUSION

The estimate of record of the amount of UO_2 remaining in the RB basement is 1.3 kg $\pm$ 54% (at one sigma). This uncertainty is based on the square root of the sums of the squares of the individual uncertainties associated with each component of the analysis.

This estimate of record is derived from existing sample analysis results. It is expected to remain static since it is not expected that

additional quantities of water from the RCS system will be added to the RB basement. The UO_2 content is not expected to change significantly if the basement is dewatered, but may be reduced if a more efficient method of desludging can be employed.

REFERENCES

1. R. E. Rogan. Residual Fuel Basis. GPU Nuclear Internal Memorandum 4410-88-M-0306/4400-88-078. Middletown, PA: GPU Nuclear Corporation. 24 August 1988.

2. TMI-2 Division. TMI-2 Core and Special Nuclear Material Accountability. Administrative Procedure 4000-ADM-4420.02, Rev. 3.01. Middletown, PA: GPU Nuclear Corporation. 10 October 1988.

3. P.J. Babel. Reactor Building Basement Fuel Estimate. TB 85-08, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 11 March 1985.

4. P.J. Babel. Reactor Building Fuel Estimate (A Supplement to TB 85-08, Rev. 0). TB 85-08, Rev. 1. Middletown, PA: GPU Nuclear Corporation. 13 May 1986.

5. K. L. Wright. Radioanalytical Report. Rockville, MD: Science Applications International Corporation. 11 August 1986.

6. P.J. Babel. Cerium 144 as a Tracer for Fuel Debris. TB 86-41, Revs. 0 - 2. Middletown, PA: GPU Nuclear Corporation. 10 October 1986, 15 September 1987, and 29 February 1988.

7. P.J. Babel. Reactor Building Basement PDSR Reactor Fuel. GPUNC Calculation 4530-3220-89-013, Rev. 3. Middletown, PA: GPU Nuclear Corporation. 19 July 1989.

8. D.E. Ferguson. Reactor Building Basement Sludge. GPUNC Memorandum 4550-87-0218. Middletown, PA: GPU Nuclear Corporation. 18 November 1987.

9. R. E. Lancaster. Estimate of the Amount of Fuel Transferred to the RB Sump. GPU Nuclear Calculation 4800-3220-89-013. Middletown, PA: GPU Nuclear Corporation. 16 May 1989.

10. D. E. Ferguson. Reactor Building Basement Water SNM. GPU Nuclear Calculation 4530-3220-88-018, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 15 July 1988.

TABLE 1

MASS INVENTORY SHEET .R REACTOR BUILDING
ELEVATION 282'-6" SEDIMENT*

AREA OF BASEMENT	ESTIMATED ORIGINAL SEDIMENT MASS (g)
Enclosed Stairwell	5.05 E+6
OTSG Leakage Clr	3.44 E+5
Impingement Area	6.85 E+6
Northwest Quadrant	1.47 E+5
Inside of D-Ring	9.37 E+5
Leakage Coolers	1.26 E+5
RC Drain Tank	1.23 E+5
RCDT Discharge	1.45 E+5
RB Sump Area and Letdown Coolers	6.95 E+5
Incore Cbl Chase	1.45 E+5
STARTING MASS:	1.46 E+7
REMOVED MASS:	4.89 E+6
REMAINING MASS:	9.71 E+6

* Reference 8.

TABLE 2

ESTIMATE OF RECORD OF REACTOR FUEL IN SEDIMENT

Area Of Basement	Reactor Fuel (kg of UO_2)*
Stairwell	0.0187
OTSG Leakage Clr	0.0013
Impingement Area	0.0250
Northwest Quadrant	0.0005
Inside of D-Ring	0.0035
Leakage Coolers	0.0005
RC Drain Tank	0.0005
RCDT Discharge	1.2000**
RB Sump Area and Letdown Coolers	0.0026
Incore Cbl Chase	0.0005
Initial UO_2 Material	1.253
Removed UO_2 Material	.176
Remaining Floor UO_2 Material	1.077

* Based on fuel content of 3.7 E-6 grams of UO_2 per gram of sediment and total estimated mass of sediment.

** Fuel content is based on direct gamma spectroscopy measurements.

FIGURE 1

REACTOR BUILDING BASEMENT FLOOR PLAN

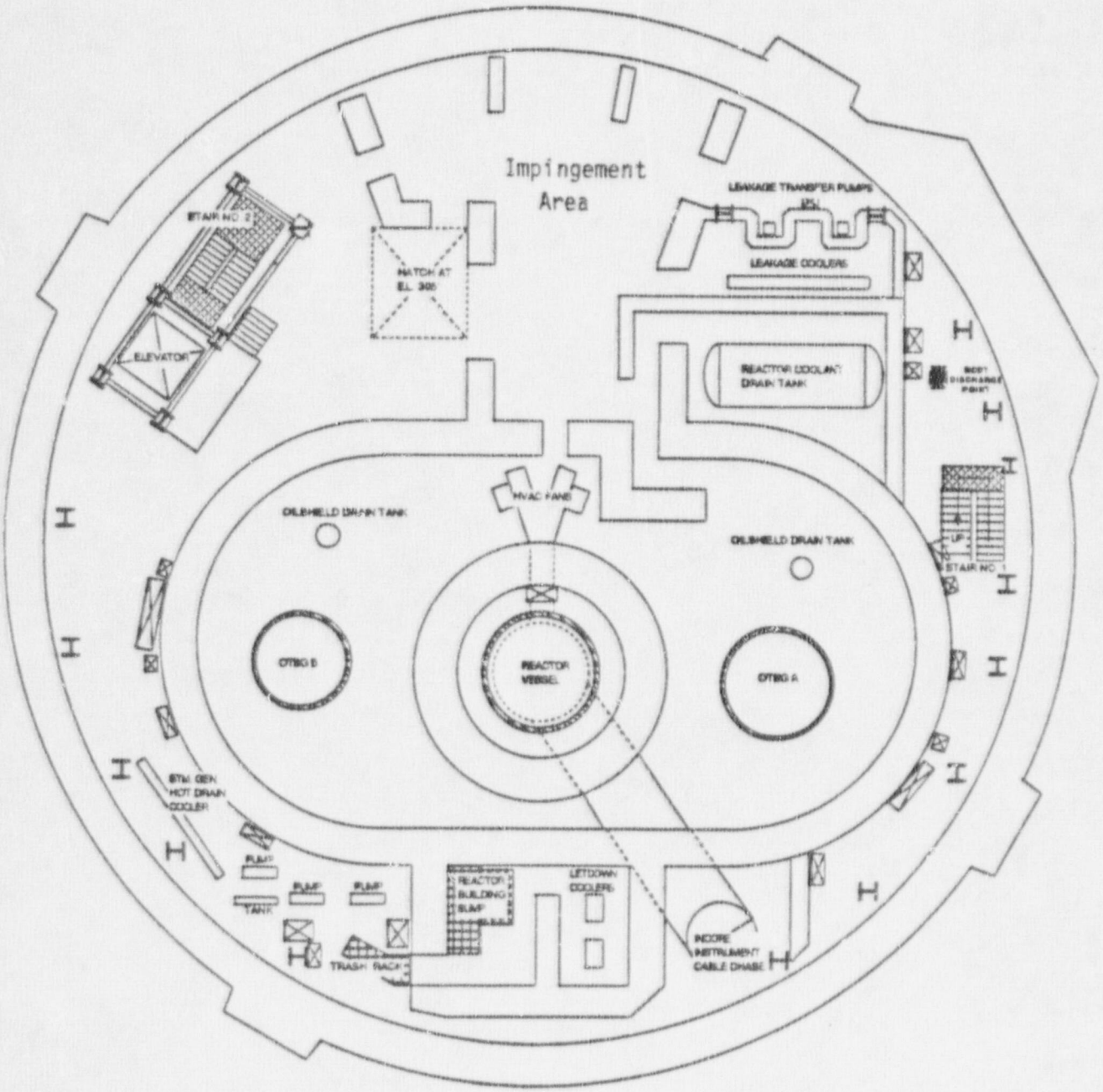


FIGURE 2

REACTOR BUILDING BASEMENT FLOOR

(View Looking North)

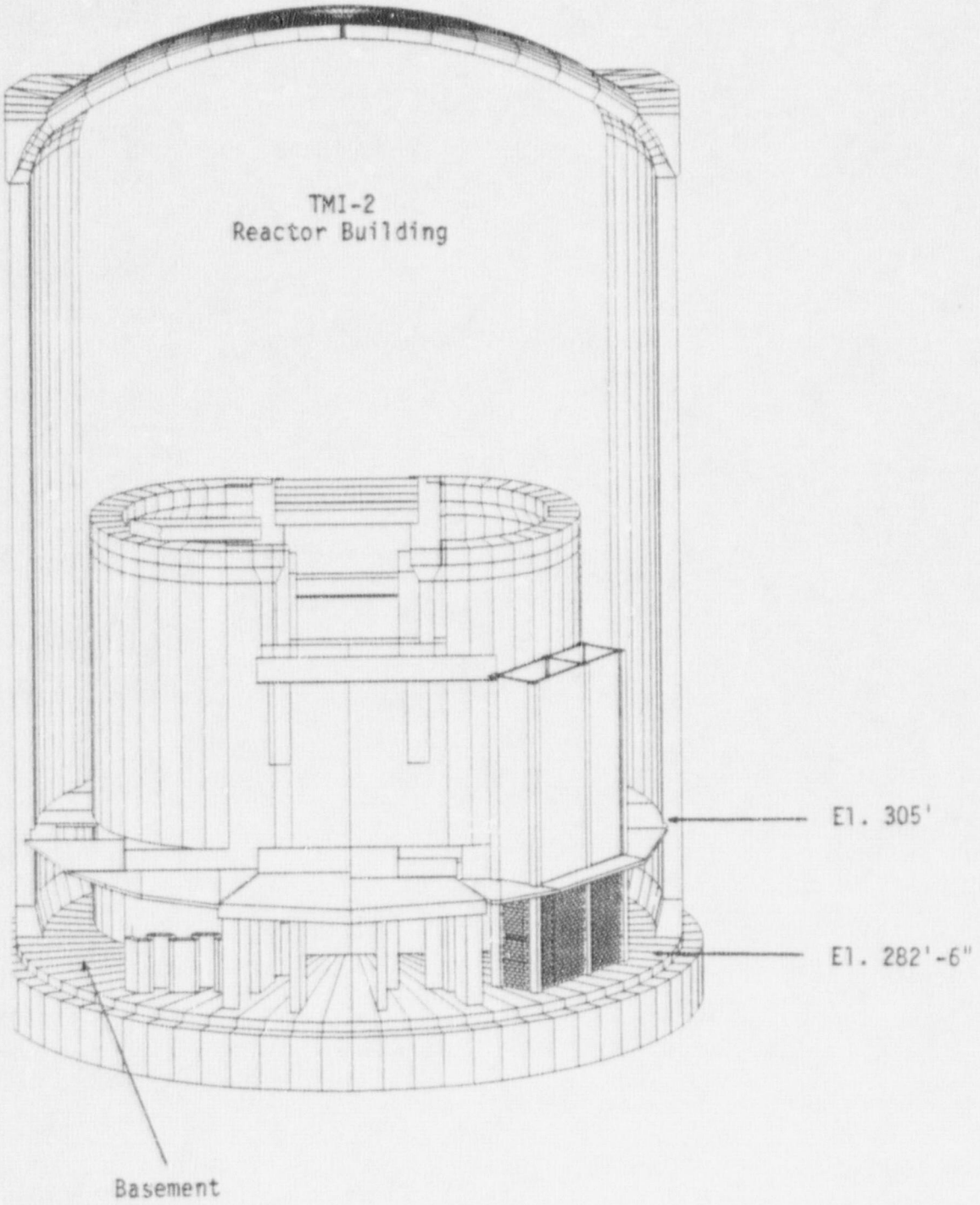
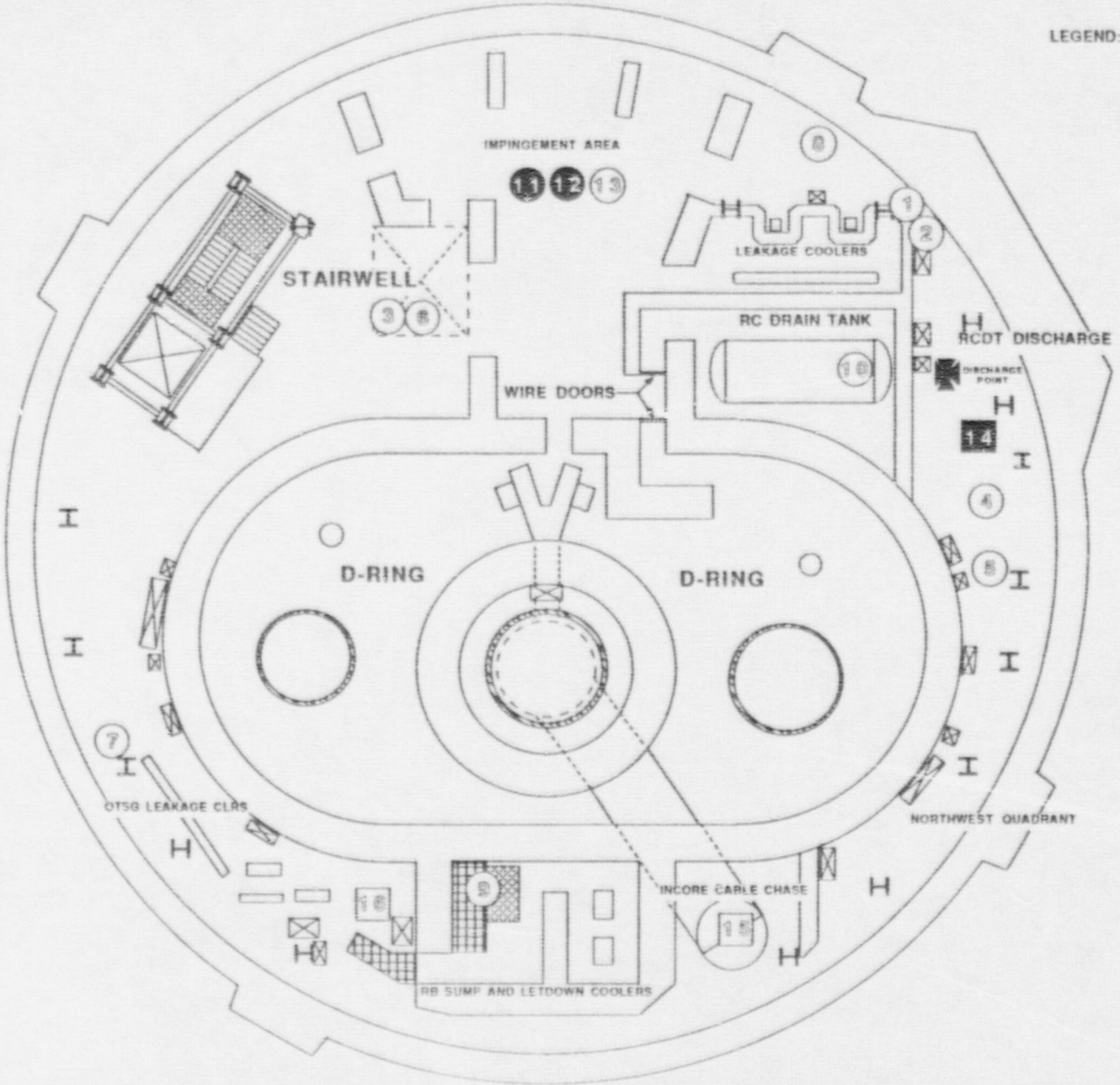


FIGURE 3

SLUDGE SAMPLE LOCATIONS



LEGEND: BLACK-FILLED SAMPLE LOCATIONS USED IN ANALYSIS; WHITE-FILLED LOCATIONS WERE NOT.

1. PENETRATION 401 8/28/79
2. PENETRATION 401 11/15/79
3. COVERED HATCH 5/14/81
4. BOTTOM OF OPEN STAIRWELL 9/24/81
5. BOTTOM OF OPEN STAIRWELL 6/23/82
6. COVERED HATCH 1/11/83
7. PENETRATION 238 1/11/83
8. PENETRATION 225 1/11/83
9. RB SUMP PUMP DISCHARGE LINE 8/22/83
10. REACTOR COOLANT DRAIN TANK 12/5,12/83
11. IMPINGEMENT AREA NUMBER 1 5/16/86
12. IMPINGEMENT AREA NUMBER 2 5/16/86
13. IMPINGEMENT AREA NUMBER 3 5/16/86
14. GAMMA SCAN NO. 1 EARLY 1985
15. GAMMA SCAN NO. 2 4/12/85
16. GAMMA SCAN NO. 3 4/22/85

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE LETDOWN COOLER ROOM

(REVISION 1)

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE LETDOWN COOLER ROOM

(REVISION 1)

REVISION SUMMARY

The TMI-2 Post-Defueling Survey Report (PDSR) for the letdown cooler room has been revised due to revisions in the analyses that formed its basis.

The estimate of record of the amount of uranium dioxide (UO_2) reactor fuel remaining in the letdown coolers and piping in the letdown cooler room has been increased from a minimum detectable level (MDL) value of $\leq$ 3.2 kg to an MDL value of $\leq$ 3.7 kg with an uncertainty of approximately $\pm$ 53% (at one sigma, defined as one standard deviation) for the following reasons:

1. Revision in the determination of the total number of gamma-ray events occurring in the region of interest (ROI) for cerium 144 (Ce-144) for the detector efficiency and the letdown cooler room measurement.

2. Recalculation of the detector efficiency.

3. Revision in the calculations for the calculated gamma-ray fluence rate from the letdown coolers to the detector.

4. Revision in the selection of the location of the reactor fuel within the letdown coolers for the estimate of record value.

Revision in the Number of Gamma-ray Events in the ROI for Ce-144

The revision in the number of gamma-ray events in the ROI for Ce-144 resulted from changing the method of determination. For the detector efficiency, the

original PDSR method to determine the number of counts in the ROI for Ce-144 was taken to be those counts in 48 channels above the 2.19 MeV peak (half-peak analysis). In Revision 1 of the PDSR, the ROI was determined from the full width at half maximum (FWHM). The original PDSR method used the number of counts in the same ROI channels where the Ce-144 peak should have occurred. In Revision 1 of this PDSR, the number of counts in the ROI channels corresponding to the FWHM were used. The net effect of these changes was to provide a 60% increase in the detector counts during efficiency determination and a 2% increase in the the number of measured events.

Revised Detector Efficiency

Three parameters were adjusted to provide a more accurate intrinsic detector efficiency. The parameters were the standard Ce-144 source activity, the method of peak integration, and the thickness of the lead shield. The overall effect increased the intrinsic detector efficiency by a factor of 1.8. Details can be found in Reference 8.

Revision of the Reactor Fuel Calculation

Three changes in the calculation of the gamma-ray fluence rate from the assumed letdown cooler sources to the detector were made. First, the letdown coolers' source region was found to have been incorrectly located and a mathematical error was found in the calculated amount of the reactor fuel. Second, the original calculation was based on preliminary source term data that were the best available data at the time of the measurement. This revision uses the most current data. These changes offset the effect of an increase in the detector efficiency described above. Revision 1 of the PDSR corrects this. Finally, a revision was made in the method of calculating the MDL value to account for all possible errors.

Revision of Reactor Fuel Location

In the original calculation, reactor fuel was assumed to be located uniformly in the bottom quadrant of the coolers. This model is now believed to be too conservative. The current model assumes that the reactor fuel is uniformly located in the inlet pipes in the coolers and the initial tube spiral in the lower bottom quadrant.

Letdown Flow Rate

The original calculations reported in the original PDSR showed the letdown flow rate to be 1.6 liters per second. However, a detailed review of plant logs showed that, in September 1986, the block orifice was removed, which resulted in an increase in the flow rate to a maximum of 3.0 liters per second.

This, however, did not affect the analysis. Additional analysis has been performed to show the flow rates through various sections of piping and the letdown coolers. This also does not change the results of the analysis.

Conclusion

These changes result in an increase in the estimate of record from an MDL value of $\leq$ 3.2 kg to an MDL value of $\leq$ 3.7 kg with an uncertainty of approximately $\pm$ 53% (at one sigma).

SUMMARY

The estimate of record of the amount of uranium dioxide (UO_2) reactor fuel remaining in the letdown coolers and piping in the letdown cooler room is ≤ 3.7 kg with an uncertainty of approximately $\pm$ 53% (at one sigma) based on the chosen model and associated minimum detectable level (MDL)* measurements and calculations. This amount includes reactor fuel in all forms. It is not possible, at this time, to determine the forms of the reactor fuel (e.g., films, loose debris) since the internals of the system are not accessible and the coolers reside in a radiologically hostile environment.

A collimated lead shielded sodium iodide detector was inserted into a letdown cooler room access penetration to measure Ce-144, which was used as a tracer for UO_2. Calculations were made by use of computer codes to model the associated piping, coolers, and detector configurations. Since the presence of Ce-144 was not identified, an MDL calculation was performed.

The quantity of UO_2 remaining in the letdown cooler room is $<0.4\%$ of the maximum allowable residual UO_2 inventory for the entire TMI-2 facility in Mode 2 (Reference 1).

---

* MDL value is defined as the maximum quantity that can be present and still not be observed due to interactions caused by background radiation. The true value can range from 0 to the stated MDL quantity.

TMI-2 POST-DEFUELING SURVEY REPORT
FOR
THE LETDOWN COOLER ROOM

1.0 INTRODUCTION

This report presents the analysis of the amount of UO_2 remaining in the TMI-2 letdown cooler room. The boundaries of this analysis are the makeup and purification (MU&P) system's two letdown coolers (MU-C-1A and MU-C-1B) and a part of the letdown piping (Figures 1 through 5 and References 2 through 5). This report is one in a series of reports to be prepared to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 6). All statistical uncertainties are expressed as $\pm$ one sigma limits (defined as one standard deviation).

Section 2, "Background", describes the physical attributes of the letdown coolers and associated piping and describes their function. The relationship of the letdown cooler room to the accident and subsequent cleanup activities is discussed as well as its current status.

Section 3, "Methods", describes how measurements were obtained through the use of Ce-144 as a tracer for reactor fuel. A calibrated system featuring a shielded sodium iodide scintillation detector connected to a multichannel analyzer was inserted into the letdown cooler room to a known location. Since the presence of Ce-144 was not identifiable, an MDL calculation was performed.

Section 4, "Analysis", explains the methodology for arriving at the estimate of record of fuel in the letdown cooler room using three measurements and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the estimate of record and uncertainty for the amount of SNM remaining in the letdown cooler room, and states that this conclusion is reasonable based upon the available data.

2.0 BACKGROUND

The letdown coolers are located in the Reactor Building (RB) basement, as shown in Figure 6. The letdown coolers are counterflow spiral tube-in-shell heat exchangers which are cooled by the intermediate cooling system (ICS). The purpose of the letdown coolers is to reduce the reactor coolant system (RCS) water temperature from 293° C to 49° C to prevent damage to the MU&P demineralizer resins. The normal RCS water flow rate through the letdown coolers is 2.8 to 4.4 liters per second. Figure 2 shows a cross section of a letdown cooler (Reference 7). Each cooler shell is carbon steel and is 1.09 m in diameter and 0.6 m long, with a 12.7 mm shell thickness and an 83 mm minimum end plate thickness. The design working pressure is 1.4 MPa. The internals consist of stainless steel inlet and outlet distribution headers and thirty 19 mm OD spiral tubes with a working pressure of 17.2 MPa. The letdown coolers are connected to the MU&P system by 76 mm schedule 160 pipe which is reduced into common 63 mm schedule 160 inlet and outlet headers. Figures 3 to 5 show the piping and coolers in the letdown cooler room. The 76 mm inlet piping taps off the bottom of the '1A' RCS cold leg (as shown in Figure 1), goes down to elevation 282'-11", and rises up to enter the letdown cooler room horizontally from the inside of the RB 'A' D-ring at elevation 299'-6" above a service platform. The inlet pipe divides, goes vertically downward through inlet isolation valves (MU-V-1A and MU-V-1B) and expands into the 76 mm pipe. The pipe enters each cooler at approximately the centerline of the cooler. The 76 mm outlet piping is at the bottom of each cooler. The outlet pipes rise through reducers and the outlet isolation valves (MU-V-2A and MU-V-2B), combining into a 63 mm outlet pipe above the service platform at elevation 299'-6". The pipe exits through the Reactor Building wall into the annulus and the Auxiliary/Fuel Handling Building (AFHB).

The letdown coolers are located in the north quadrant of the RB at elevation 282'-6" (see Figure 6). They are in a concrete shielded enclosure adjacent to the RB sump. The room is normally accessed through the RB sump room. Since the March 1979 accident, manned access to the room has been prohibited by high radiation exposure rates in the RB basement. Access by wheeled robots has not been possible because of piping, equipment, and structural congestion.

The RCS flow rate through the letdown coolers during the TMI-2 accident, as controlled by the plant operators, ranged from 0 to approximately 10 liters per second.

During the post-accident recovery, letdown flow was established at approximately 1.6 liters per second and was maintained at this level until the block orifice was removed in September of 1986. The letdown coolers are currently in service as part of the letdown path from the RCS to maintain the reactor vessel water level. The current flow rate has been no greater than 3.0 liters per second, limited by the hydraulic head on the letdown pipe, and is intermittent since it is being used for RCS level control. At 3.0 liters per second, the maximum flow velocity through the piping is about 1.3 meters per second (Reference 8).

The letdown coolers have been examined for residual UO_2 because they are one of the low points of the MU&P system.

Radiation Environment

The radiation measured in the letdown cooler room is approximately 10 Rem per hour near the enclosure ceiling. Normal ingress to the letdown cooler room required access to the Reactor Building basement. Absorption of soluble fission products into porous basement surfaces produce present day general area exposure rates that are frequently >10 Rem per hour.

Due to the hostile environment, measurement access was limited to a spare penetration at the 298'-6" elevation that could be reached from the more accessible 305' elevation. The size of the penetration requires a shielded measurement system that was less than 25 cm in diameter. The 10 Rem per hour exposure rate required the use of at least 7.6 cm of lead around the sodium iodide detector. The measurements were completed in April 1986. No measurable Ce-144 activity was observed which resulted in reporting the minimum detectable quantity.

For the reasons discussed above, it is ALARA to accept the uncertainties associated with the present fuel assay. Additionally, the value is a small part of the total ex-vessel fuel and is not considered easily transferred by accident or design.

3. METHODS

Ce-144 was selected as a tracer for reactor fuel because of its long half-life (284 days), relative high abundance, low escape rate coefficient from reactor fuel, and its characteristic 2.19 MeV gamma-ray that is readily identifiable on a multichannel analyzer (MCA). Using sample data, a Ce-144 to reactor fuel (UO_2) ratio was developed (Reference 9). This value was 5.6×10^6 Bq g^{-1} on August 1, 1987.

To detect the gamma-rays, shielded sodium iodide scintillation detectors were connected to a preamplifier, amplifier, and an MCA. The sodium iodide detector crystals were 12.7 mm in diameter and ranged in length from 12.7 to 19.1 mm. The detectors were shielded with either tungsten or lead to reduce detector dead time and pulse pileup due to the high exposure rate in the room. The detectors were inserted through a spare penetration in the east wall of the letdown cooler room from the incore instrumentation chase. This penetration was one of only three into the letdown cooler room from an accessible location. The penetration is at elevation 298'-6", as shown in Figures 4 and 5. The penetration positioned the detector at about 3.66 m above the coolers and about 0.3 m below the horizontal inlet and outlet pipes.

Prior to their insertion into the room, the sodium iodide detectors were calibrated with a standard Ce-144 source for the 2.19 MeV gamma-ray. The detector's efficiency was determined from the net count rate in the ROI on the MCA for the 2.19 MeV gamma-ray, the energy release rate from the standard Ce-144 source, and the source-to-detector distance. The detector was then inserted into the room to a known distance and spectral data were obtained. Due to the detector shielding and the distance from the coolers to the shield, long count times were necessary to obtain statistically significant data. If an identifiable peak in the ROI at 2.19 MeV for Ce-144 was present, the

measured photon fluence rate at the detector was calculated from the net count rate and the calculated detector efficiency. If the Ce-144 peak was not visually or statistically identifiable, an MDL calculation was performed, as described below.

To complete the calculation for an identifiable Ce-144 peak, radiation shielding computer codes were used to determine a calculated photon fluence rate at the detector from the coolers and piping per kilogram of reactor fuel. The codes modeled the piping, coolers, and platforms as closely as possible in three dimensions. ISOSHLD-II (Reference 10) was used to model the piping sources. QAD-UE (Reference 11) was used to model the coolers because of its ability to model complex geometries and source distributions. The source input to the computer codes was based on the Ce-144 to reactor fuel ratio, taking into account decay time. Parametric studies were performed in the engineering calculations for this report (References 8 and 13) using various reactor fuel distributions in the coolers to determine a range of values. From the range of values, an estimate of record was determined. To determine the quantity of reactor fuel in the coolers and piping, the measured photon fluence rate was divided by the calculated photon fluence rate per kilogram of reactor fuel in each source.

If the Ce-144 peak was not identifiable, an MDL calculation was performed (Reference 12). The gross counts in the ROI for Ce-144, converted to an MDL value, was divided by the calculated detector efficiency, the calculated photon fluence rate per kilogram of reactor fuel, and the count time.

Alternative methods considered for this measurement were gamma spectroscopy by using a high purity germanium (HPGe) detector and neutron interrogation. The HPGe detector has a superior gamma-ray spectrum resolution compared to an NaI detector. The HPGe detector was not used, however, due to space limitations with available equipment. Neutron interrogation techniques could not be used due to the low emission rate of neutrons from the reactor fuel and the high boron content (> 4500 ppm) in the RCS water in the letdown coolers and MU&P system piping.

4. ANALYSIS

On October 25, 1985 (approximately 6.5 years after the accident), a preliminary measurement was performed in the letdown cooler room. The sodium iodide detector crystal was 12.7 mm in diameter and 12.7 mm in length. The detector was shielded with tungsten. The detector was not collimated and, therefore, measured radiation from the entire system piping and the letdown coolers. The configuration of the detector and shield in the penetration is shown in Figure 7. The intent of this measurement was to provide an initial assessment of the quantity of reactor fuel in the room. The count time for this measurement was 100,000 seconds.

This preliminary measurement had essentially no spatial resolution as well as no observed peak corresponding to Ce-144. The MDL* results were equally consistent with having 0.02 kg of fuel assuming uniform distribution (at 2.5×10^{-3} g mm^{-1}) in the inlet and outlet piping, or up to 64.3 kg (Reference 13) in the letdown coolers. These two configurations could produce essentially the same radiation signal since the piping was very close (0.3 m) to the detector and constituted a relatively low shielding mass for attenuating the radiation, while the coolers were both more distant (3.66 m) and more massive. Since the spread of values was great and the larger values are a significant fraction of the then existing 70 kg administrative limit (now 140 kg) for sub-critical mass (Reference 14), two additional measurements were performed with the goal of better defining locations and quantities of reactor fuel.

The later measurements replaced the uncollimated detector with a collimated detector. The tungsten shield was also replaced with lead since it was found that the tungsten was contaminated with thorium. A 2.6 MeV gamma-ray is produced in the thorium decay chain which interfered with the acquisition of spectral data in the ROI of the characteristic gamma-ray of Ce-144. The effect of the thorium contamination was accounted for in the calculations of the quantity of reactor fuel in the room.

The final measurement set was performed from April 16 to 21, 1986 (approximately 7 years after the accident). The sodium iodide detector crystal was 12.7 mm in diameter and 19.1 mm in length. A collimated lead shielded detector assembly was used, as shown in Figure 8. The sodium iodide detector was placed inside a lead cylinder with a hemispherical head. This assembly was placed inside an aluminum cylinder lined with 19 mm-thick lead. A slot cut in the end of the cylinder provided the desired collimation. The detector/shield assembly was designed so that two different measurements could be made by moving the shielded detector in the shielded tube. In the first measurement, the detector was located in the front of the shield tube as shown in Figure 8. In this position, the detector scanned the coolers and vertical inlet and outlet piping and valves, as shown in Figure 9. For the second measurement, the detector was moved back slightly in the shield tube so that the detector scanned the coolers and minimized scanning the inlet and outlet piping. The field of view of the detector is shown in Figures 10 and 11.

The measurement times for the final set in the first and second measurement positions were 155,493 and 240,948 seconds, respectively, resulting in gross counts in the ROI for Ce-144 of 170 and 117, respectively. The statistical quality of the data was improved from the first measurement by using longer count times. Although there were counts in the ROI for Ce-144, the results of the measurements showed that there was not a statistically identifiable gamma peak at 2.19 MeV energy location on the MCA. This resulted in an MDL calculation for the quantity of reactor fuel. The sodium iodide detector efficiency was 1.2×10^{-2} counts gamma^{-1} cm^{2}. The calculated photon fluence per kilogram of UO_2 using the QAD-UE computer code ranged from 7.2×10^{-4} to 9.7×10^{-3} gammas cm^{-2} s^{-1} kg^{-1}. The estimate of record is an MDL value of $\leq$ 3.7 kg $\pm$ 53% of UO_2 in the letdown cooler room. The 53% value is a 1σ error that accounts for all other errors in the calculation other than the counting error for the letdown coolers. This estimate of record assumes that the reactor fuel is uniformly distributed in the cooler inlet distribution header and in the bottom quadrant in the first spirals of the tubes. Particles of reactor fuel would most likely have settled here due to a decrease in the flow velocity from the inlet pipe into the tube bundle.

The MDL quantity of reactor fuel in the letdown coolers was calculated to range from <1.9 kg to <25.6 kg, using improbable reactor fuel distributions with an uncertainty of approximately ± 53% (Reference 8). These values take all sources of error into account (i.e., calibration and measurement counting, physical measurements, calibration, and standard source). This range in quantity of reactor fuel assumed that all the reactor fuel in the room was in the coolers. The low value assumed that the fuel was in the inlet distribution header. The high value assumed that the reactor fuel was uniformly distributed in the bottom of the cooler near the outlet distribution header. The region where deposition was most likely was in the cooler distribution header and in the bottom quadrant of the first spirals of the tubes. This region was used to establish the estimate of record of ≤3.7 kg ± 53%. The form of the reactor fuel is not known since this part of the MU&P system is not available for internal visual inspection or sampling. The piping was calculated previously to have an MDL value of ≤ 0.02 kg (Reference 13), which is insignificant when compared to the coolers.

The quantity of reactor fuel in the two letdown coolers is currently considered static based on analyses of the flow rate from the cold legs to and through the coolers, particle size distribution of the reactor core debris, data from other plant locations, and the layout of the MU&P system.

The maximum flow velocity in the cold legs is 0.004 meters per second, and through the piping and coolers is approximately 1.3 meters per second and 0.2 meters per second, respectively. Samples of the reactor core fuel debris show that a majority of the particles by weight (70%) are greater than 1000 μm (Reference 14). The density of the material is assumed to be approximately 5 g cm^{-3} (Reference 14). Assuming that the particle size distribution and density has remained the same from the time of the accident, a flow velocity of approximately 0.004 meters per second in the cold leg piping is insufficient to transport reactor fuel particles greater than 40 μm (Reference 8) to the MU&P piping and the coolers. Particles less than 40 μm either remain in suspension or, if settled out on termination of flow through the cooler, become resuspended and are carried out of the cooler when flow is reestablished.

The samples from the reactor core debris are conservative, since the sample may have had washout that removed fine fuel debris that was less than 1 μm.

Fine reactor fuel debris of this size would be readily moved. The once-through steam generator (OTSG) 'B' upper tube sheet sample data showed that greater than 92% (by weight) of the particles were greater than 1000 um (Reference 15). Sampling attempts in the RCS J-Legs showed very fine debris. A 1 μm filter used as a collector was not able to remove debris observed as dark-colored water flowing through the sampling equipment. Visual inspections of these areas also showed fine powder-like reactor fuel debris that would be easily removed.

There are low points in the piping and the letdown coolers. The MU&P piping to the letdown coolers taps off the bottom of the RCS 1A cold leg (Figure 1) and continues down to elevation 282'-11". After a horizontal run of approximately 8 m, the letdown line rises to elevation 299'-6". The piping then drops to the letdown coolers where the center line elevation is 282'-11". These low points in the piping and the letdown coolers would tend to prevent large (>10,000 μm) particles from getting to or leaving the letdown coolers at current flow rates. These particles could have been transported to the letdown coolers when the flow rates were as high as 10.1 liters per second into the MU&P piping. The current maximum flow rate of 3.0 liters per second would be insufficient to remove large particles that were transported to the letdown coolers under the higher flow rate.

Although the letdown coolers have not been isolated, the quantity of reactor fuel they contain, as stated in this report, is considered stable. For approximately the last three years, the flow rate through the letdown coolers has been at approximately 2.8 liters per second with a maximum flow rate of 3.0 liters per second. This continuous flow rate results in a stable equilibrium where reactor fuel will not be added or removed from the letdown coolers. This flow rate through the cold leg and the MU&P piping is not conducive to moving additional large reactor fuel debris to or from the coolers. Small fuel debris particles will remain in suspension or be moved through the coolers. After RCS system draindown, the letdown cooler system will be isolated and a confirmatory survey may be performed, if deemed appropriate, to ensure that the reported values have not significantly changed.

The goal of the current defueling program is to remove more than 99% of the original core inventory of approximately 94,000 kg. In that context, the $\leq$ 3.7 kg quantity of UO_2 remaining in the pressurizer is $<$0.4% of the maximum allowable residual UO_2 inventory for the entire TMI-2 facility in Mode 2 (Reference 1).

5. CONCLUSION

A MDL value of $\leq$ 3.7 kg with an uncertainty of approximately $\pm$ 53% (at one sigma) is the estimate of record of UO_2 remaining in the letdown coolers and piping in the letdown cooler room. The piping is calculated to have an MDL value of $\leq$0.02 kg, which is insignificant compared to the MDL value of the letdown coolers. Therefore, the estimate of record for the letdown cooler room is considered the same as the letdown coolers, which is $\leq$ 3.7 kg $\pm$ 53%. Although the letdown coolers have not been isolated and reactor coolant still travels through them, the estimate of record of the amount of reactor fuel is considered valid. The flow rate is currently low and intermittent as opposed to the high and continuous flow rate during, and for a long period of time following, the accident. The flow rate is no greater than 3.0 liters per second and is intermittent since it is being used for RCS level control. The maximum flow velocity through the piping is approximately 1.3 meters per second, which is sufficient to have long ago removed fine fuel debris from the letdown coolers, and to have prevented the buildup of fuel debris.

REFERENCES

1. R. E. Rogan. Residual Fuel Basis. GPU Nuclear Internal Memorandum 4410-88-M-0306/4400-88-078. Middletown, PA: GPU Nuclear Corporation. 24 August 1988.

2. Burns and Roe Drawing 2024. Flow Diagram, Reactor Coolant, Makeup & Purification.

3. Burns and Roe Drawing 2168. Makeup & Purification & Intermediate Cooling-Reactor Bldg., Plan El. 282'-6".

4. Burns and Roe Drawing 2413. Makeup & Purification, Intermediate Cooling & Nuclear Services C.C. Wtr.-Reactor Bldg.-Sections.

5. Burns and Roe Drawing 2066. General Arrangements, Reactor Building, Plan El. 282'-6".

6. J.S. Schork. TMI-2 Core and Special Nuclear Accountability Program. Administrative Procedure 4000-ADM-4420.02, Rev. 3-01. Middletown, PA: GPU Nuclear Corporation. 10 October 1988.

7. TMI File Drawing 007-20-1, Graham Manufacturing Co., Inc. Drawing 30-540-4066. Weldseal Heliflow Letdown Cooler.

8. P.J. Babel. Calculation of Letdown Coolers Fuel Debris. GPUN Calculation 4550-3211-86-056, Rev. 5. Middletown, PA: GPU Nuclear Corporation. 2 August 1989.

9. P. J. Babel. Ce-144, Eu-154 and Eu-155 as Tracers for Fuel Debris. TB-86-41, Rev. 2. Middletown, PA: GPU Nuclear Corporation. 29 February 1988.

10. ISOSHLD-II: Code Revision to Include Calculation of Dose Rate from Shielded Bremsstrahlung Sources. BNWL-235, Supplement 2. April 1969.

11. QAD-UE, Revised Numerical Integration Option for Gamma-Ray Volume Source Problems in the QAD-CG Point Kernal Shielding Code System. RSIC Computer Code Collection CCC-448.

12. L. A. Currie. Analytical Chemistry. Vol. 40, 1968, "Limits for Qualitative Detection and Quantitative Detection-Application to Radiochemistry", p. 586-593 (Table II).

13. P. J. Babel. Calculation of Letdown Coolers Fuel Debris. 4550-3211-85-008, Rev. 0. Middletown, PA: GPU Nuclear Corporation. 4 November 1985.

14. Criticality Safety Evaluation for Increasing the TMI-2 Safe Fuel Mass Limits. Safety Evaluation Report 4710-3220-89-1, Rev. 0. Middletown, PA: GPU Nuclear Corporation. February 1989.

15. D.W. Ackers, et al. TMI-2 Core Debris Grab Samples - Examination and Analysis. GEND-INF-075. September 1986.

16. T.L. Hardt and G.O. Hayner. TMI-2 B-Loop Steam Generator Tube Sheet Loose Debris Examination and Analysis. October 1987.

FIGURE 1

LETDOWN COOLERS AND REACTOR COOLANT SYSTEM

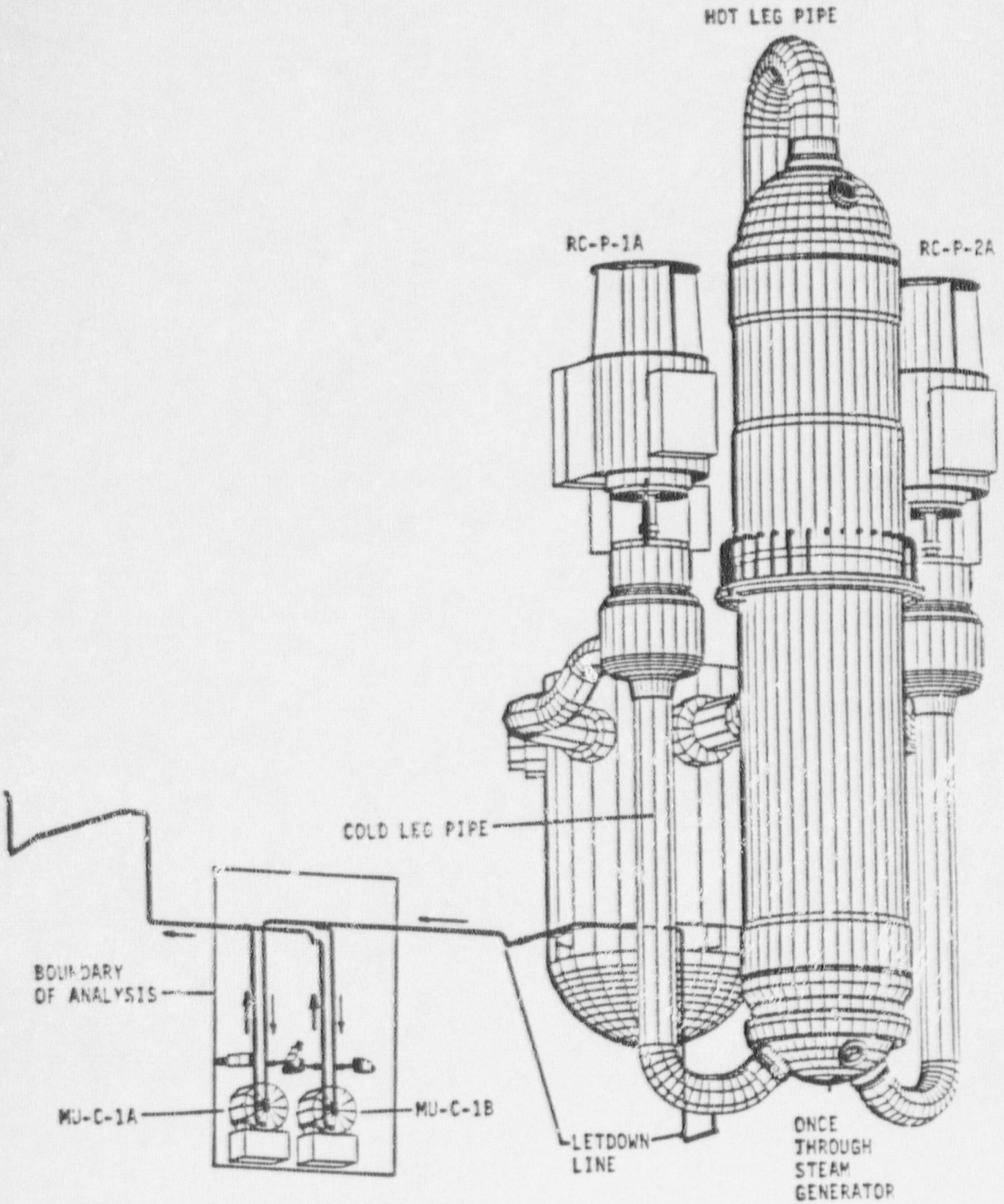


FIGURE 2

LETDOWN COOLER ROOM, ISOMETRIC VIEW

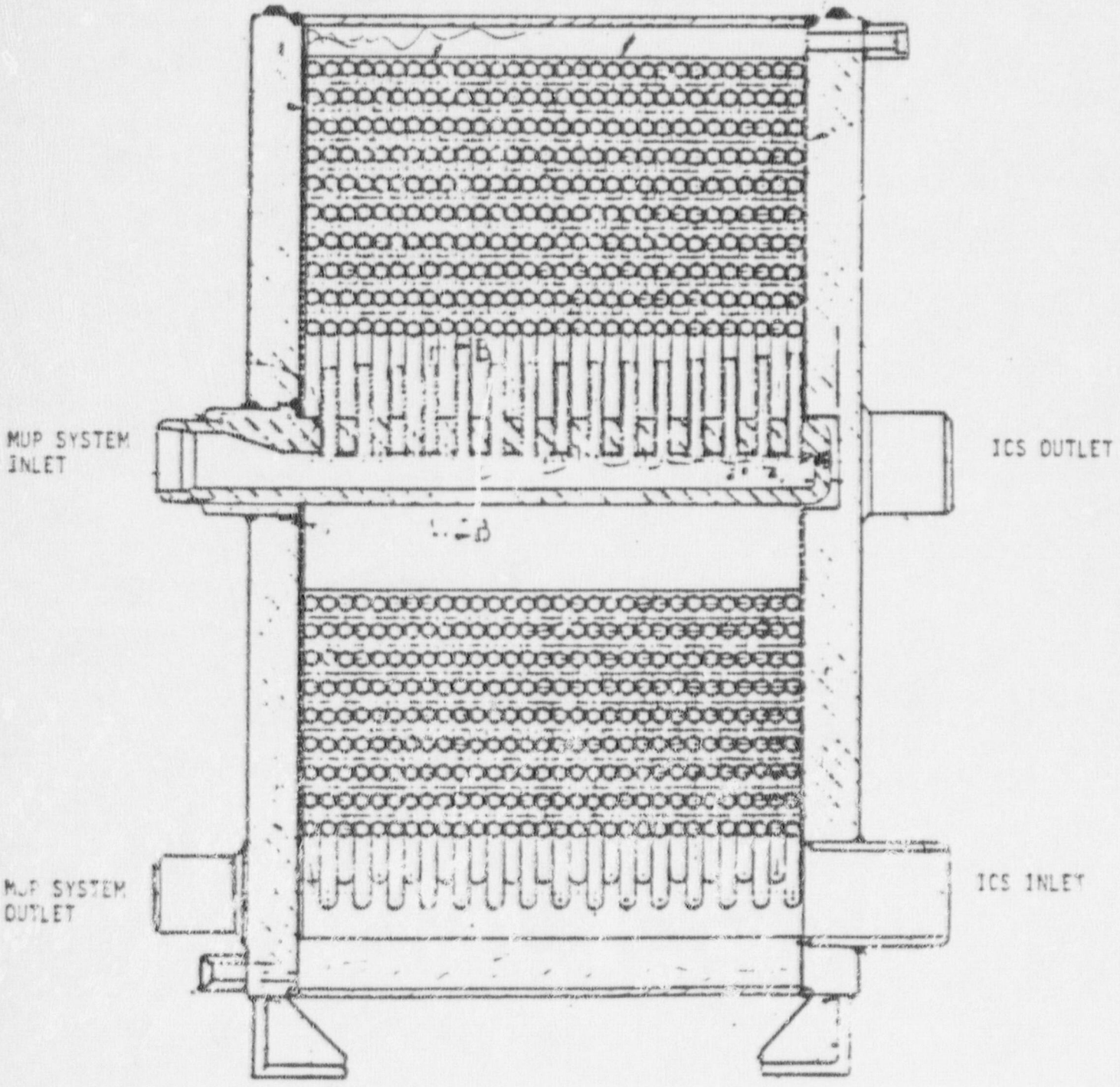


FIGURE 3

LETDOWN COOLER ROOM, PLAN VIEW

FIGURE 4

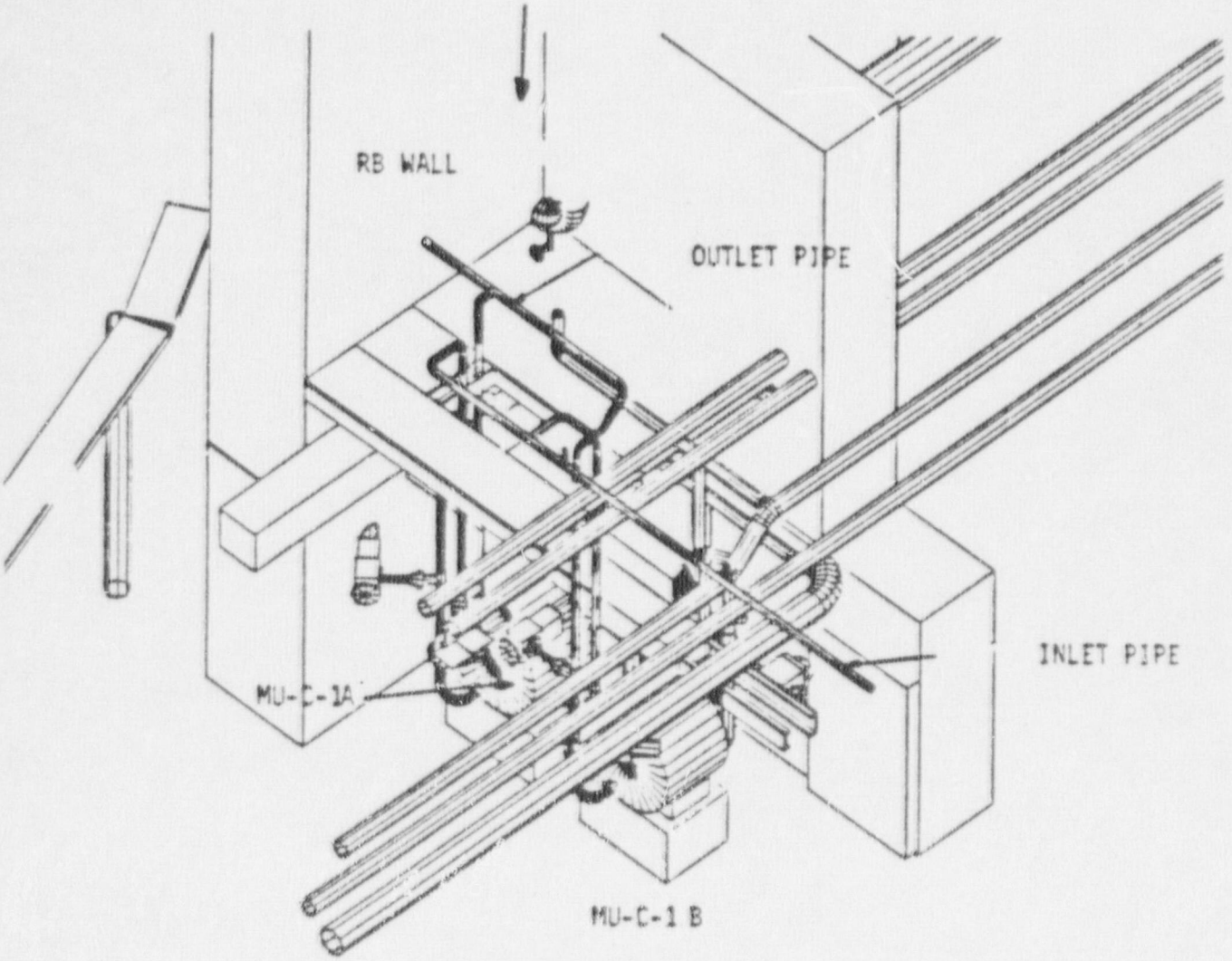


FIGURE 4

LETDOWN COOLER ROOM PENETRATIONS, PLAN VIEW

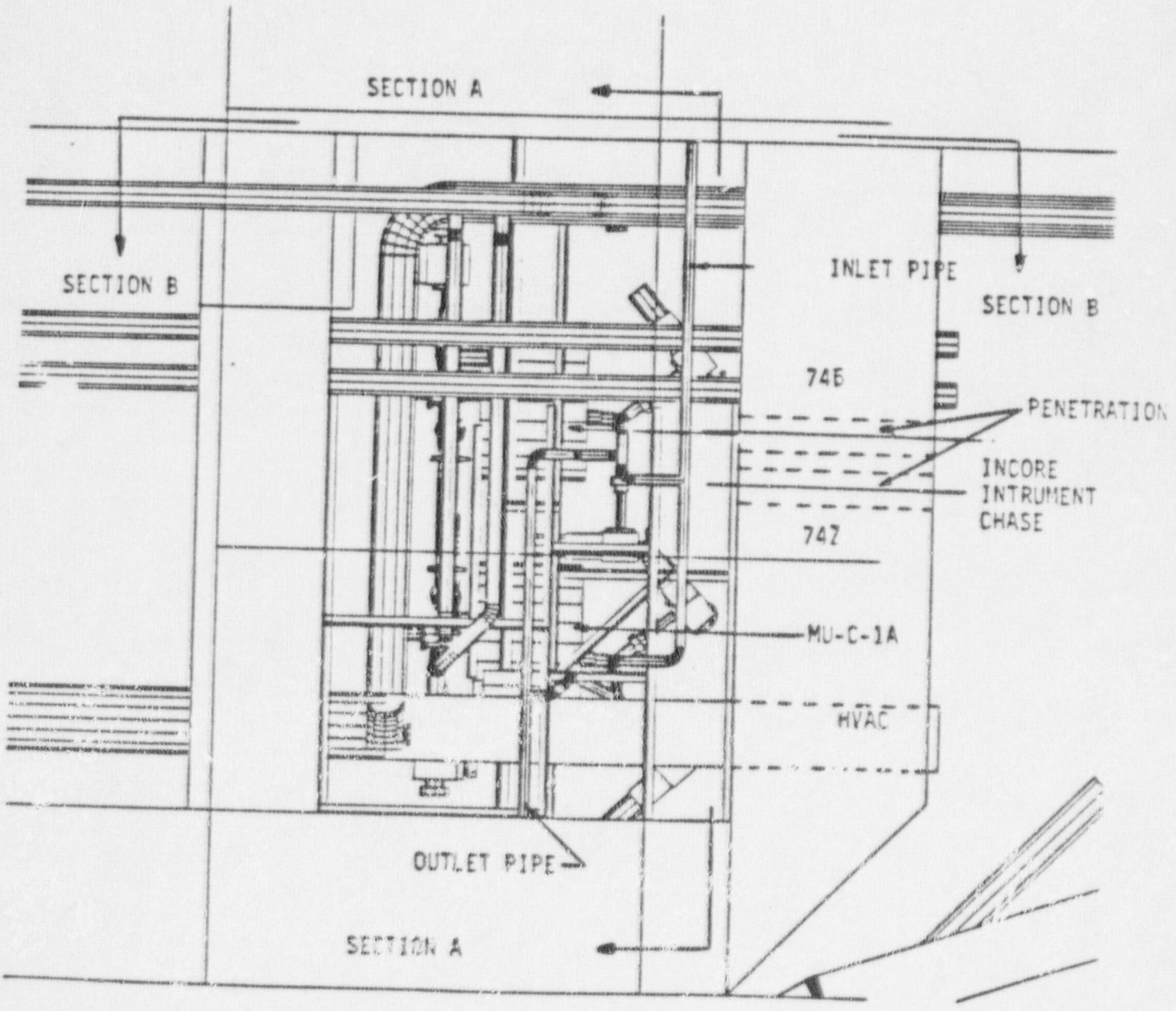


FIGURE 5

LETDOWN COOLER ROOMS, SECTION A-A PENETRATIONS

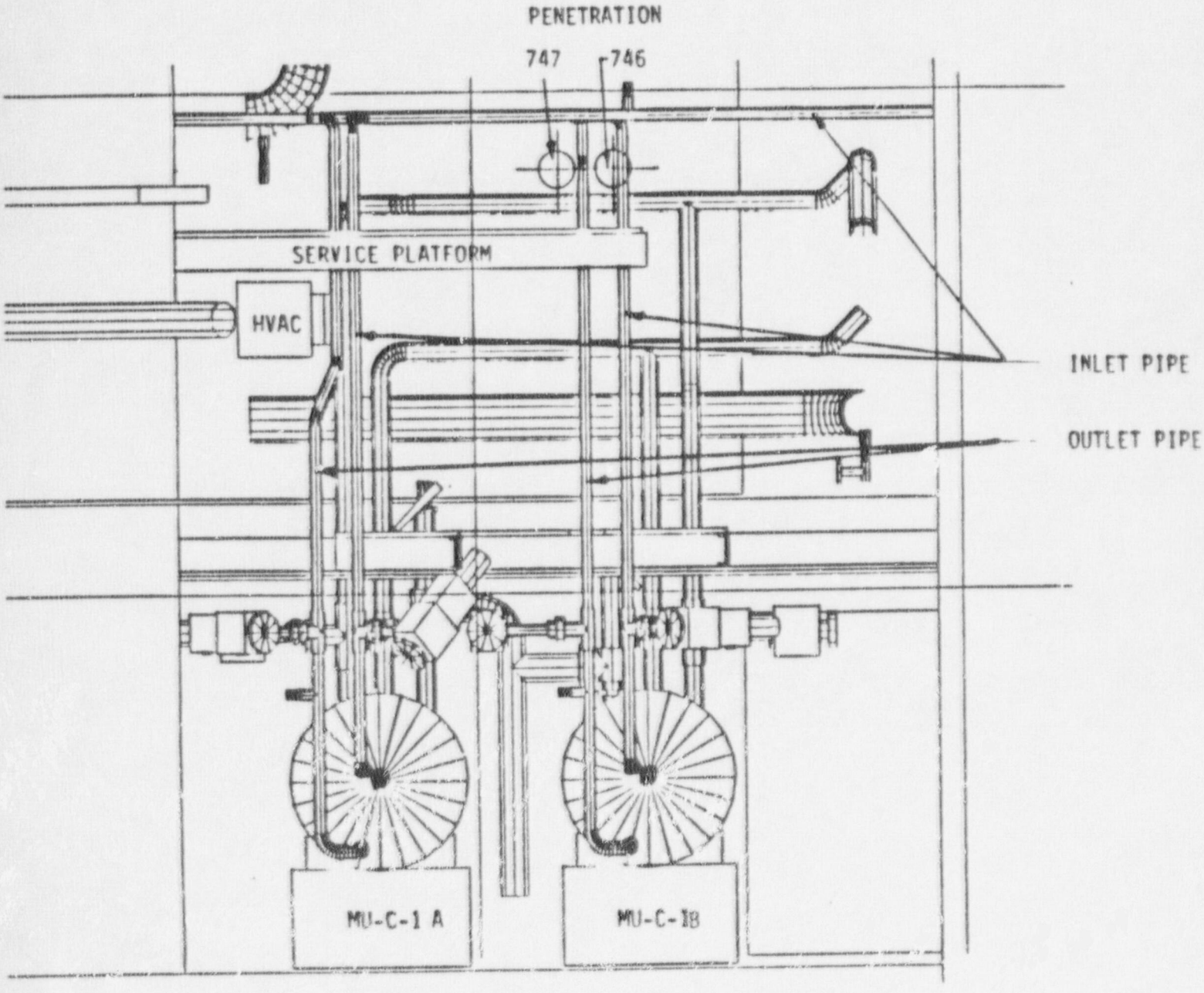


FIGURE 6

REACTOR BUILDING BASEMENT, GENERAL ARRANGEMENT

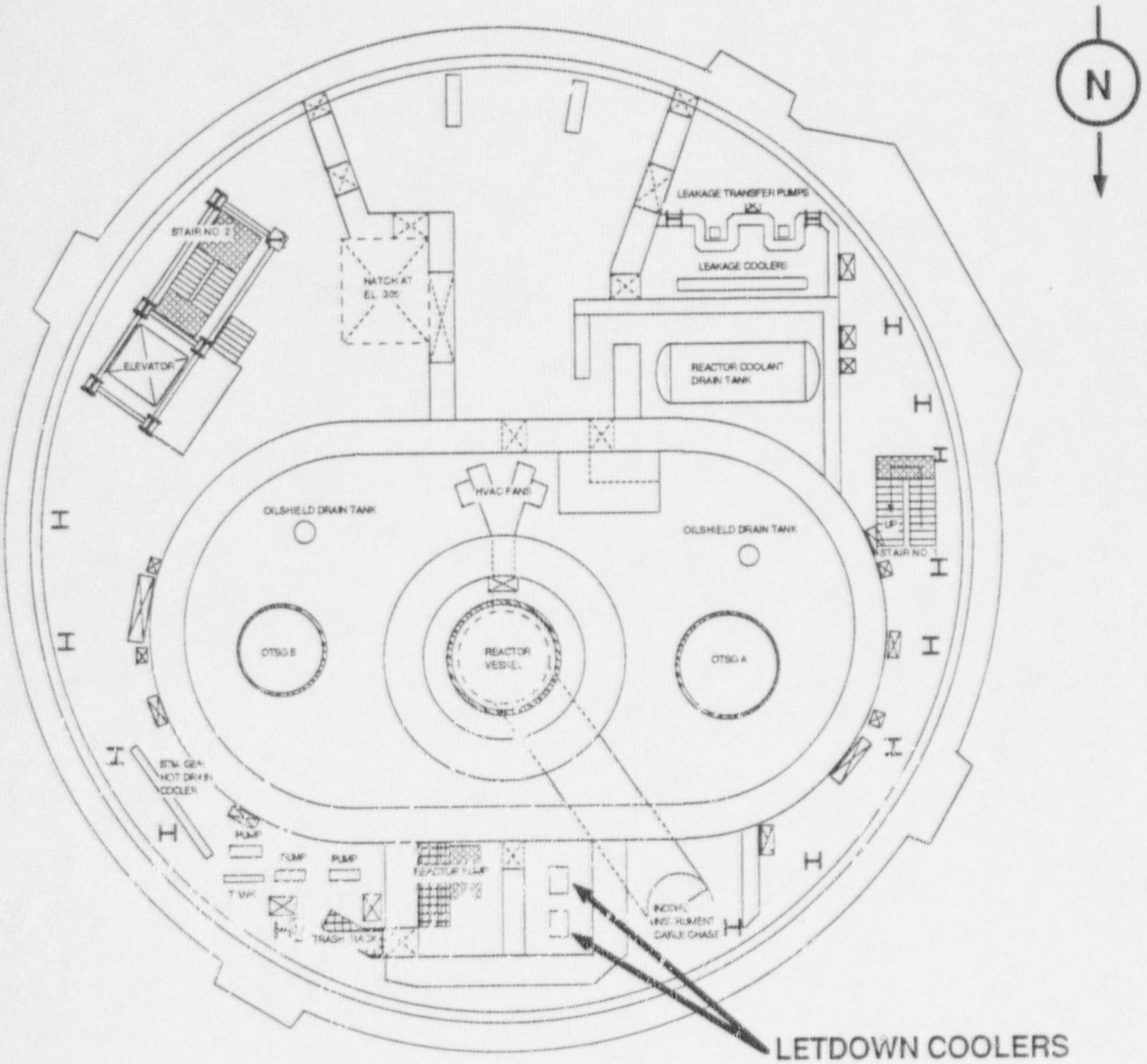
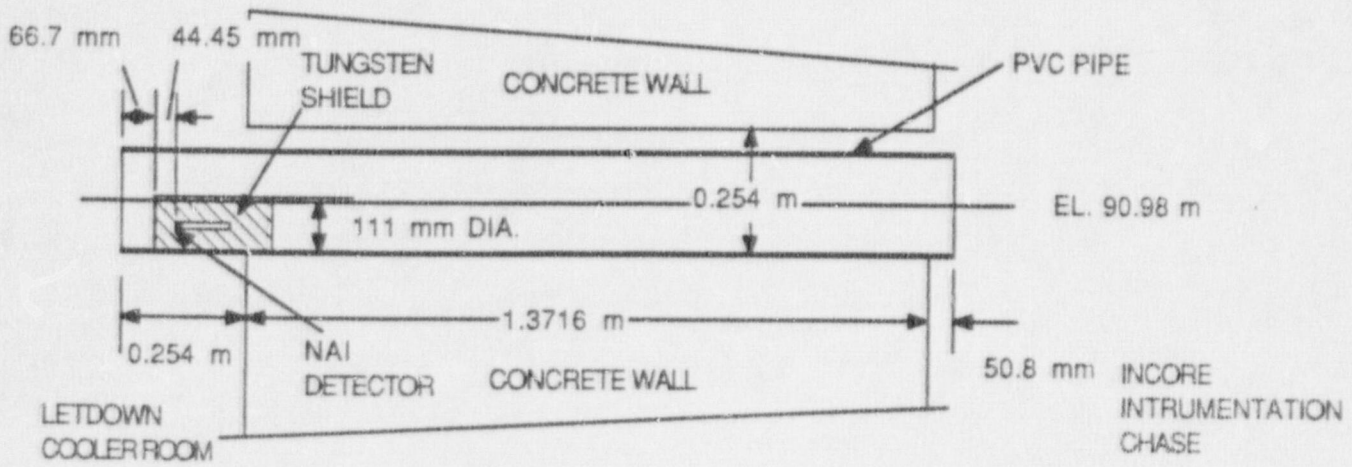


FIGURE 7

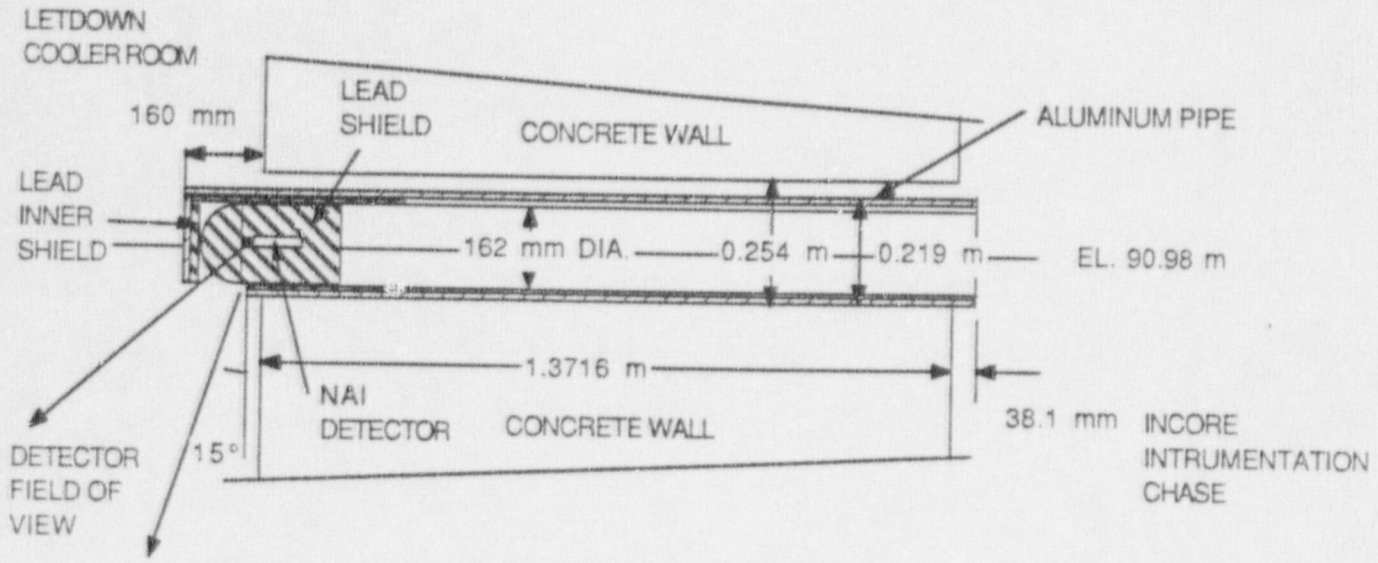
SODIUM IODIDE DETECTOR IN LETDOWN COOLER ROOM WALL



PENETRATION NO. 747

FIGURE 8

SODIUM IODIDE DETECTOR IN LETDOWN COOLER ROOM WALL



PENETRATION NO. 746

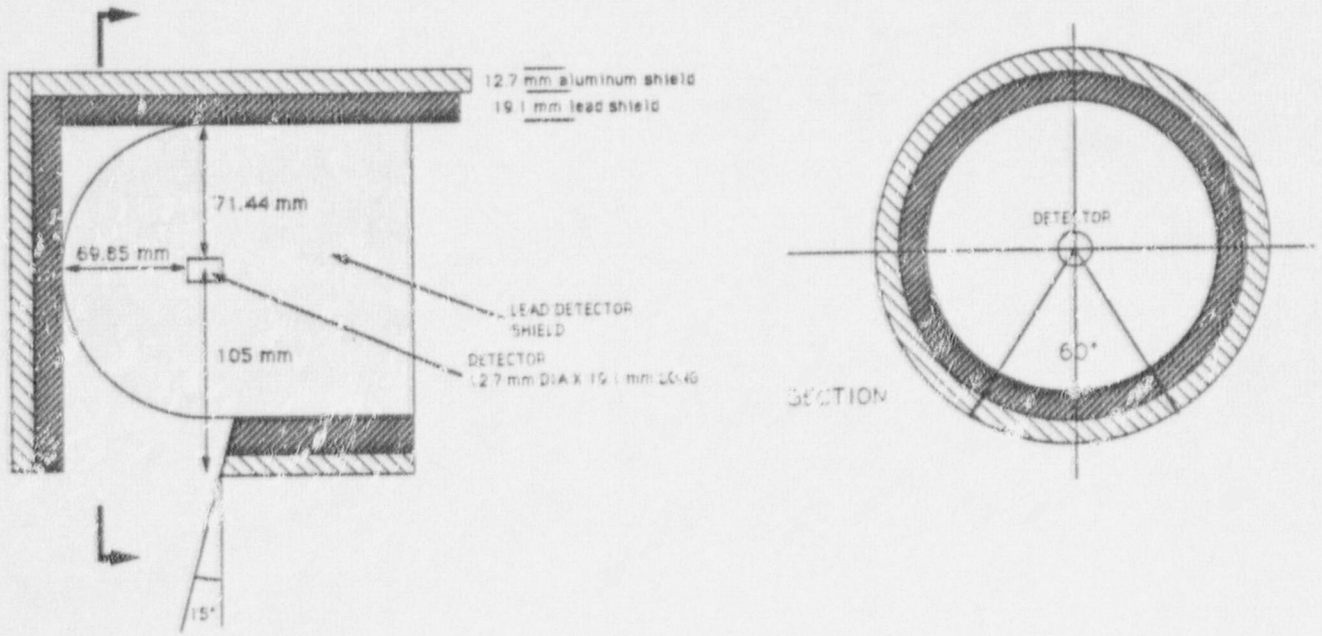


FIGURE 9

LETDOWN COOLER ROOM - FIELD OF VIEW, SECTION B-B

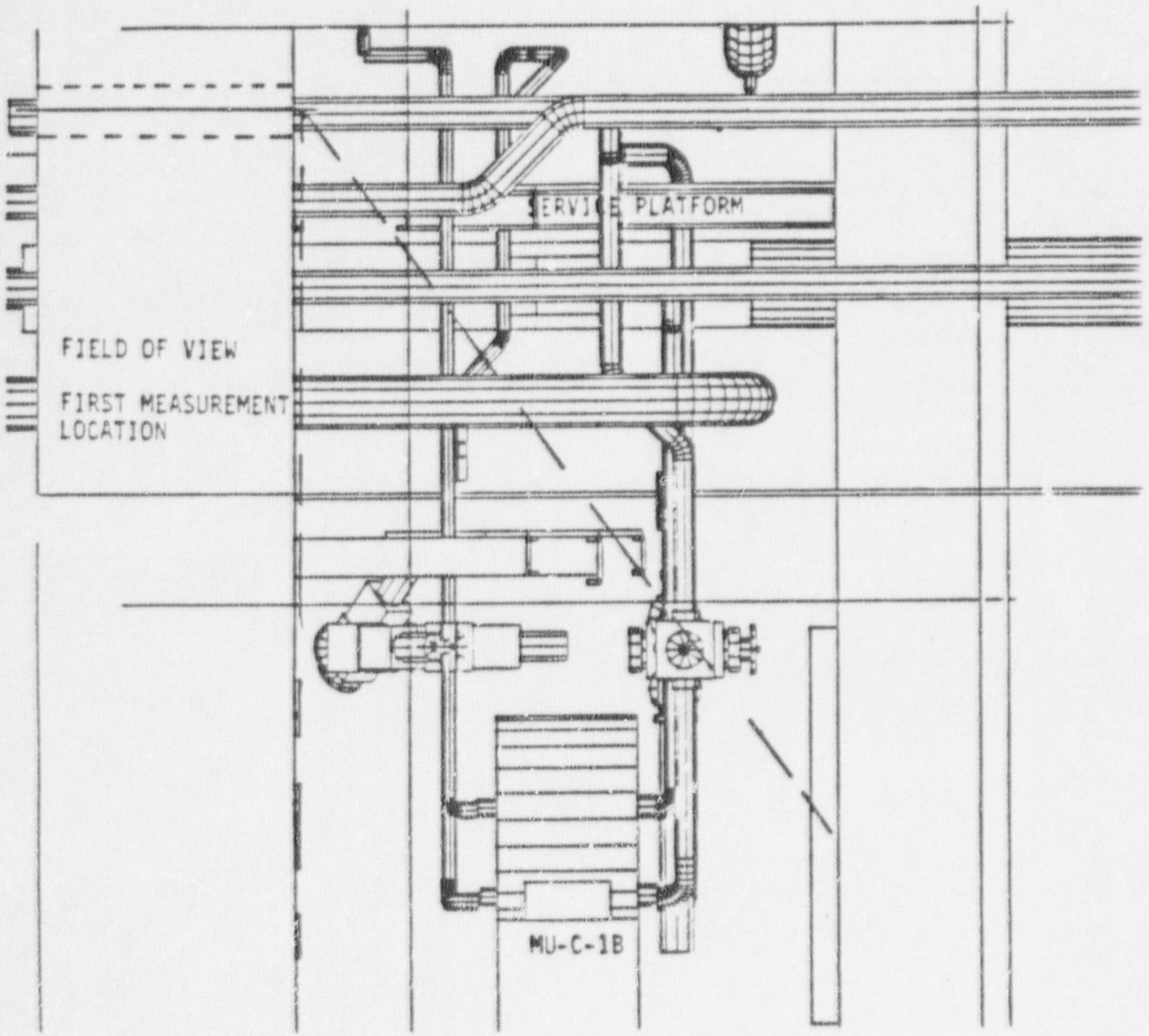


FIGURE 10

LETDOWN COOLER ROOM FIELD OF VIEW, SECTION A-A

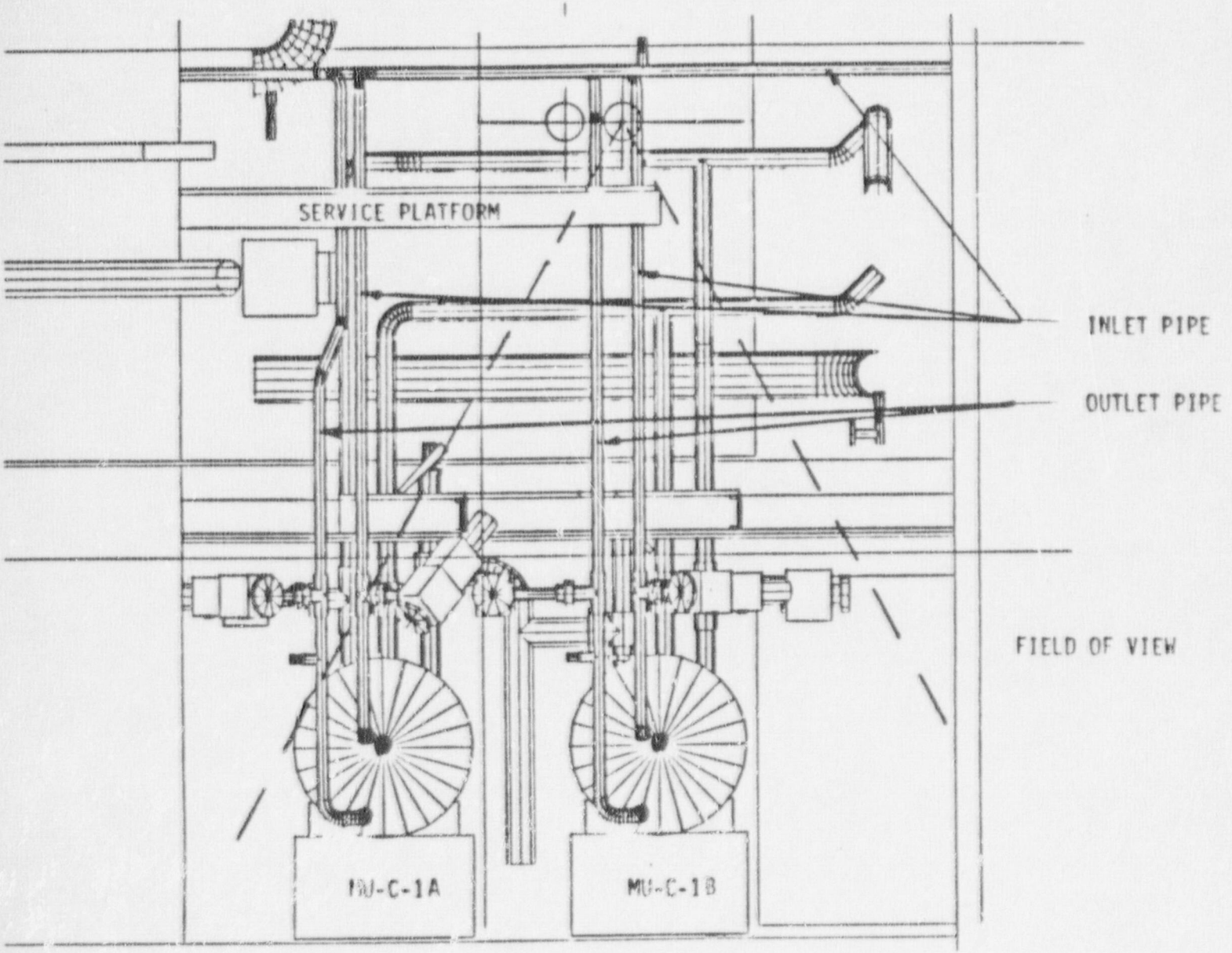


FIGURE 11

LETDOWN COOLER ROOM FIELD OF VIEW, SECTION B-B