ILLINOIS POWER COMPANY

CLINTON POWER STATION - UNIT #1

POST ACCIDENT SAMPLING SYSTEM EVALUATION REPORT

April 1986

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LIST OF ACRONYMNS

ACRONYM	DEFINITION
ALARA	As Low As Reasonably Achievable
BOP	Back of Panel
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
CA	Containment Atmosphere
CAM	Containment Atmosphere Monitoring
CPS	Clinton Power Station
DAAS	Data Acquisition and Analysis System
DO	Dissolved Oxygen
FOP	Front of Panel
GC	Gas Chromatograph
GSP	Grab Sample Panel
HEPA	High-Efficiency Particulate Air
IC	Ion Chromatograph
IMCC	Inpanel Multi-Counting Cave
L&N	Leeds & Northup
LOCA	Loss-of Coolant Accident
LRG	Licensing Review Group
PASS	Post Accident Sampling System
PWR	Pressurized Water Reactor
RC	Reactor Coolant
RHR	Residual Heat Removal
RWCUS	Reactor Water Cleanup System
SAP	Sample Analysis Panel

LIST OF ACRONYMNS (Cont'd)

ACRONYM	DEFINITION
SEC	Sentry Equipment Corporation
SMP	Sample Monitor Panel
TID	Total Integrated Dose
YSI	Yellow Springs Instrumentation

April 86
Revision 1

POST-ACCIDENT SAMPLING SYSTEM

Illinois Power Company has provided a Sentry Equipment Corporation Model B Post Accident Sampling System (PASS) for the Clinton Power Station in response to Requirement II.B.3 of NUREG-0737 "Clarification of the TMI Action Plan Requirements". Item II.B.3 titled "Post Accident Sampling" provided 11 criteria which the sampling system must satisfy. The purpose of this report is to address the 11 criteria as they apply to the Clinton Power Station.

This report is organized as follows:

I. Main body of report - Each of the 11 criteria is presented. Following each criteria is the verbatim clarification to the criteria. This clarification is taken from the post-implementation review letter issued by the NRC to numerous operating plants in 1982. Following each clarification statement is the Illinois Power Company position on the design criteria.

II. Tables - Table references are provided throughout the text.

III. Figures - Figures depicting equipment designs and arrangement are provided.

IV. Attachments - The following attachments are provided:

Attachment 1 - PASS - Operator Dose Report

This report is being provided to assist the NRC post-implementation review of the Clinton Power Station Post-Accident Sampling System as required in License Condition No. 6 of the Clinton Safety Evaluation Report (NUREG-0853).

Criterion 1

The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Clarification 1

Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of offsite power (i.e. designate an alternative backup power source, not necessarily the vital (Class 1E) bus, that can be energized in sufficient time to meet the 3-hr sampling and analysis time limit).

Position 1

The design scope of the Post Accident Sampling System (PASS) is to obtain reactor coolant (RC) samples, and containment and drywell atmosphere (CA) samples for radiological analysis in the event of a loss-of-coolant accident (LOCA).

The Clinton PASS consists of: 1) a Sample Analysis Panel (SAP) located in the Diesel Generator Building at elevation 737' (see Figure 1 and 2 for sample room layout and location); 2) a Sample Monitor Panel (SMP) situated within the sample station room; 3) associated equipment (liquid sample cask/cart, etc.) and inter-connecting tubing.

The Sample Analysis Panel (SAP) contains equipment to handle and analyze the sample (See Figures 3 through 5), and the Sample Monitor Panel (SMP) contains the controls and indicators for the sampling process (See Figures 6 and 7). This station is located in close proximity to the Containment Building; therefore minimizing radiation exposure from sample lines.

The SAP is divided into two sections, one for liquids and one for gases. The liquid samples routed to the panel consist of:

1. Reactor water clean-up effluent (for normal plant operation).
2. Reactor coolant via a reactor vessel Jet Pump Instrument line.
3. Drywell Equipment Drain Sump.
4. Containment Equipment Drain Sump.

Position 1 (Continued)

5. Drywell Floor Drain Sump.
6. Containment Floor Drain Sump.
7. RHR pump 1A or 1B effluent (Reactor coolant or Suppression pool water).

The gas samples routed to the system consist of:

1. Containment Atmosphere (2 sample locations, elevations 790 and 740 feet at AZ 82°).
2. Drywell Atmosphere (2 sample locations, el. 790 and 740 feet at AZ 0°).

Sample points locations along with system schematics are shown on Figure 8.

The operator has control and indication of the sampling process at the SMP. This panel includes a mimic, switches, instrument monitors and lights for: 1) the control and indication of valves for the routing of samples; and 2) monitoring and control of PASS instrumentation and systems (See Figure 9). The isolation valves for sample lines penetrating the containment have their status indicated at the SMP, but they are controlled from the Main Control Room.

The SAP is provided with a mimic of pertinent flowpaths within the panel's analyzers, grab samples, etc. (See Figure 10).

The reactor coolant sample is obtained from a reactor jet pump pressure instrumentation sensing line until the reactor is depressurized. After the reactor is depressurized, the sample is taken from either RHR A or RHR B pump discharge to assure that a sample representative of the core condition is obtained. Dissolved gases in the reactor coolant are extracted in a depressurization chamber. The gas is pressurized with a carrier gas and transported to the gas chromatograph in the sample panel. The carrier gas is prevented from entering the reactor coolant system by isolating the reactor coolant before the carrier gas is introduced. The diluted gas may also be directed to the SAP for a grab sample.

During normal operation, a liquid sample may be taken at the sample sink on the SAP. The sink drain, line pressure relief valve, and analyzers normally drain to the Fuel Building floor drain tank. In the event a high level of radiation is present, the liquid sample return drain is manually directed to the suppression pool via a switch located at the SMP.

A suppression pool liquid sample can be obtained from the RHR loop that is lined up in the suppression pool cooling mode, or low pressure coolant injection mode.

Position 1 (Continued)

The liquid sample lines are backflushed with demineralized water (except during a loss of offsite power event) to minimize plateout, blockage and contamination of the next sample. The flushing water is normally routed to the Fuel Building drain tank when radiation levels are low and to the suppression pool when radiation levels are high. The routing of the flushing water is controlled by means of a manual switch on the SMP. A liquid sample is first routed to the sample cooler in the SAP where it is cooled to within the limits for the inline analyzers. Then, the sample is passed near a radiation detector to inform the operator at the SMP of the radiation level. The liquid is then routed through a pH analyzer, a specific conductivity analyzer, and a dissolved oxygen analyzer. Stripped off-gas from the liquid sample is routed through the hydrogen analyzer. Readings are displayed on indicators at the SMP. The sample can be diluted to a ratio of 1,000 to 1 with demineralized water.

The SAP will provide access to liquid grab samples collected in a shielded, portable, sealed vial. The panel is shielded to protect personnel when taking a grab sample to reduce radiation exposure. The sample is transported to the chemistry laboratory for radionuclide, chloride and boron analysis.

Gas samples are passed near a radiation detector to inform the operator at the SMP of the radiation level. Gases from the drywell and containment plus off-gas from the reactor coolant are obtained at the gas section of the SAP manually by opening a valve which injects the gas via a hypodermic needle into an evacuated vial inside a tong. The unused gas is returned to containment.

After obtaining a gas sample, the system is backpurged with the carrier gas to clear the lines and eliminate contamination of the next sample. Purging gas can be returned to the containment or the drywell.

After a gas sample has been obtained, it is taken to the Chemistry laboratory for radionuclide analysis. Hydrogen and oxygen analyses are performed on-line by a separate system, i.e., the Containment Atmosphere Monitoring system. Gas sample lines are heat traced to minimize condensation of steam (from the containment atmosphere) and resultant loss of iodine.

The chemistry laboratory and counting room to be used for post accident analysis are located in the Control Building, at elevation 737. The location of the lab with respect to the PASS panel is shown on Figure 11.

Position 1 (Continued)

Figure 12 presents a layout of the laboratory complex, including a cold lab, chemistry lab, high level area, radchem office, chemical storage room, Radiation Protection Calibration Facility and a bioassay laboratory. All of the facilities have been sized upon experience at operating plants, and have been laid out to permit an efficient operation. The radiological chemistry laboratories are maintained at a slightly negative pressure to keep any airborne contamination from escaping to the general areas. The counting room, shown in Figure 12, is shielded on all sides to maintain a low background radiation level and make it less sensitive to changes in the radiation levels outside. Additionally the ventilation system is designed to supply filtered outside air and maintain a slightly positive pressure to help keep out any airborne contamination. The total distance to be covered to transport a grab sample from the PASS panel to the lab is approximately 300 ft. The maximum time required for the transportation of the sample to the lab is conservatively estimated to be 20 minutes.

Sentry Equipment Corporation (SEC) secured adequate time and motion data by performing the requisite sampling and analyses exercises on a Breadboard model built in accordance with SEC Specification No. B10-01. Clinton's specific times for sampling and performing inline analyses were determined by factoring a 10% safety margin into Sentry's time estimates.

Total time for the PASS liquid sampling exercises is predicted to be 101 minutes.

The sampling and analyses exercises include: purging to ensure a contemporary and representative liquid sample; obtaining three grab samples (undiluted liquid, diluted liquid and diluted RC off-gas); performing four inline chemical analyses (Hydrogen in RC off-gas, liquid dissolved oxygen, pH and conductivity); and flushing PASS flowpath during and after exercises.

An extra 18 minutes are needed for completing gaseous sampling exercises which include capturing a gas sample and flushing the gas sample lines.

The following are estimated times and procedures to perform onsite inline or laboratory analyses:

- Liquid Dissolved Hydrogen

The total time to analyze liquid dissolved hydrogen (RC off-gas) is approximately 76 minutes. The analysis is performed using an inline Gas Chromatograph (GC).

Position 1 (Continued)

- Dissolved Oxygen, Liquid pH, Liquid Conductivity

The analysis for dissolved oxygen, liquid pH, and liquid conductivity is also via inline analyzers; the total required time for these analyses is approximately 40 minutes.

- Liquid Nuclides

These analyses will be performed by counting grab samples in the Radiological Chemistry Laboratory counting room:

The total time to complete these analyses is 134 minutes. The total time includes 84 minutes for obtaining a grab sample, 20 minutes for transport time to the lab, and 30 minutes to acquire a gamma spectrum in the counting room.

- Liquid Boron

This analysis will be performed on a diluted sample via a Tetrafluoroborate Selective Ion Electrode (TSIE). The time required for this analysis is approximately 15 minutes. The total time (from the start of sampling) for this task is approximately 119 minutes.

- Liquid Chloride

An undiluted reactor coolant sample will be analyzed at an offsite facility within four (4) days. In the event of a minor accident (sample activity is a fraction of the worst case activity), a liquid sample may also be analyzed onsite via an Ion Chromatograph. Completion of this analysis does not fall within the 3-hour requirement. (See Position No. 5)

-Gaseous Sampling and Nuclide Analyses

This analysis will be performed by counting a grab sample in the Radiological Chemistry laboratory counting room:

The gas grab sample is ready for transport to the counting room 107 minutes after beginning the PASS exercises. The transport time to the site analytical laboratory is 20 minutes, and the time to complete all counting is predicted to be 45 minutes. Therefore, the counting could be completed in 172 minutes (107+20+45).

Position 1

However, the counting facility will be occupied with earlier liquid PASS sample through minute 134. Consequently, the realistic time to complete the count is 179 minutes (134+45).

Times above include times for adequate sample recirculation per the requirements of Criterion #1. The total time for sampling and analysis is summarized in Table 1; considering that the 3-hour requirement applies to one sample (i.e., liquid or gas), the total times are well within regulatory limits.

It should be noted that the analysis presented herein assumes the grab samples are taken as the final steps of the sampling exercises. In actuality, the operating sequence does not necessarily have to follow this guideline and grab samples may be taken at earlier times, thus reducing the total time for completion of post accident exercises.

The Clinton Power Station PASS is designed to be powered from Emergency Power within thirty (30) minutes of a loss of offsite power event.

Loads in the PASS are electrically isolated from the diesel generator bus in the event of a LOCA through a shunt trip. Power is restored to the PASS when an operator, through administrative procedures, manually bypasses the LOCA shunt trip signal.

Criterion 2

The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hr time frame established above, quantification of the following:

1. Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and nonvolatile isotopes);
2. Hydrogen levels in the containment atmosphere;
3. Dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.
4. Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Clarification 2

1. A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation to minimize personnel radiation exposures (ALARA). Also, a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:
   a) Monitoring for short and long lived volatile and nonvolatile radionuclides such as 133_{Xe}, 131_{I}, 137_{Cs}, 134_{Cs}, 85_{Kr}, 140_{Ba}, and 88_{Kr} (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

   b) Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.
2. Show a capability to obtain a grab sample, transport and analyze for hydrogen.
3. Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97, Rev. 2.
4. Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See [ed. Criteria] (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Position 2

The Clinton Power Station has established a radiological and chemical analysis capability to quantify specified parameters within the established three (3) hour time frame.

1. Certain radionuclides in the reactor coolant and containment atmosphere are quantified because they may be indicators of the degree of core damage.

   The estimation of core damage is calculated by comparing the measured concentrations of major fission products in either gas or liquid samples, after appropriate normalization with reference plant data from a BWR-6/238 with a Mark III Containment.* This procedure provides locations for obtaining the most representative samples depending on accident severity and system conditions. Water samples (reactor coolant, suppression pool and RHR) and gas samples (containment and drywell) are analyzed by gamma spectroscopy in the lab for determination of I-131, Cs-137, Xe-133 and Kr-85 concentrations. The measured fission products are corrected for decay, and the concentrations are normalized to the reference plant data for comparison to graphs to indicate percent cladding failure, percent fuel overheating, or percent fuel meltdown. Isotopic ratios for noble gases and iodine are calculated for comparison with the ratios that are normally expected to be found in the core inventory and in the fuel gap.

   There are several other plant parameters which are measured in the BWR which can provide information to confirm the initial core damage estimate based on radionuclide measurements.

*The Clinton Power Station Procedure for estimating the degree of core damage is based upon the generic procedure submitted to the NRC by the BWR TMI Owners' Group (BWROG) via the letter to Darrell G. Eisenhut (NRC) from T. J. Dente (Chairman, BWROG) dated June 17, 1983 (letter No. BWROG 8724). This procedure was found acceptable on an interim basis (as noted in the Clinton Power Station Safety Evaluation Report (NUREG-0853), Supplement 2, Section 9.3.5.1) pending incorporation of other plant indicators and discussion of fuel overheating (metal water reaction). Illinois Power Company has revised the plant-specific procedure to include other plant indicators and fuel overheating. The revised procedure has been submitted to the NRC via a letter to the Director of Nuclear Reactor Regulation from F. A. Spangenberg, Director-Nuclear Licensing and Configuration (IPC) dated June 11, 1985. The NRC has indicated their acceptance of this procedure in the CPS Safety Evaluation Report, Supplement 5, section 9.3.5, page 9-3.

Drywell radiation level provides a measure of core damage. It is an indication of the inventory of airborne fission products released from the fuel to the containment. The procedural method involves correlating dose rate time history to the percentage of fuel inventory released which is a function of the core damage scenario involved. Containment hydrogen and oxygen levels, measured by the containment atmosphere monitoring system (CAM), provide a measure of the extent of metal water reaction which can be used to estimate the degree of clad damage. The method involves the use of a correlation which relates hydrogen concentration in the containment to the percent metal-water reaction for Mark III type containments.

The reactor vessel water level is another indicator of core damage. It is used to determine if there has been an interruption of adequate cooling. Significant periods of core uncovery, as evidenced by reactor water level readings and recordings, would be an indicator of a situation where core damage is likely. Water level measurements would be useful in distinguishing between bulk core damage caused by loss of adequate cooling to the entire core, and localized core damage caused by a flow blockage in some portions of the core.

2. Hydrogen levels in the containment atmosphere are quantified (in percent by volume) by inline monitoring via the Containment Atmosphere Monitoring (CAM) System (See Figure 13).

The Clinton Station has two (2) inline H_2/O_2 monitors for the containment and drywell atmosphere. One of the monitors is primary, the second is the redundant backup; therefore, no grab sampling capability is necessary. The system's local equipment is located in the Fuel Building, while the control and indication instrumentation is situated in the Main Control Room.

The H_2/O_2 monitoring subsystem is preprogrammed to analyze (once per day during normal operation) gas samples sequentially from three zones in the drywell and two zones in the containment. In the event of an accident, the system can be started, from the Main Control Room, within 30 minutes. The gas sample lines are heat traced to provide samples representative of containment and drywell atmosphere conditions. While a sample is being analyzed, the sample line is continuously purged with the next gas sample; the purge sample gas and the sample are returned to the Containment. The Containment Atmosphere Monitoring H_2/O_2 System is designed to meet the specific regulatory requirements of Reg. Guide 1.97 and the industry standards listed in Table 2.

3. Following is a discussion of the accident sampling capabilities required by Table 2 of Regulatory Guide 1.97 Rev. 3, and the clarification to Criterion #2.

Primary Coolant

Gross Activity and Gamma Spectrum

The PASS provides diluted and undiluted samples for this analysis.

The grab sample count is carried out at the Radiological Chemistry Laboratory counting room on diluted samples.

The PASS captures a 4 ml grab sample of undiluted, reactor coolant. The liquid sample is first captured in a vessel inside the SAP, at source pressure. Then it is vacuum degassed and routed to an evacuated 4cc bottle. The bottle is recessed in a mobile cask shielded to a surface dose rate of about 150 mR/hr for the design basis accident. The cart has a four foot long handle to reduce operator exposure. In the event of a worst case accident, the bottle can be transported offsite for analysis.

The PASS also provides a 1,000:1 diluted liquid grab sample. The sample is prepared by mixing a 0.023cc "bite" of undiluted, degassed, depressurized liquid with 23cc of demineralized H_2O in a behind-shield, septum equipped mixing chamber. After the sample is diluted, 10cc of it are removed from the chamber by inserting the needle of a shielded syringe (aliquoter) through a plug valve, the panel shield, and the septum. The diluted sample can be analyzed for gross activity and gamma spectrum at the Radiological Chemistry Laboratory.

Position 2

The PASS also provides a grab sample of gases vacuum stripped from a liquid sample. This sample can be taken to the Radiological Chemistry laboratory for nuclide analysis.

Chloride Content

An undiluted reactor coolant sample is analyzed within 96 hours (4 days) at an offsite laboratory. (See position No. 5 for details)

Boron Content

A diluted (1,000:1) reactor coolant sample is analyzed in the Radiological Chemistry Laboratory via a Tetrafluoroborate Selective Ion Electrode (TSIE). (see position No. 7 and 10 for details).

Dissolved Hydrogen

This analysis is performed inline at the SAP by measuring the hydrogen concentration of "RC Off-Gas" via a Gas Chromatograph. The results of the analysis are remotely provided at the SMP (see position No. 4 for details).

Dissolved Oxygen

Dissolved Oxygen (DO) concentration is measured in the Sample Analysis Panel (SAP) using an Orbisphere DO analyzing system in which a cooled, depressurized liquid sample flows past an oxygen probe. The digital readout is remotely mounted in the Sample Monitor Panel (SMP) (see position No. 4 for details).

pH

pH is measured inline by a Leeds and Northrup (L&N) No. 117489 probe. The probe is sealed and has automatic temperature compensation. The monitor is an L&N model No. 7075. The range of the pH analysis is from 1 to 13. The analog readout is remotely mounted in the SMP.

Containment and Drywell Air (CA)

Hydrogen and Oxygen Content

Hydrogen and Oxygen concentration are quantified by inline monitors from the Containment Atmosphere Monitoring (CAM) System (see Position 2 Item 2. for details).

Position 2 (Continued)

Gamma Spectrum

For gamma spectrum measurement, the PASS provides a partitioned containment atmosphere sample. An air sample flows through a device to separate particulates (via a HEPA filter) and iodines (via a silver-zeolite-cartridge), and to collect diluted noble gases (in a noble gas collection flask). The noble gas flask is counted using the Spectral Analysis, ALARA, and Dosimetry Records Keeping System in the Radiological Chemistry Laboratory.

4. The PASS inline monitoring capabilities for dissolved hydrogen, dissolved oxygen, pH (in reactor coolant) and Hydrogen/Oxygen (in containment air) were described above. The following is an evaluation of the instruments.

The Orbisphere dissolved oxygen analyzer is designed to measure the oxygen content of a flowing sample at a temperature between 32°F and 113°F, and a pressure less than 290 psig. Temperature compensation is automatic. The Three Mile Island Station has used an Orbisphere to monitor post accident reactor water and has established a good performance record for the instrument. Typically, the only maintenance required is the replacement of the membrane and electrolyte every three to six months. Calibration is performed using air-saturated demineralized water.
A combination KCl/AgCl gel-type pH electrode is used to measure the liquid sample pH. The probe is a dual element measuring/reference electrode with geometry suitable for pH measurements in high purity BWR water where streaming potential and currents can affect other style electrodes.

The temperature in the system is automatically compensated. The probe is calibrated using 2 buffer solutions contained in calibration tanks located on the front of the SAP. The many power plant applications for these L & N pH components suggest that they are reliable, low maintenance items. Maintenance involves the replacement of modular components.

The two CAM System divisions are electrically and physically separated so that no single design basis event is capable of damaging equipment in more than one division. No single failure or test, calibration, or maintenance operation can prevent function of more than one division. Each CAM system can be tested during plant operation to determine the operational availability of the system components. The system has the capability for test, calibration, and adjustment of the electronics in each channel.

Position 2 (Continued)

Each CAM Hydrogen/Oxygen monitoring subsystem is provided with a calibration gas to check the Hydrogen/Oxygen sensors during normal plant operation and after an accident. The calibration is checked automatically prior to taking any samples during normal operation or after an accident. The CAM system local equipment is designed to be operable during normal and postaccident conditions.

The Spectral Analysis, ALARA, and Dosimetry Records Keeping System used will be capable of acquiring nuclear spectra and identifying constituents of the spectrum as well as quantifying each constituent. The system consists of a ND66 and ND6685 computer system tied in with 2 high efficiency high purity germanium (HpGe) detectors made by EG&G Ortec. The system, as described, is capable of determining isotopic concentrations in liquids, gases, and solids of varying sample sizes and storing this information for future use.

Criterion 3

Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

Clarification 3

System schematics and discussions should clearly demonstrate that post-accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Position 3

Reactor coolant and containment atmosphere sampling during post accident conditions (including recirculation) do not require an isolated auxiliary system to be placed in operation in order to use the Post Accident Sampling System.

System schematic drawing No. M05-1045 (See Figure 8) shows two reactor coolant sampling lines connecting the Reactor Water Cleanup (RWCU) System effluent and the reactor vessel jet pump pressure tap line to the sample panel. During normal plant operation, reactor coolant samples are taken using these lines.

Under accident conditions, the RWCU System lines are isolated and reactor coolant samples are drawn from the Residual Heat Removal (RHR) System sample lines, or until the reactor is depressurized, the reactor vessel jet pump pressure tap lines. The RHR System and the PASS line from the jet pump nozzle sample line are operational post-accident since they are "essential" as classified in response to TMI action plan II.E.4.2, Item 2.

The Containment Atmosphere Monitoring (CAM) system (see Figure 13) is also operational post accident, and classified as "essential." The Clinton Power Station has two inline H_2/O_2 monitors for the containment and drywell. The monitors are 1E qualified (i.e., qualified to satisfactorily function unattended for 100 post-accident days and qualified to the requirements of Reg. Guide 1.97).

PASS employs a minimum number of valves (1PS004, 1PS005, 1PS009, 1PS010, 1PS016, 1PS017, 1PS022, 1PS023, 1PS031, 1PS032, 1PS034, 1PS035, 1PS037, 1PS038, 1PS043A, 1PS043B, 1PS044A, 1PS044B, 1PS047, 1PS048, 1PS055, 1PS056, 1PS069, 1PS070) which are inaccessible for repairs after an accident. These isolation valves are environmentally qualified to withstand the post accident conditions in which they will operate. These valves are qualified in accordance with FSAR Section 3.11.

Criterion 4

Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification 4

Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is less than 0.1 ppm by measurement of a dissolved hydrogen residual of greater than or equal to 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Position 4

The Clinton Power Station has the capability to take pressurized and unpressurized reactor coolant (RC) samples, and quantify the amount of dissolved gases in each.

The reactor coolant sample for the PASS is obtained from a reactor jet pump instrumentation sensing line until the reactor is depressurized. After the reactor is depressurized, the reactor coolant sample is taken from either RHR pump A or B discharge to assure that a sample representative of the core condition is obtained.

The hydrogen concentration in a RC sample is quantified inline by measuring the H_2 content of RC off-gas (i.e., the gas stripped from a cooled sample of liquid captured at full reactor system pressure). The sample is routed to a Baseline Model 1030A Gas Chromatograph (GC). The detector converts the volume of gas present to a proportional electronic signal which is amplified and recorded in a chromatogram format. The chromatogram is then compared to calibration curves to determine the hydrogen concentration.

The dissolved oxygen (DO) content is measured inline with an Orbisphere DO analyzing system. The oxygen display signal is continuously available at a recorder output connection.

Position 4 (Continued)

The Orbisphere DO analyzing system will measure from 0 to 20 mg/l (20 ppm); therefore, the system has the capability to determine that DO concentration is less than 0.1 ppm in the event that chlorides exceed 0.15 ppm.

The Orbisphere is an online instrument and will yield accurate measurements within minutes after flow through the sample cell has begun; therefore, the DO concentration can be determined with the very first sampling exercise (40 minutes), well within the 30 day requirement.

The direct monitoring capability for DO concentration results in minimum radiation exposure to plant personnel, as described in Position 6 of this report.

Criterion 5

The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions, the licensee shall provide for a chloride analysis within 24 hr. of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification 5

BWRs on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers (e.g., shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hr. All other plants have 96 hr to perform a chloride analysis. Samples diluted by up to a factor of 1,000 are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system, and (2) that dissolved oxygen can be verified at 0.1 ppm, consistent with the guidelines above in Clarification No. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Position 5

An undiluted reactor coolant sample will be analyzed at an offsite facility within four (4) days; since Clinton Power Station is a freshwater cooled plant, four (4) days are allowed for this analysis to be completed. In the event of a minor accident (sample activity is less than 1/1,000 of the worst case activity), a liquid sample may also be analyzed onsite via an Ion Chromatograph.

Dissolved oxygen concentration can be verified to be less than 0.1 ppm (See Position 4).

Criterion 6

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). Note that the design and operational review criterion was changed from the operational limits of 10CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979, letter from H. R. Denton to all licensees).

Clarification 6

Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Position 6

The design basis for post-accident sampling system equipment for reactor coolant (RC) and containment atmosphere (CA) sampling and analysis assures that it is possible to obtain and analyze a sample without radiation exposure to any individual exceeding 5 Rem whole body and 75 Rem extremities.

The Clinton Power Station post accident sampling man-rem exposure is predicted to be substantially less than the maximum radiation exposure limits. The following paragraphs provide estimated dose rates that will be received by the operator(s) during transport and analysis. The dose values are based on the operator's inability to use reach rods to manipulate valve handles during the early hours following a worst case accident. The radiation source terms are for: fluids and gases per Table 3 at one hour after the accident.

The predicted activity for samples obtained at the SAP is listed on Table 4.

These source terms are a result of accident conditions as specified by the NRC in Reg. Guide 1.3.

A. Transporting Samples

1. Diluted RC Off-Gas (DG-1)

The maximum possible time required to transport the sample from the SAP panel to the onsite counting room is 20 minutes. The sample is contained in a vial inside a tong. The whole body dose received is 6.5 mRem. The extremities dose for this time period is 13.5 mRem.

2. Undiluted RC (L-1)

The sample is contained in a mobile shielded cask (Figure 14). The cask's shield is 4.7 inches of solid lead; its weight is 500 lbs. A worker will tow the mobile cask to a vehicle for offsite analysis using its 4 ft. long handle

Assuming the worst possible case, the worker takes 3 hrs. to tow the cask. The whole body dose received is 7.4 mRem, and the dose for the extremities is also 7.4 mRem.

3. Diluted RC (DL-1)

The sample to be transported to an onsite counting room is contained in a syringe. The maximum transport time is 20 minutes.

The whole body dose for this time period is 10.6 mRem; the extremities dose is 31 mRem.

4. Diluted CA (CA-1)

The diluted sample is contained in a glass vial inside a 26" long tong (Figure 15). This assembly is transported to an onsite counting room in 20 minutes. The radiation dose for this transport time is 1.1 mRem for the whole body, and 2.2 mRem for the extremities.

B. Chemical Analyses

1. Boron

The analysis time is about 15 minutes and the size of diluted liquid sample to be used is 5cc. While doing the analysis, the sample can be maintained at a 15cm distance from hands and 40cm from the body.

The radiation exposure to an operator is 11 mRem for the whole body and 77 mRem for the extremities.

2. Chloride

This analysis will be performed within 4 days of the sampling exercises at an offsite facility. In this case, no exposure will result to plant personnel.

In the event of a minor accident (sample activity is at or below 1/1,000 of the worst case activity), a liquid sample may also be analyzed onsite via an ion chromatograph. After the reactor coolant sample is injected into the ion chromatograph (IC), the operator moves away from the IC while the analyses are carried out. Thus, the only significant radiation exposure is caused by the sample injection process. The sample can be injected into the IC in 120 seconds; the minimum body distance for this process is 15cm while the hands will be at 5cm from the sample.

The radiation exposure (with a 6cc RC sample) is 6mRem for the whole body and 54mRem for the extremities.

3. Nuclide Analyses

Nuclide analysis will be performed via counting of grab samples in the Radiological Chemistry laboratory counting room. The dose to the hands is 69 mRem; 31 mRem whole body.

To the above radiation doses, it is necessary to add the extra dose for the PASS exercises. The dose for such exercises is 1,305 mRem for hands and 171 mRem for whole body.

Additional exposure is incurred due to radiation levels present in the PASS panel area (after sampling is completed), the route to the Rad Chem laboratory and the laboratory itself. This radiation is caused by piping, equipment and penetrations present in these areas. Ambient post-accident radiation levels have been determined during the post-accident shielding evaluation performed for Clinton Power Station in response to NUREG-0737 Requirement II.B.2.

The radiation levels for the subject areas are 0.1 to 1 Rem/hr in the PASS panel area, 0 to 15mRem/hr for the Rad Chem laboratory and the route to it (reference Figure 16). Assuming 20% occupancy in the PASS panel area for the 1.5 hours after sampling, the maximum radiation exposure dose to an individual is 0.3 Rem. The maximum required time for transporting samples to the Rad Chem laboratory is 20 minutes and the worst possible total dose (transporting 4 samples) is 20 mRem. The longest time period spent in the laboratory for post-accident analysis is 3 hours, and the worst possible dose resulting is 45mRem.

Position 6 (Continued)

The above information is summarized in Table 5.

From this table, assuming the worst possible case (one person performing all post-accident functions), the dose for the Clinton post accident sampling, transporting and analyses is substantially less than the maximum radiation exposure limits.

The above transport and sampling times are conservative in order to account for possible abnormalities which may arise during sampling/transport/analyses. Additionally, the dose to any one individual will be substantially less since more than one person will be involved in the post-accident procedures.

Criterion 7

The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants.)

Clarification 7

PWRs need to perform boron analysis. The guidelines for BWRs are to have the capability to perform boron analysis, but they do not have to do so unless boron was injected.

Position 7

Boron analysis will be performed in the Radiological Chemistry Laboratory via a Tetrafluoroborate Selective Ion Electrode (TSIE).

Criterion 8

If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Clarification 8

A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an offsite laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Position 8

The Clinton Power Station utilizes inline monitoring for: 1) dissolved hydrogen, dissolved oxygen, pH, and conductivity in primary coolant samples; and 2) hydrogen and oxygen concentration in containment air samples.

The following is a discussion of the backup samples and procedures to analyze them:

A. Reactor Coolant

1. Dissolved Hydrogen Concentration

It is Illinois Power's position that backup analysis capability of grab samples for dissolved Hydrogen (H_2) content is not necessary.

Dissolved (H_2) content in the reactor coolant (RC) can be related to the extent of metal-water reaction and subsequently to the degree of clad damage. However, since over 95% of the H_2 in RC is released to the Containment atmosphere regardless of pressure (Reference GE Letter MFN-006-84, FRH-003-84 to the U. S. Nuclear Regulatory Commission dated January 18, 1984), dissolved H_2 in RC data is not utilized in the Core Damage Estimation Procedure section concerning Metal-water reaction (Reference Emergency Plan Implementing Procedure No. EC-13, Rev. 1, dated 11-27-85). The H_2 quantity used in this iteration is taken from the Containment Atmosphere Hydrogen Analyzers.

Position 8 (Continued)

Hydrogen concentration in RC can also be related to RC corrosion potential. The major variables to be monitored for assessing RC corrosion potential are Chloride, pH and Oxygen; the verified absence of either Chloride ($<$0.1 ppm) or Oxygen ($<$0.1ppm) will practically eliminate this concern. Additionally, if pH $<$ 7.0, RC corrosive potential is further reduced. It is the NRC's position (as stated in the clarification to Criterion No. 4), that the measured presence of a dissolved H_2 residual of $\geq$ 10cc/kg of RC is an acceptable verification that dissolved Oxygen (O_2) is $<$ 0.1ppm, however, O_2 must be measured directly within 30 days of an accident.

The Clinton Power Station Post Accident Sampling System design provides for inline measurement of O_2 in RC (see Position No. 4). This direct measurement provision for O_2 in RC questions the need for measurement of dissolved H2; per the above, dissolved H_2 is <u>not</u> a primary post accident quantity since it is not utilized in any of the post accident analyses. It is IP's position that this quantity could be used as a viable backup to the dissolved O_2 analysis (in addition to the primary "backup" described in section A(2) of this position).

2. <u>Dissolved Oxygen and Conductivity</u>

The backup analyses are performed at an offsite facility on an undiluted liquid sample.

3. <u>pH</u>

The backup samples are: 1) undiluted liquid (4cc of depressurized, degassed liquid in a mobile cask); and 2) diluted liquid (6cc of 1,000/1 diluted, depressurized, degassed liquid in a shielded syringe).

For this analysis, an undiluted RC sample should be used for any lab measurement. In fact, a 1,000 to 1 diluted liquid sample is predicted to cause significant quantification errors due to large pH uncertainty results when extrapolating diluted results to undiluted conditions.

Position 8 (Continued)

The undiluted sample is too "hot" at 2 hours after a Design Basis Accident for onsite lab facilities. Therefore, this analysis will be performed at an offsite hotroom facility.

B. Containment Air (CA)

1. Hydrogen and Oxygen

The primary monitor for Containment and Drywell Air inline analysis (as described in Criterion #2) has a backup via a redundant 1E qualified monitor.

The PASS is designed to adequately perform one sampling exercise per day for a minimum of 182 days following the accident.

Operating procedures for PASS require flushing of all sample flow paths, and sample wetted monitors and components after each exercise, in order to minimize contamination and facilitate maintenance. This flushing solution (demineralized water) is sent to the Fuel Building Floor Drain tank during testing/training/normal operation and to the suppression pool during accident conditions.

For transporting samples offsite, prior arrangements will be made to have a shipping container (cask) sent from the offsite laboratory or have one available on-site. The current intent is to have a shipping container available which will hold the large volume cask (undiluted) thus avoiding the exposure which would result from transferring the sample from the sampling cask to another container.

Criterion 9

The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification 9

(a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also, provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

Position 9

(a) The dilution process is performed manually at the Sample Analysis Panel by the operator. Dilution factors up to 1,000: 1 are achievable. The liquid sample is prepared by mixing 0.023cc (one "bite") of undiluted, degassed, depressurized liquid with 23cc of demineralized water.

The predicted activities of diluted and undiluted samples are listed on Tables 3 and 4.

The Spectral Analysis, ALARA, and Dosimetry Records Keeping System used in the Radiological Chemistry laboratory will be capable of acquiring nuclide spectra and identifying constituents of the spectrum as well as quantifying each constituent. The system consists of a ND66 and ND6685 computer system tied in with 2 high efficiency high purity germanium detectors made by EG & G Ortec. The system as described is capable of determining isotopic concentrations in liquids, gases, and solids of varying sample sizes and storing this information for future use. By varying the distance between the detector and the radioactive source, samples with activity ranges from 1 uCi/cc to 10 Ci/cc can be analyzed.

The laboratory counting equipment will also be used for routine analysis.

b) The background radiation levels in the counting room are predicted to be a maximum of fifteen (15) mr/hr as determined during the post-accident shielding evaluation performed for Clinton Power Station in response to NUREG-0737 requirement II.B.2. As a sample is being counted, other reactor coolant samples will be located outside of the counting room, therefore, they will not cause any increase in background radiation. Additionally, the counting cave is shielded by a 4.5" layer of lead to prevent any additional radiation present in the room from interfering with the counting of samples.

The background radiation should not have any effect on the counting accuracy. The normal practice is to count for a 95% confidence level.

Criterion 10

Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Clarification 10

The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

(a) Gross activity, gamma spectrum: Measured to estimate core damage, these analyses should be accurate within a factor of 2 across the entire range.

(b) Boron: Measured to verify shutdown margin.

In general, this analysis should be accurate within ±5% of the measured value (i.e., at 6,000 ppm B the tolerance is ±300 ppm while at 1,000 ppm B the tolerance is ±50 ppm). For concentrations below 1,000 ppm, the tolerance band should remain at ±50 ppm.

(c) Chloride: Measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride, the analysis should be accurate within ±10% of the measured value. At concentrations below 0.5 ppm, the tolerance band remains at ±0.05 ppm.

(d) Hydrogen or Total Gas: Monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of ±10% is desirable between 50 and 2000 cc/kg but ±20% can be acceptable. For concentration below 50 cc/kg, the tolerance remains at ±5.0 cc/kg.

(e) Oxygen: Monitored to access coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen, the analysis should be accurate within ±10% of the measured value. At concentrations below 0.5 ppm, the tolerance band remains at ±0.05 ppm.

Clarification 10 (Continued)

(f) pH: Measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within ±0.3 pH units. For all other ranges, ±0.5 pH units is acceptable. To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX FOR UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituent	Nominal Concentrations (ppm)	Added as (chemical salt)
I-	40	Potassium iodide
Cs+	250	Cesium nitrate
Ba+2	10	Barium nitrate
La+3	5	Lanthanum chloride
Ce+4	5	Ammonium cerium nitrate
Cl-	10	
B	2000	Boric acid
Li+	2	Lithium hydroxide
No^{-}_{3}	150	
NH_4^+	5	
K+	20	
Gamma radiation (induced) field)	10^4 rad/gm of reactor coolant	Adsorbed dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

Clarification 10 (Continued)

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedures have been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post-accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every 6 months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

Position 10

Table 6 summarizes the ranges and accuracies for each chemical procedure or online instrument used for post-accident exercises at Clinton Power Station.

The following is a discussion of the accuracy of the instruments used to perform the required analyses:

A. Gross Activity and Gamma Spectrum

The Spectral Analysis, ALARA, and Dosimetry Records Keeping System will allow counting within a factor of 2 across the entire range. The normal practice for this system is to count at a 95% confidence level. The range for this analysis will be 1 uCi/cc to 10 Ci/cc.

B. Chloride in RC

The range for chloride is 0 to 20 ppm.

Ion chromatography (IC) gives accurate results. The accuracy of the IC measurements is ± 15% in the 0.1 to 1.0 ppm chloride range and is ± 25% for higher concentrations. By calibrating at higher concentrations, the accuracy can be maintained at ±15%.

The advantages of the procedure are the measurement range for chloride, normal and accident usage, small sample size, the lack of chemical interference, remote operability, simplicity of operation, and potentially short analysis time. This instrumentation sees no special effects during post accident conditions.

C. Boron in RC

This analysis is carried out via a Tetrafluoroborate Selective Ion Electrode. The Selective Ion Electrode has the capability of quantifying boron in the 0.5 to 6 ppm range on a direct measurement. In the event of a worst case accident, a diluted (1,000:1) reactor coolant sample would be analyzed; the overall range would then be 500 to 6,000 ppm. Testing at the site laboratory indicates good results in the 500 to 1500 ppm range with accuracies within ±10%. The advantages of this procedure are its wide measurement range and accuracy, the small sample size required, the lack of chemical interferences, its adaptability to routine and accident condition usage and the short analysis time required. No post accident radiological effects are anticipated.

D. Dissolved H_2 in RC (via PASS Inline Analyzer)

H_2 dissolved in RC is quantified by measuring the H_2 concentration of RC Off-Gas via a Baseline Model 1030A Gas Chromatograph (GC).

1. Range = 10 to 2,000 cc/Kg

2. Accuracy = ± 15% of reading

The Baseline GC accuracy does not verbatim comply with clarification requirements at 50 cc/Kg and below. However, these accuracies are sufficient for dissolved gas measurement in the BWR. At low pressure following an accident, the gases evolved from the core will be essentially all released to the CA, and the dissolved gas concentrations will not be useful. For high pressure conditions following an accident, 1) the mass of hydrogen in the reactor pressure vessel is a small percentage of the total H_2 generated by the metal-water reaction due to discharge of the safety/relief valves to maintain system pressure. Then, even if the dissolved gas contribution is totally ignored under high pressure conditions, the absolute error in the total H_2 release for estimation of core damage would be very minimal.*

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* This subject of dissolved gas measurement accuracy for BWRs was discussed in detail at a meeting between the NRC and General Electric on December 12, 1983 and is documented in the letter to D. G. Eisenhut (NRC) from G. S. Sherwood (GE) dated January 18, 1984. The NRC has approved GE's position on dissolved gas system adequacy.

Position 10 (Continued)

The Baseline 1030 A is a process GC which has over a 7 year reliability record in its two largest fields of use (oil well logging and industrial/environmental gas monitoring).

The gases wetting the stainless steel and plastic surfaces in the GC are vacuum stripped from the RC. The gases are substantially free of the chemicals in the standard test matrix and are predicted to have no adverse effect on the GC life and performance.

The PASS Baseline GC is separated into 2 modules: Analysis and Control.

a. The Analysis module has its components wetted by radioactive samples. This module is located in the Sample Analysis Panel, behind the radiation shield. The dose to the wetted component is estimated to be 5120 Rad, Total Integrated Dose (TID). The sample wetted components are stainless steel and plastic. It is estimated that these components will satisfactorily function up to 50,000 Rad, TID.

b. The Control Module, containing the microprocessor controller, is located in the Sample Monitor Panel. The field radiation in this area is predicted to be less than 1,000 R, TID. As a generic guideline, microprocessors will satisfactorily survive 2,000 Rad, TID.

E. Oxygen in RC (Via PASS Inline Analyzer)

Dissolved O_2 (DO) in RC is quantified by a Polarographic sensor. The DO analyzer is an Orbisphere #2713 system. The range of the Orbisphere #2713 is 0 to 20 ppm.

The DO instrument accuracy is ± 5% of reading (or ± 0.005 ppm, whichever is greater), thus complying to the clarification requirements (reference Orbisphere laboratories catalog "Model 2713 Dissolved Oxygen Measurement System") The DO accuracy can be reduced when: 1) a pressurized RC sample contains substantial dissolved hydrogen; and 2) the sensor is located downstream of a depressurizing valve. However, the clarification to Criterion #4 indicates that the maximum dissolved O_2 will be 0.1 ppm when dissolved H_2

is 10cc/Kg and greater. At this condition, the DO sensor will measure DO in depressurized RC with accuracy of + .005 ppm/-.016 ppm which surpasses the ± .05 ppm requirement of the clarification to Criterion #10.

The Orbisphere #2713 system has an excellent reliability record in fossil and nuclear power plants. TMI is presently using an Orbisphere to monitor post accident RC.

Orbisphere construction and wetted materials are essentially identical to those of a Yellow Springs Instrumentation (YSI) DO probe and cell holder. The YSI performance was not compromised by a post accident test matrix which was chemically similar to that specified in the clarification (reference NUS Corporation report, April 1981, to Sentry Equipment Corporation, entitled "Development of Procedures and Analysis Methods for Post Accident Reactor Coolant").

The worst radiation dose to the Orbisphere sensor is predicted to be .318 Megarads TID; testing has demonstrated that the sensor functioned satisfactorily to 3.5 Megarads TID .

F. pH in RC (Via PASS Inline Analyzer)

pH is quantified by using a Leeds & Northrup (L&N) #117489 probe. The probe is sealed and has automatic temperature compensation. The Monitor is L & N #7075. The pH probe is calibrated, in-place, using 2 buffer solutions from tanks located on the front of the SAP.

a. Range = 1 to 13

b. Accuracy = ±0.21 units (Assuming a 10°C difference in temperature between calibration solution and PASS liquid temperature) (reference: 1) NSAC/46 dated 4/82 and 2) L & N Publications #1211-D5 and #177924, Rev. E)

Virginia Electric Power and Commonwealth Edison have demonstrated that a post accident chemical matrix (similar to that specified in the clarification to Criterion #10) had no degrading effect on the performance of an L&N #117489 pH probe. The probe was used in a Sentry Equipment Corporation (SEC) modified Ionics Digichem Boron Analyzer, to measure pH inflection (reference SEC Report #12490, Rev.0).

Postion 10 (Continued)

The worst radiation dose to the pH probe is predicted to be .318 Megarads TID. EPRI testing demonstrated that 27 Megarads TID had no significant effect on the performance of an L & N #117489 pH probe (reference NSAC/46 dated 4/82).

Sentry Equipment Corporation has performed tests demonstrating that SAP and SMP components (including inline instruments) will successfully survive their design total dose criteria with adequate safety factors.

Inline instruments in the PASS panel will be calibrated according to the frequency prescribed in the vendor manual. Laboratory equipment will be calibrated according to plant procedure CPS No. 6000.01, Quality of Chemistry Activities.

A training program for personnel involved in maintenance/ operation of the PASS will be established at CPS according to CPS Technical Specification, Section 6.8.4.c. The program will include instructions on important system components. After initial qualification, requalification will occur every 2 years, as a minimum. Additionally, refresher training on the system's operating characteristics will occur on a semiannual basis through the continuous usage of PASS for routine analysis and drills. Evaluation for requalification will involve written and performance testing. Evaluation for refresher training will involve performance testing.

Criterion 11

In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

Clarification 11

(a) A description of the provisions which address each of the items in Clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e., sampling from a hot or cold leg loop which may have a steam or gas pocket), describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWRs should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal adsorbers and HEPA filters.

Position 11

PASS liquid sampling lines are purged and flushed at approximately 7 times the normal sampling flow rate before and after each sampling to reduce contamination of samples when switching from one sample to another. The design flow rate is in the turbulent zone, thus maintaining a representative sample and minimizing blockage and plateout. The liquid sample residue and flush water are returned to the suppression pool via a dedicated return line. The small size of the sample lines limits the loss of reactor coolant in the event of a sample line rupture. The gas sample lines are heat traced to prevent vapor condensation (from the containment atmosphere) and resultant loss of iodine. Those sample lines are purged with nitrogen which is exhausted back to the containment or the drywell.

The reactor coolant samples are obtained from a tap off the jet pump pressure instrument system.* In order to ensure that these samples are representative of core conditions, it is necessary to provide sufficient core flow to circulate water from the core to the jet pump intake.

After a small break or non-break accident, the reactor water level is maintained at or near the normal level by the operator using emergency procedures. For decay power above 1% rated power, the core flow is estimated to be greater than 10% rated flow due to natural circulation. The entire reactor water inventory would be circulated through the jet pumps in about 3-4 minutes, thus providing a representative sample at the jet pumps.

At decay power levels less than 1% rated flow, the reactor water level is raised 18 inches, thereby fully flooding the moisture separators. This provides a thermally induced recirculation flow path for mixing.

Make up water does not significantly dilute the sample. The make up water flow is approximately 2% of the core flow for small steam line breaks and 18% for small liquid line breaks. Thus, there is no significant dilution of the reactor coolant sample.

In addition, sample lines in the RHR system provide for a reactor coolant sample when the reactor is depressurized, and at least one of the RHR loops is operating in the shutdown cooling mode.

*Providing representative reactor coolant samples via the jet pump pressure instrument tap was described in the Licensing Review Group (LRG)-II position paper 1-CHEB dated March 12, 1982. This paper was found acceptable as indicated in the Clinton Power Station Safety Evaluation Report (NUREG-0853), Supplement 2, Section 9.3.5.2.

In the event of a larger line break (in this case, reactor water level cannot be maintained) reverse flow is provided through the core to the suppression pool. Suppression pool samples are obtained from the RHR pump discharge. To ensure a representative sample, the RHR pumps will be operated for approximately 30 minutes prior to taking a sample.*

The PASS sampling lines external to the analysis panel have been designed according to ASME Section III (safety-related piping) and ANSI Standard B31.1 (non safety-related piping).

The lengths of the sample lines outside of the PASS panel are listed in Table 7. The liquid and gaseous sample line bores meet the criteria recommended in Sentry Specification B10-01 (ref. SEC Report #12490, Rev.0). The liquid lines were designed to be as short as possible to minimize the volume of fluid to be taken from containment.

Passive flow restrictors are not utilized. PASS sampling lines have redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage. The automatic containment isolation valves close on containment isolation and safety injection signals (Reactor Vessel Water Level 2).

To minimize sample plateout, PASS in-panel liquid sample lines are designed with .17 inch diameter bore tubing. This size provides the optimum trade-off between pressure loss and turbulence. High velocity ensures a contemporary sample and minimizes plateout.

PASS in-panel lines for containment atmosphere samples are designed according to Appendix B of ANSI N13.1-1969 (Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities). Measures include: 1) smallest bore and highest sample flowrate to obtain the maximum sample velocity; 2) use of long-sweep elbows (radius of bend approximately 10 diameters) to avoid particle deposition in tubing bends; 3) use of plug and ball valves (orifice = linebore) to avoid sharp expansions/contractions which increase turbulent deposition of particulate iodine. These containment atmosphere sample lines are also heat traced internally to the SAP.

*Providing representative suppression pool samples was described in the Licensing Review Group (LRG)-II position paper 2-CHEB dated March 12, 1982. This paper was found to be acceptable as indicated in the Clinton Power Station Safety Evaluation Report (NUREG-0853), Supplement 2, Section 9.3.5.2.

Position 11 (Continued)

Flow paths (liquid and gaseous) are designed without dead legs to prevent 1) crud deposition; and 2) cross-contamination between sampling exercises.

Flowpaths and components (containing radioactive samples) in the Sample Analysis Panel are housed in a plenum, behind a radiation shield. The plenum's effluents are routed through the Auxiliary Building HVAC system during normal operation and through the Drywell Purge System post-LOCA. The latter ventilation system contains charcoal adsorbers and high-efficiency particulate air (HEPA) filters. The plenum is maintained at a negative pressure (about 0.25 inches of H_2O), so that any gaseous leakage is contained within the plenum and routed to the Drywell Purge system.

TABLE 1

SAMPLING AND ANALYSIS TIMES

SAMPLE	ANALYSIS	METHOD	ELAPSED TIME (minutes)
Liquid	Dissolved H_2	Inline PASS analyzer	76
Liquid	Dissolved O_2	Inline PASS analyzer	40
Liquid	pH	Inline PASS analyzer	40
Liquid	Conductivity	Inline PASS analyzer	40
Liquid	Boron	Tetrafluoroborate Selective Ion Electrode	119
Liquid	Nuclides	Sample & Onsite counting lab	134
Liquid	Chloride	Ion Chromatography at onsite lab.	Within 4 days
CA	Nuclides	Sample & Onsite counting lab	179
Maximum Total Time (For RC *and* CA)			179
Criterion Maximum			180 (For RC *or* CA)

TABLE 2

CONTAINMENT ATMOSPHERE MONITORING SYSTEM

Industry Standards Applicability

IEEE Standards and Editions

279 (1971)
323 (1974)
334 (1974)
336 (1971)
338 (1977)
344 (1975)
379 (1972)
383 (1974)
384 (1974)
622 (1979)

TABLE 3

SAMPLE RADIOCHEMISTRY DESIGN PARAMETERS
@ 1 HR AFTER AN ACCIDENT

1	RC LIQUID NOT DEGASSED: GAMMA	
	(a) R/hr @cm/cc*	17,495
	(b) Ci/cc	3.09
2	RC LIQUID DEGASSED: GAMMA	
	(a) R/hr @ cm/cc	13,784
	(b) Ci/cc	1.61
3	RC OFF-GAS: GAMMA	
	(a) R/hr @ cm/cc	3,711
	(b) Ci/cc of RC	1.48
4	DRYWELL ATMOSPHERE: GAMMA	
	(a) R/hr @ cm/cc	618
	(b) mCi/cc	101

* Radiation level for 1cc of sample at 1cm.

TABLE 4

PREDICTED ACTIVITIES OF GRAB SAMPLES @ 1 HR AFTER AN ACCIDENT

1. Undiluted Reactor Coolant, Degassed (L-1)

Sample Size = 4cc

R/hr @ 1cm/cc	13,784
R/hr @ 1cm; whole bottle	55,136

2. Diluted (1,000:1) RC (DL-1)

Sample Size - .023cc RC + 23cc of demineralized H_2O

R/hr @ 1cm/cc	13.784

3. RC Off Gas

R/hr @ 1cm; whole bottle	85.4

4. Drywell Atmosphere (DA-1 or PCA-1)

R/hr @ 1cm; whole bottle	14.2

TABLE 5

PERSONNEL RADIATION EXPOSURE PREDICTIONS

ACTION		EXPOSURE (mR)	
		hands	body
PASS Exercises		1305	171
Ambient contribution (after sampling)		300	300
Transporting Samples	Diluted RC off-gas	13.5	6.5
	Undiluted RC	7.4	7.4
	Diluted RC	31	10.6
	Diluted CA	2.2	1.1
Ambient contribution		20	20
Analysis of Samples	Chloride(Worst Case)	54	6
	Boron	77	11
	Nuclides	69	31
Ambient contribution		45	45
TOTAL (Worst Case)		1924mR(1.9 Rem)	609mR(0.6 Rem)
Criterion Maximum		75 Rem	5 Rem

TABLE 6

CHEMICAL ANALYSIS CAPABILITY

Analysis	Method/Instrument	Range	Accuracy
Gross Activity	Spectral Analysis System (in the laboratory)	1u Ci/cc to 10 Ci/cc	± 50%
Gamma Spectrum	Spectral Analysis System (in the laboratory)	1u Ci/cc to 10 Ci/cc	± 50%
Boron	Tetrafluoroborate Selective Ion Electrode (in-lab)	500-1,500 ppm	± 10%
Chloride	Ion Chromatograph (in-lab)	0-20 ppm In Lab	± 15% (0.1 to 1.0 ppm) ± 25% (1.0 to 20 ppm)
Hydrogen	Gas Chromatograph (inline)	10-2,000cc H_2/Kg	± 15%
Dissolved Oxygen	Dissolved Oxygen Probe (inline)	0-20 ppm	± 5%
pH	pH Probe (inline)	1-13	± 0.21 units
Specific Conductivity	Conductivity Cell	0.1-100 umhos/cm	± 5%

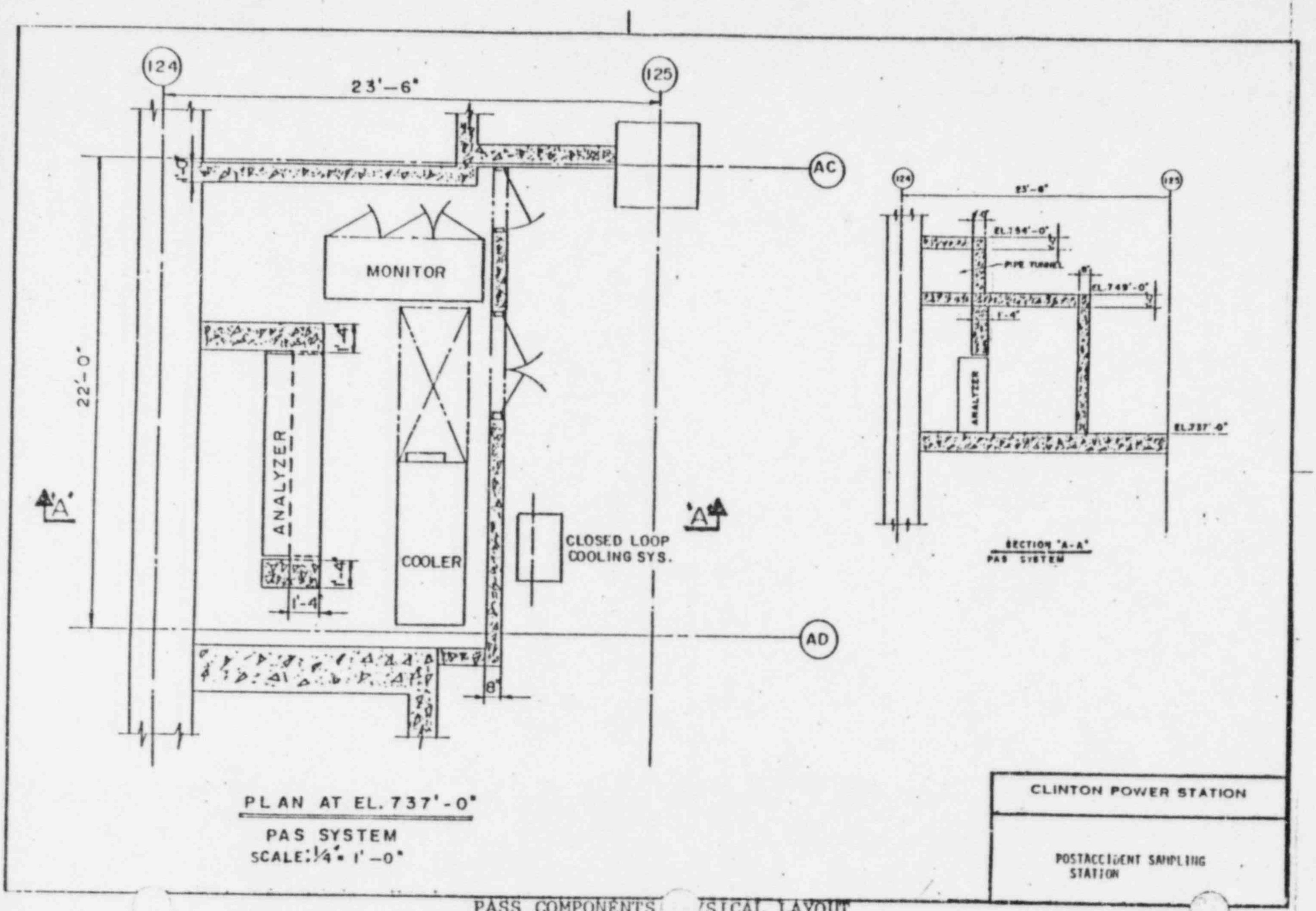
TABLE 7

POST ACCIDENT SAMPLING SYSTEM
LENGTHS AND INSIDE DIAMETERS OF SAMPLE TUBING AND PIPING

Connection # on Panel 1PS02J	Total Length[1] ft	in	Run Length ft	in	Type and Identification	Inside Diameter in
7	287	4				
			147	8	3/8" SS Tube (1PS71C,E 3/8)	0.245
			7	10	1/2" SS Pipe (1PS71D 1/2)	0.464
			87	10	3/4" SS Pipe (1PS71B,F 3/4)	0.612
			113	2	1/2" SS Pipe (1PS71A 1/2)	0.546
2	220	7				
			147	0	3/8" SS Tube (1PS75C,E 3/8)	0.245
			8	8	1/2" SS Pipe (1PS75D 1/2)	0.464
			14	0	3/4" SS Pipe (1PS75B,F 3/4)	0.612
			50	11	1/2" SS Pipe (1PS75A 1/2)	0.546
4	316	3				
			150	11	3/8" SS Tube (1PS58C,E,G 3/8)	0.245
			7	0	1/2" SS Pipe (1PS58D 1/2)	0.464
			146	3	1/2" SS Pipe (1PS58A 1/2)	0.546
			12	1	3/4" SS Pipe (1PS58B,F 3/4)	0.612
6	288	1				
			153	0	3/8" SS Tube (1PS60C,E,G 3/8)	0.245
			7	0	1/2" SS Pipe (1PS60D 1/2)	0.464
			114	0	1/2" SS Pipe (1PS60A 1/2)	0.546
			14	1	3/4" SS Pipe (1PS60B,F 3/4)	0.612
3	112	1				
			60	0	3/8" SS Tube (1PS62D,E 3/8)	0.245
			47	5	1/2" SS Pipe (1PS62B 1/2)	0.546
			4	8	1/2" SS Pipe (1PS62C 1/2)	0.464
5	119	3				
			56	2	3/8" SS Tube (1PS64D,E 3/8)	0.245
			59	10	1/2" SS Pipe (1PS64B 1/2)	0.546
			3	3	1/2" SS Pipe (1PS64C 1/2)	0.464
1	503	7[2]				
			494	9	3/8" SS Tube (1PS72F, H, CB 3/8)	0.245
			6	10	3/4" SS Pipe (1PS72E,AB,BB,DB 3/4)	0.612
			2	0	1/2" SS Pipe (1PS72G 1/2)	0.464
10	241	9[3]				
			76	1	1/2" SS Pipe (1PS67B,AB 1/2)	0.546
			155	0	3/8" SS Tube (1PS67D, 3/8)	0.245
			10	8	3/4" SS Pipe (1PS67C 3/4)	0.612
11	119	11[4]				
			51	0	1/2" SS Pipe (1PS68B,AB, 1/2)	0.546
			60	0	3/8" SS Tube (1PS68D 3/8)	0.245
			8	11	3/4" SS Pipe (1PS68C 3/4)	0.612

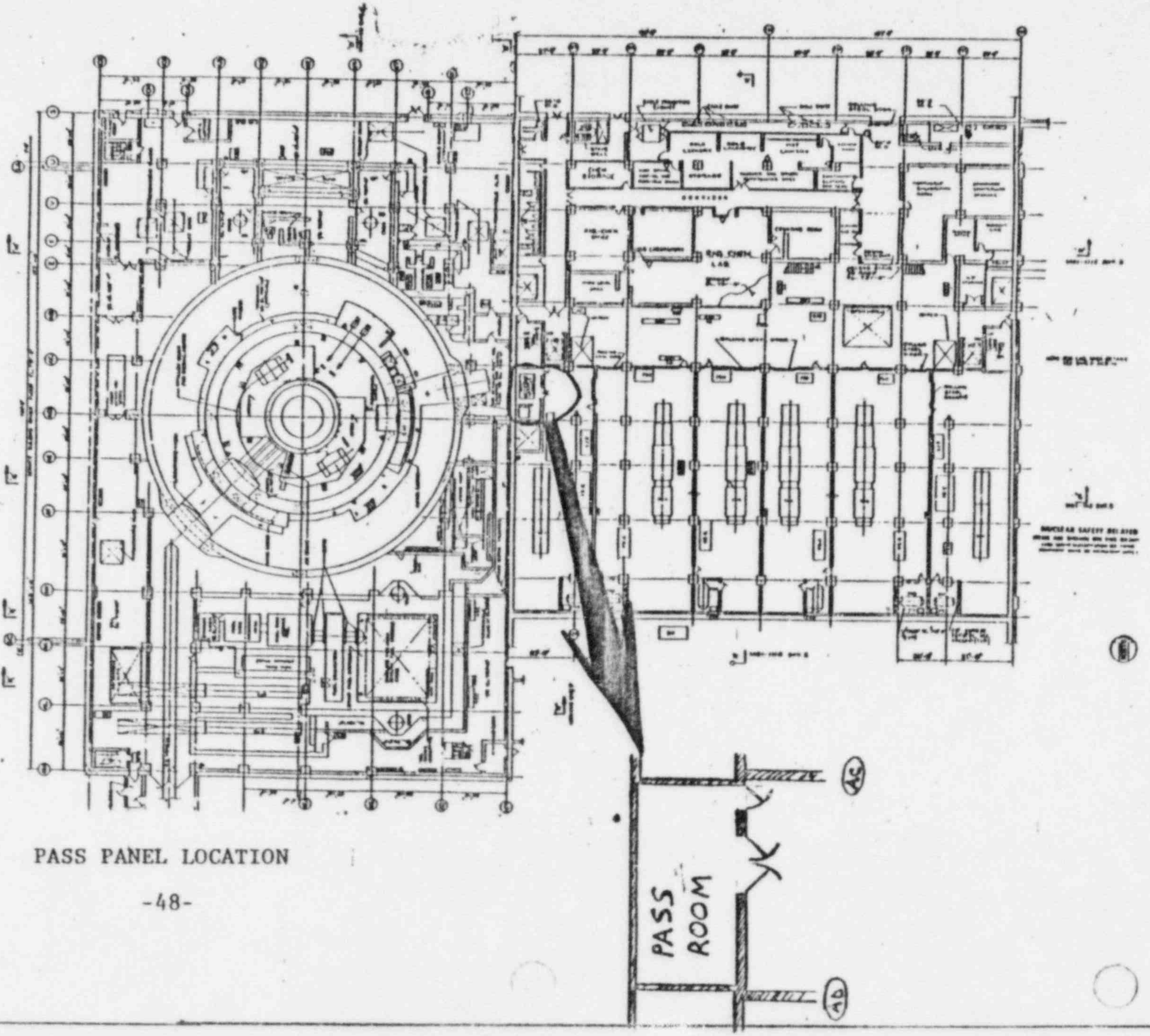
NOTE:
1. Totals for containment, fuel handling, diesel generator buildings.
2. Total length of longest run to pump "B".
3. Longest run, (1PS67AA 3/8, short run, not included).
4. Longest run, (1PS68AA 1/2, short run, not included).

FIGURE 1



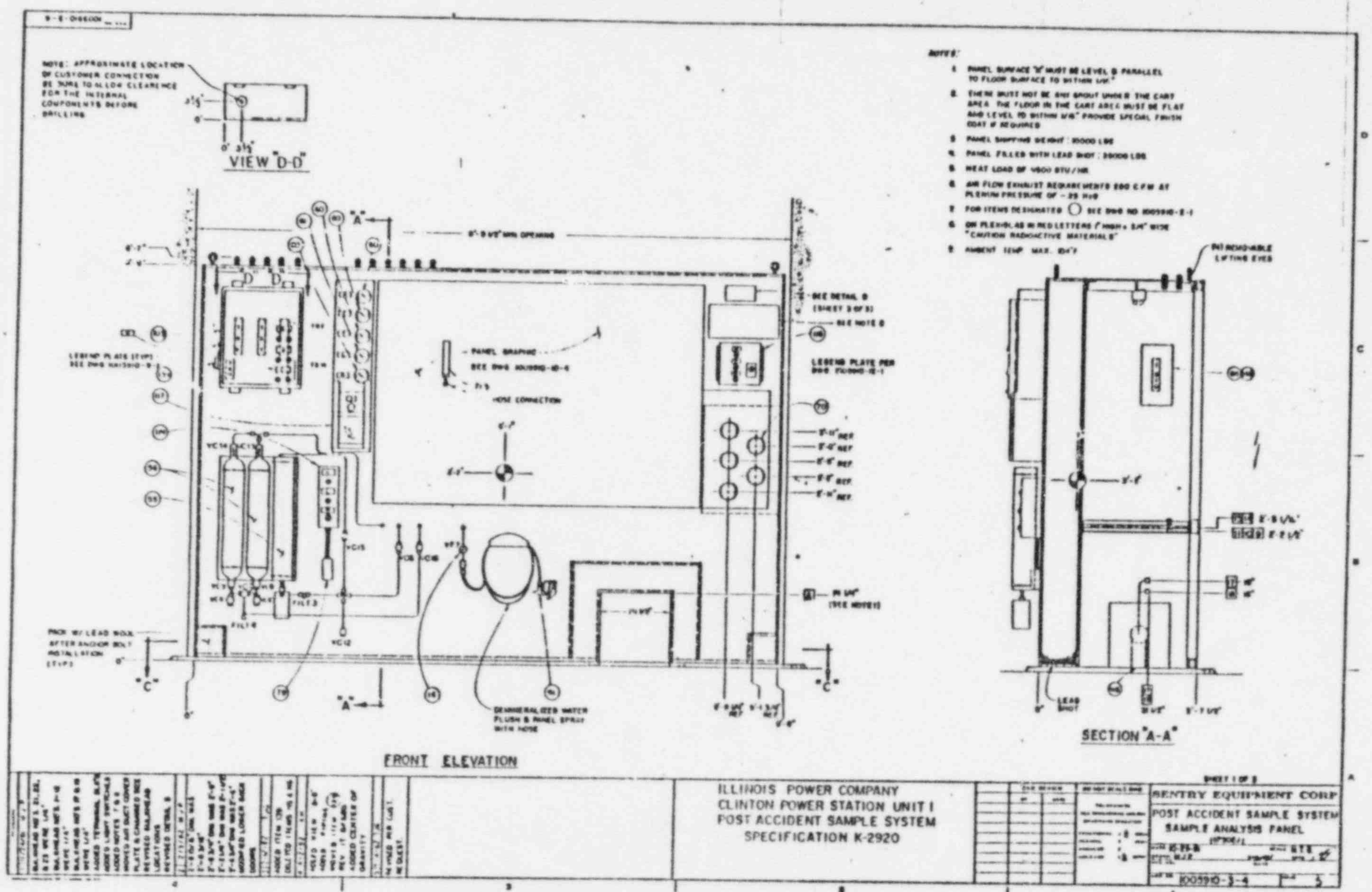
PASS COMPONENTS YSICAL LAYOUT

FIGURE 2



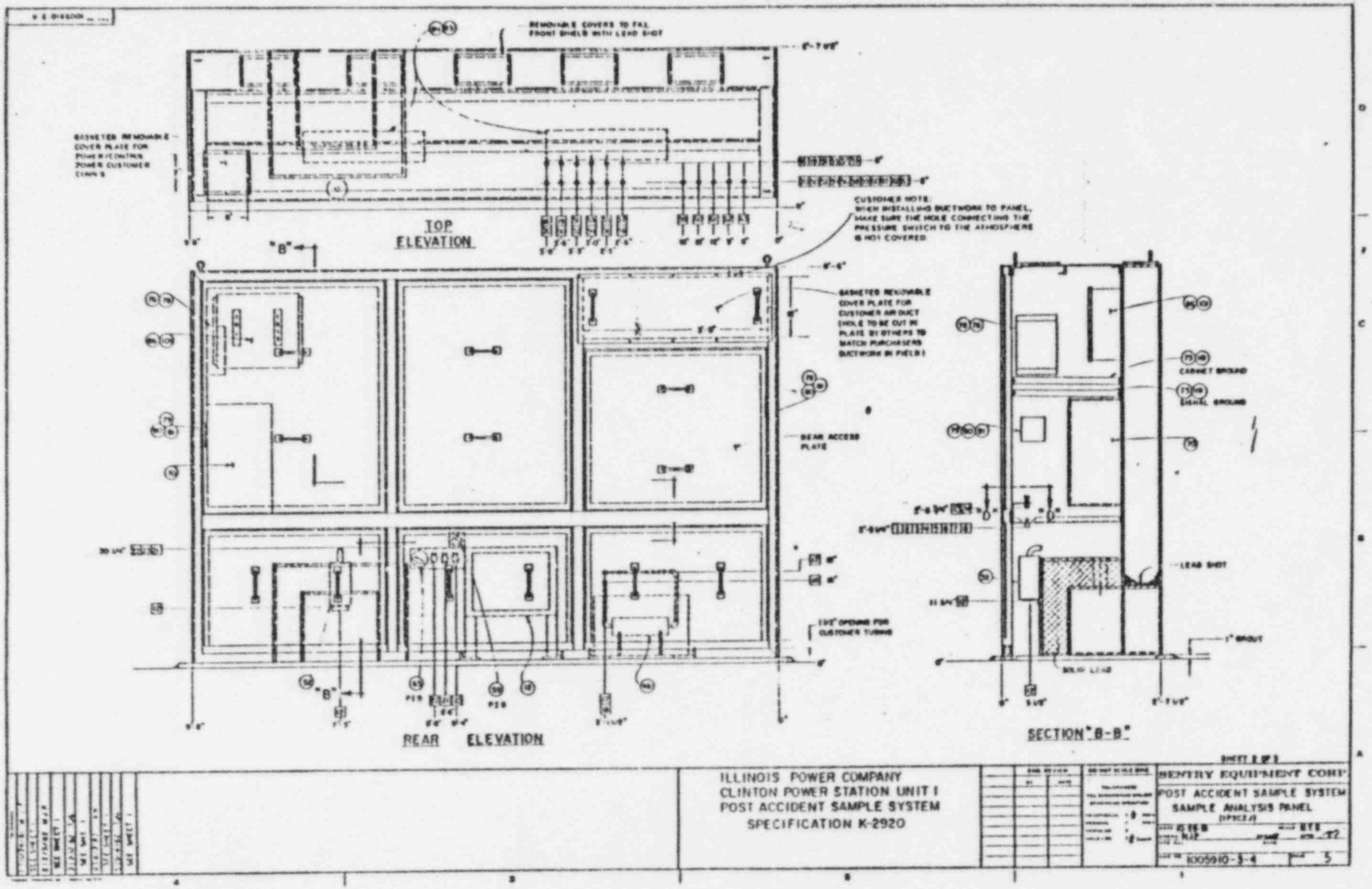
PASS PANEL LOCATION

FIGURE 3



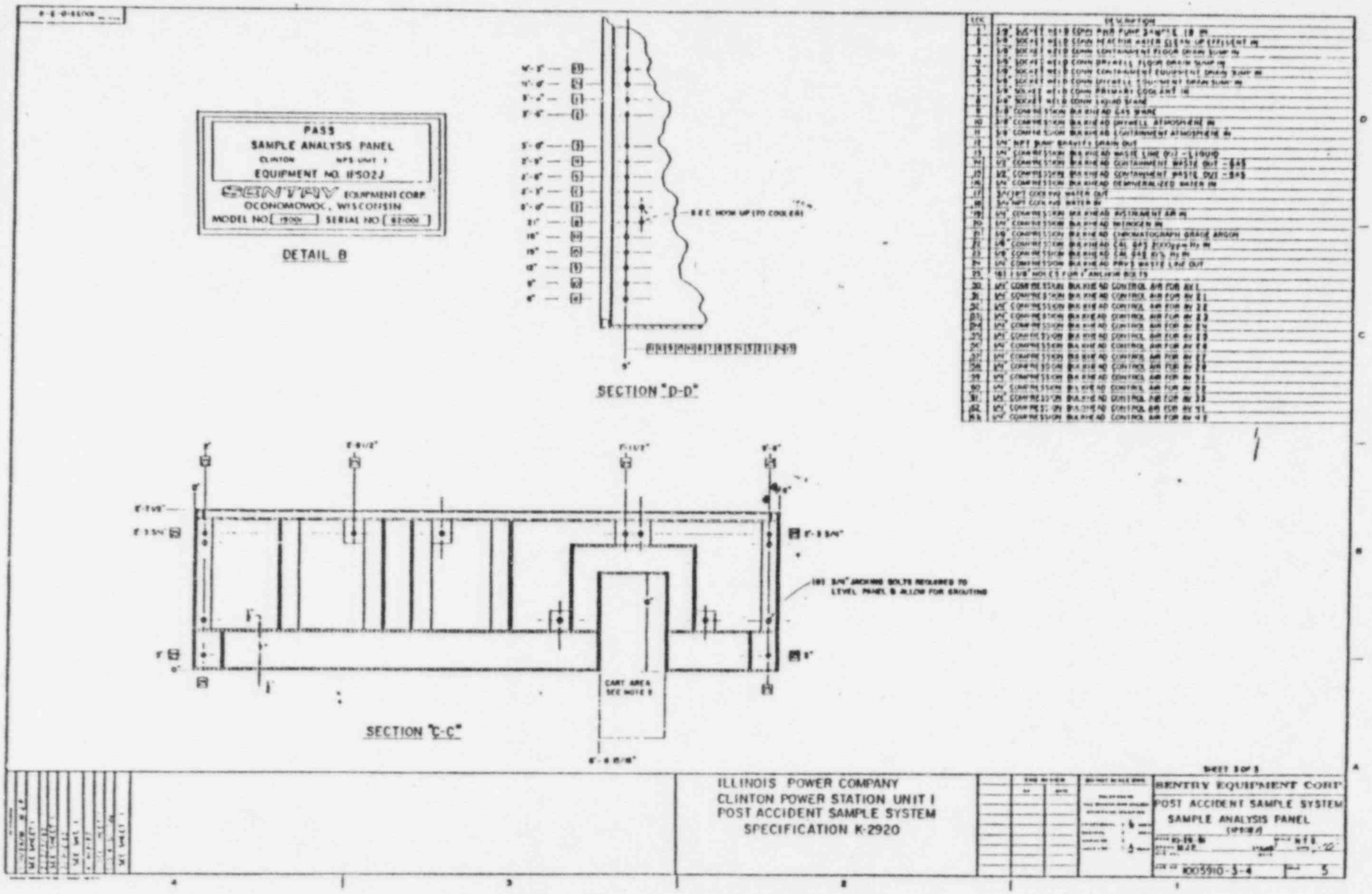
SAMPLE ANALYSIS PANEL (FRONT VIEW)

FIGURE 4



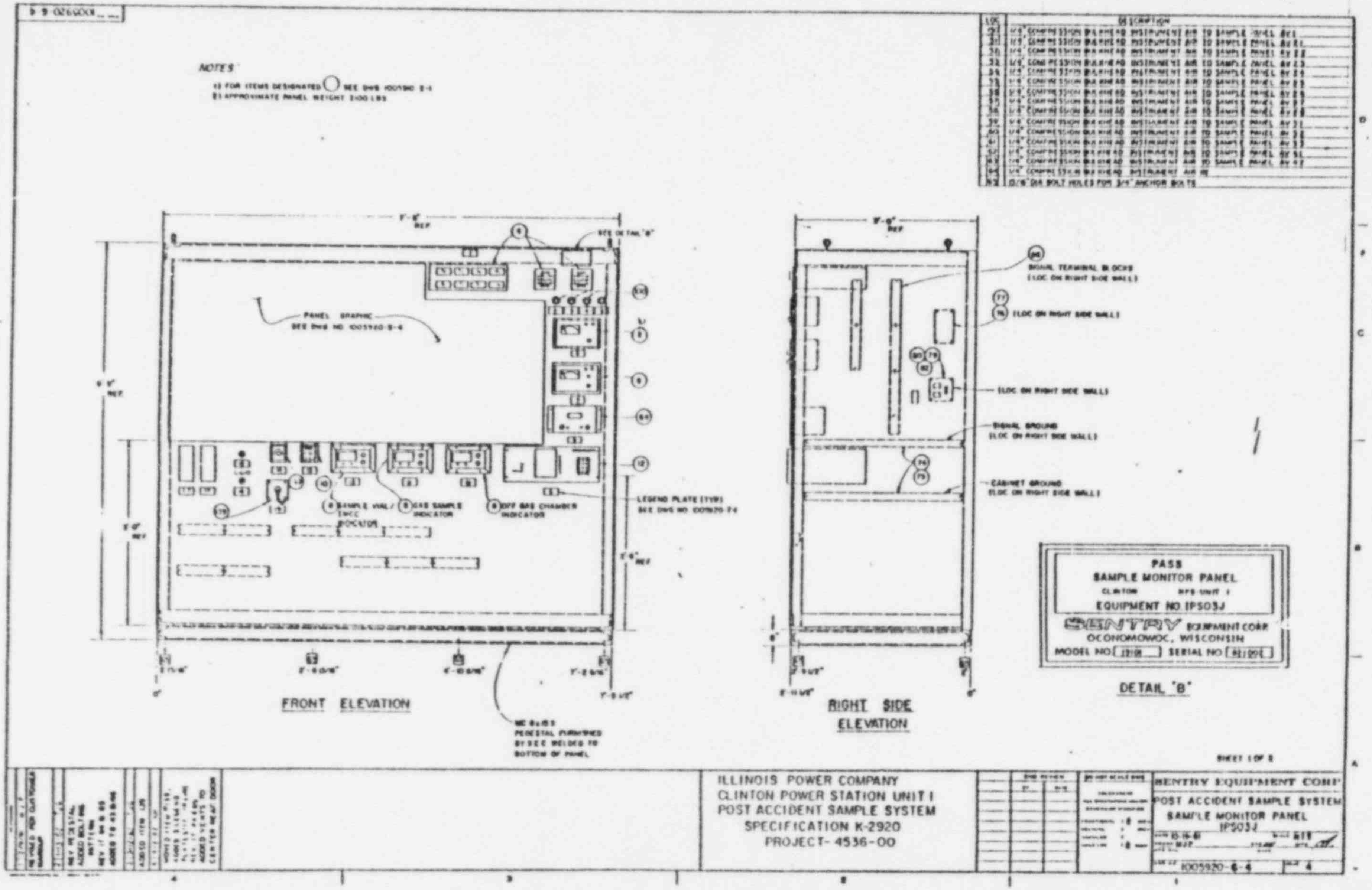
SAMPLE ANALYSIS PANEL (TOP AND REAR VIEW)

FIGURE 5



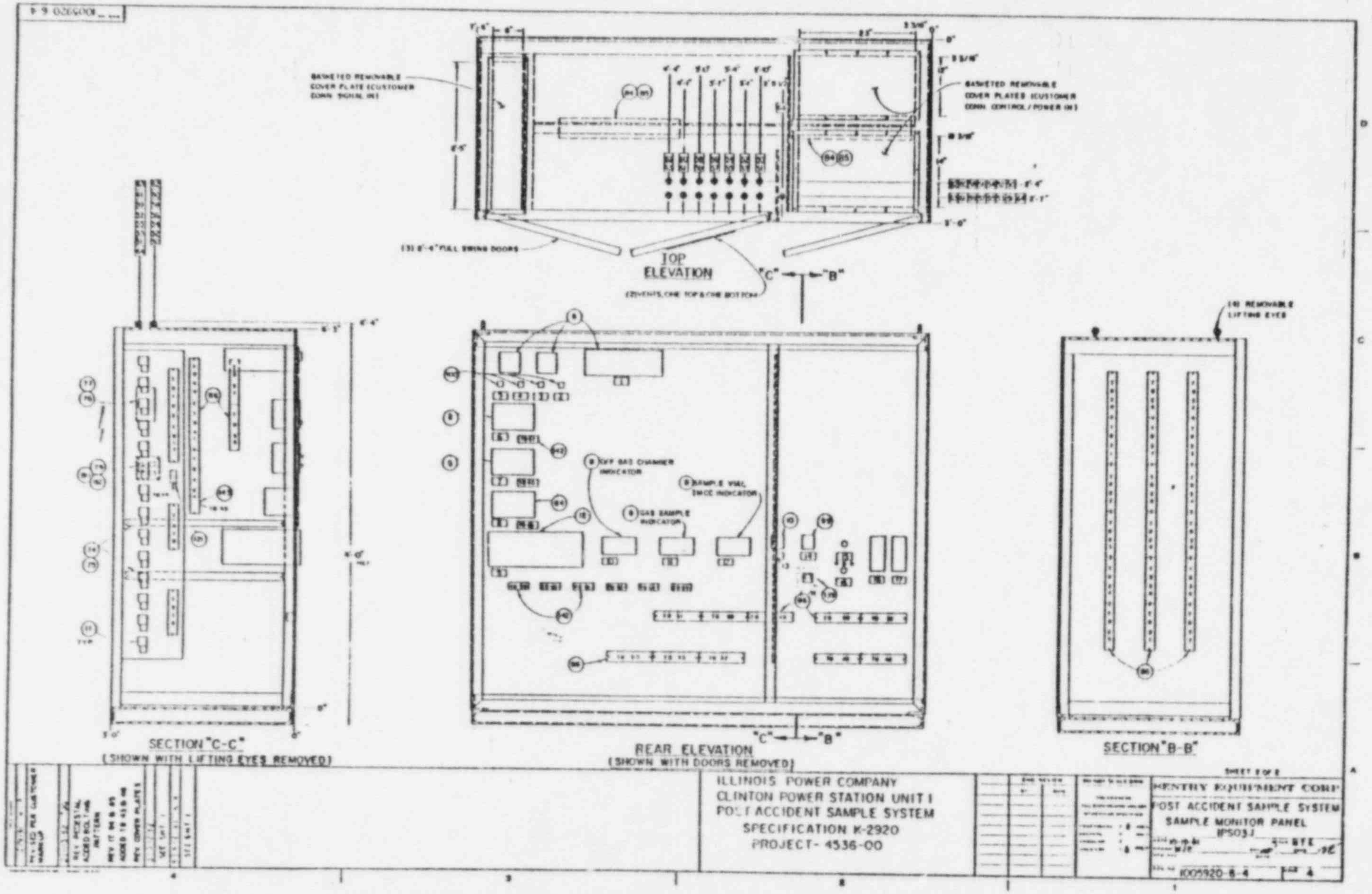
SAMPLE ANALYSIS PANEL (DETAILS)

FIGURE 6



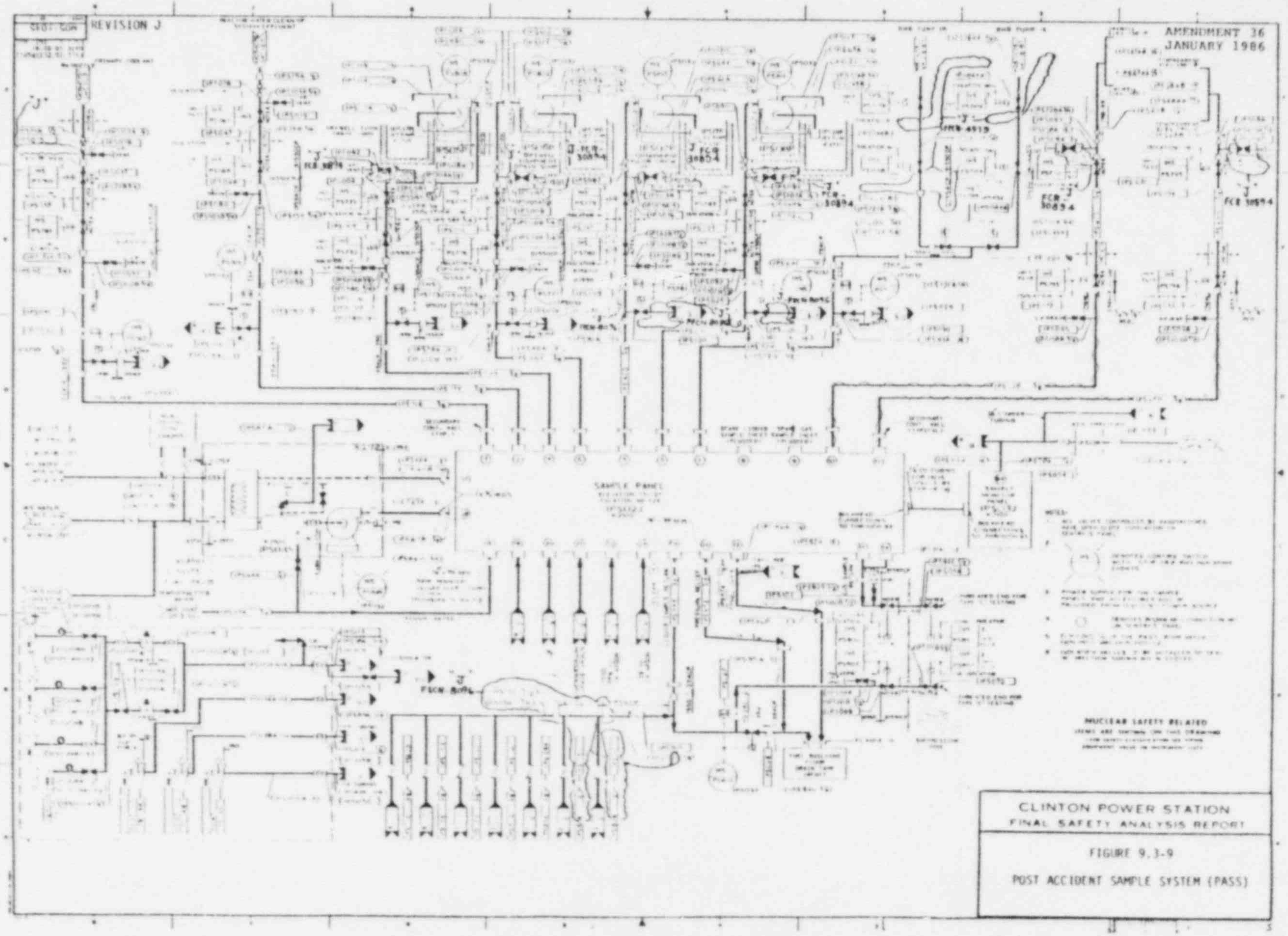
SAMPLE MONITOR PANEL (FRONT AND RIGHT VIEW)

FIGURE 7



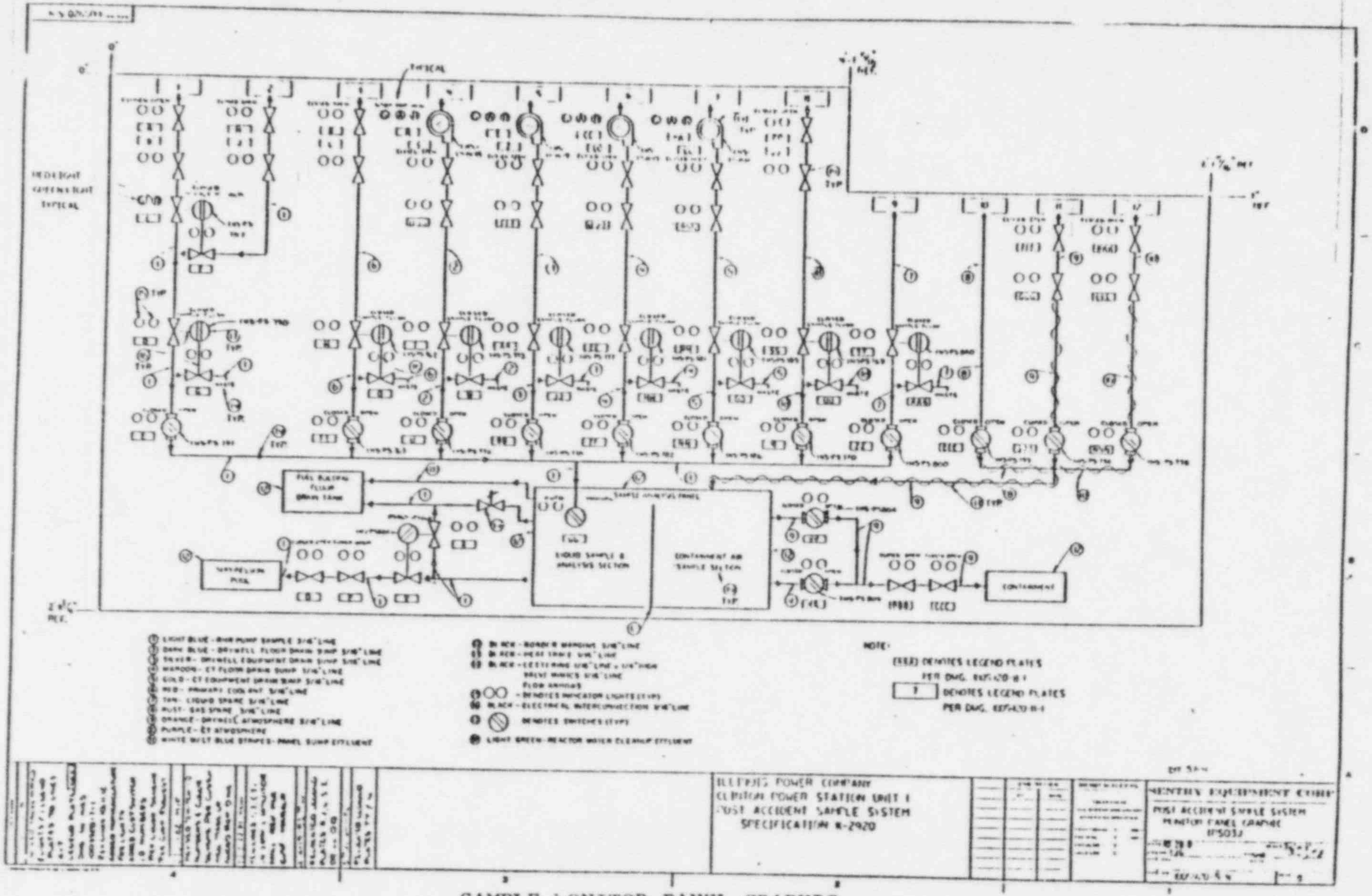
SAMPLE MONITOR PANEL (TOP AND REAR VIEW)

FIGURE 8



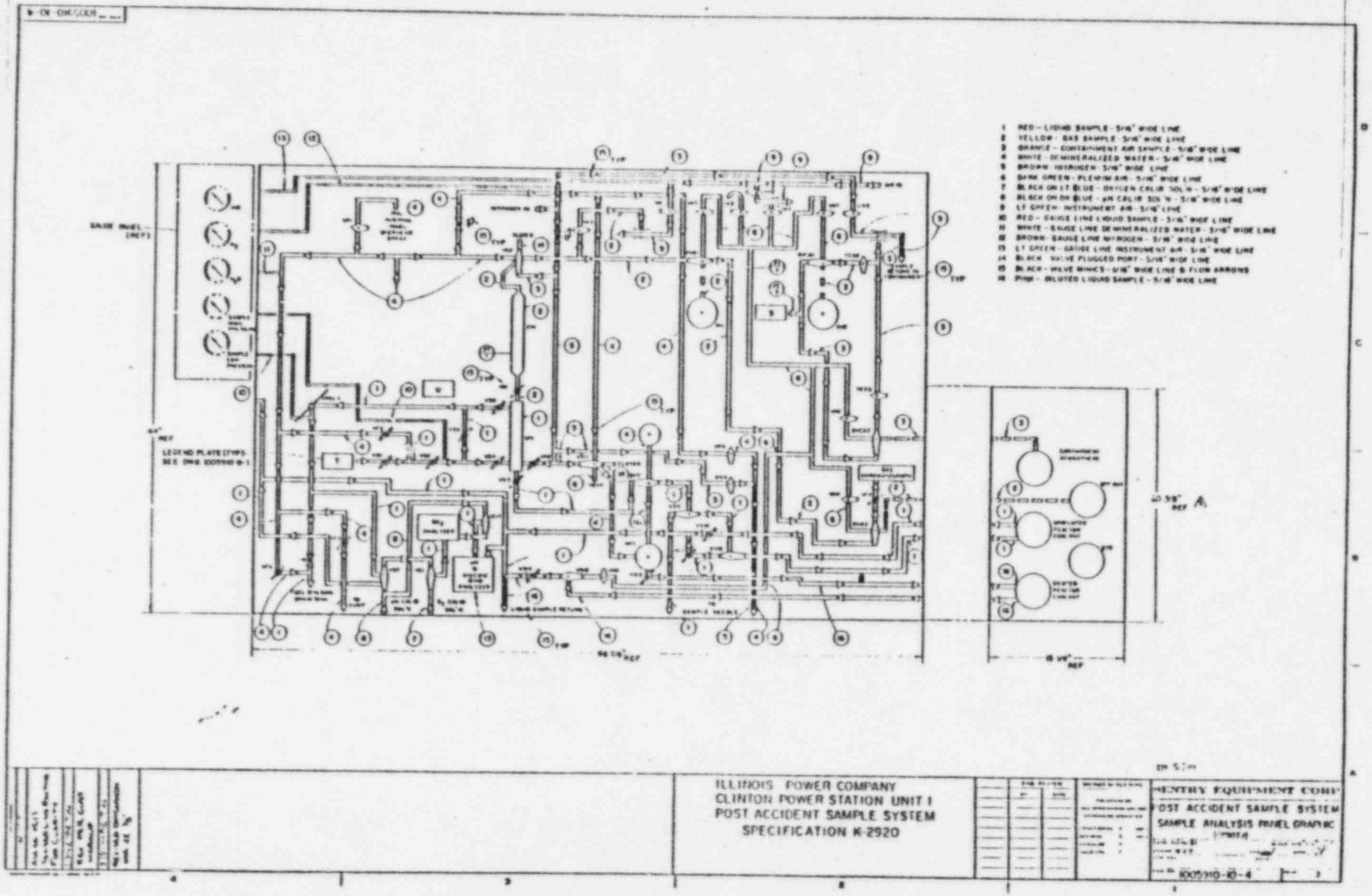
POST ACCIDENT SAMPLE SYSTEM (PASS)

FIGURE 9



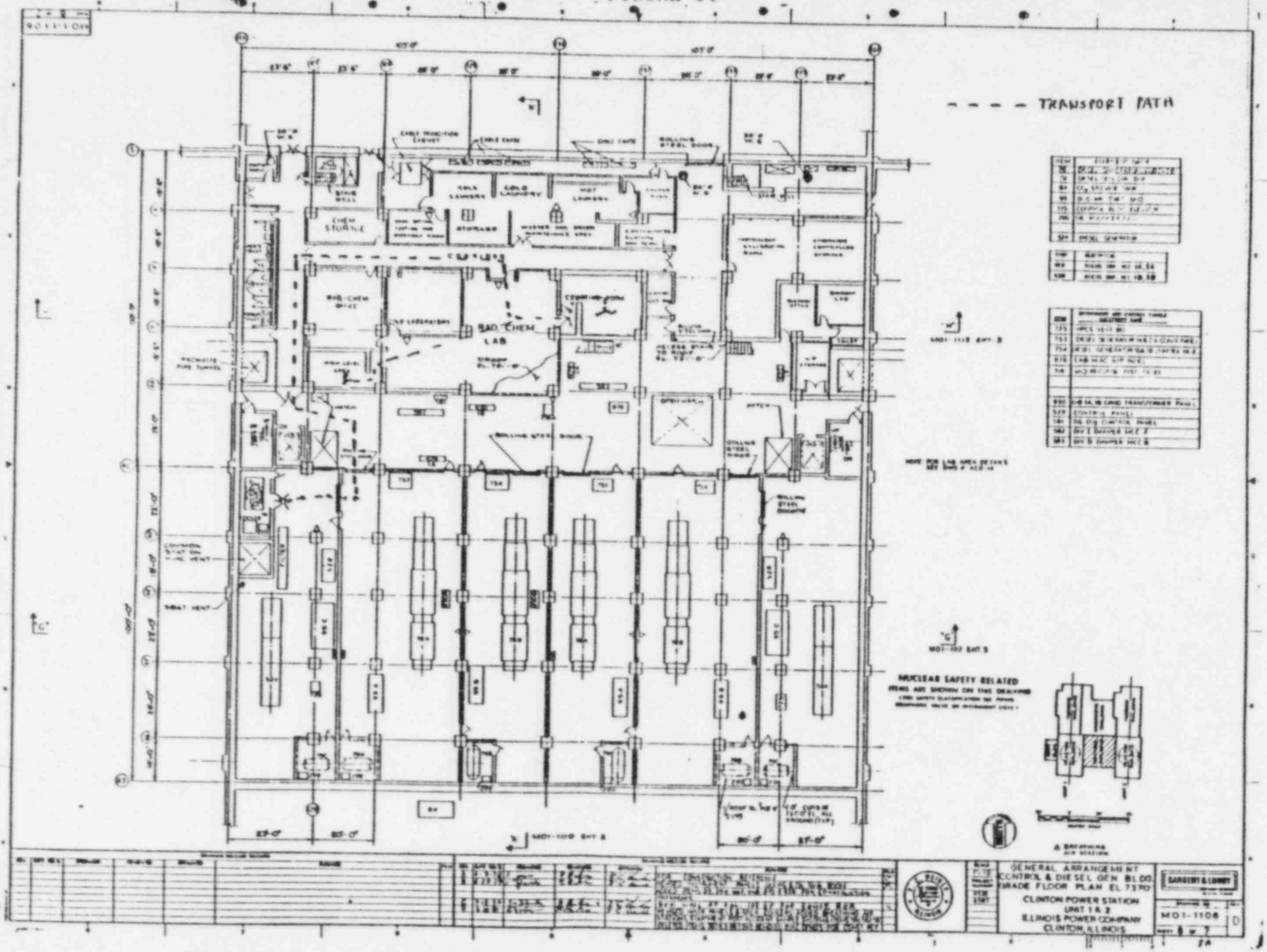
SAMPLE MONITOR PANEL GRAPHIC

FIGURE 10



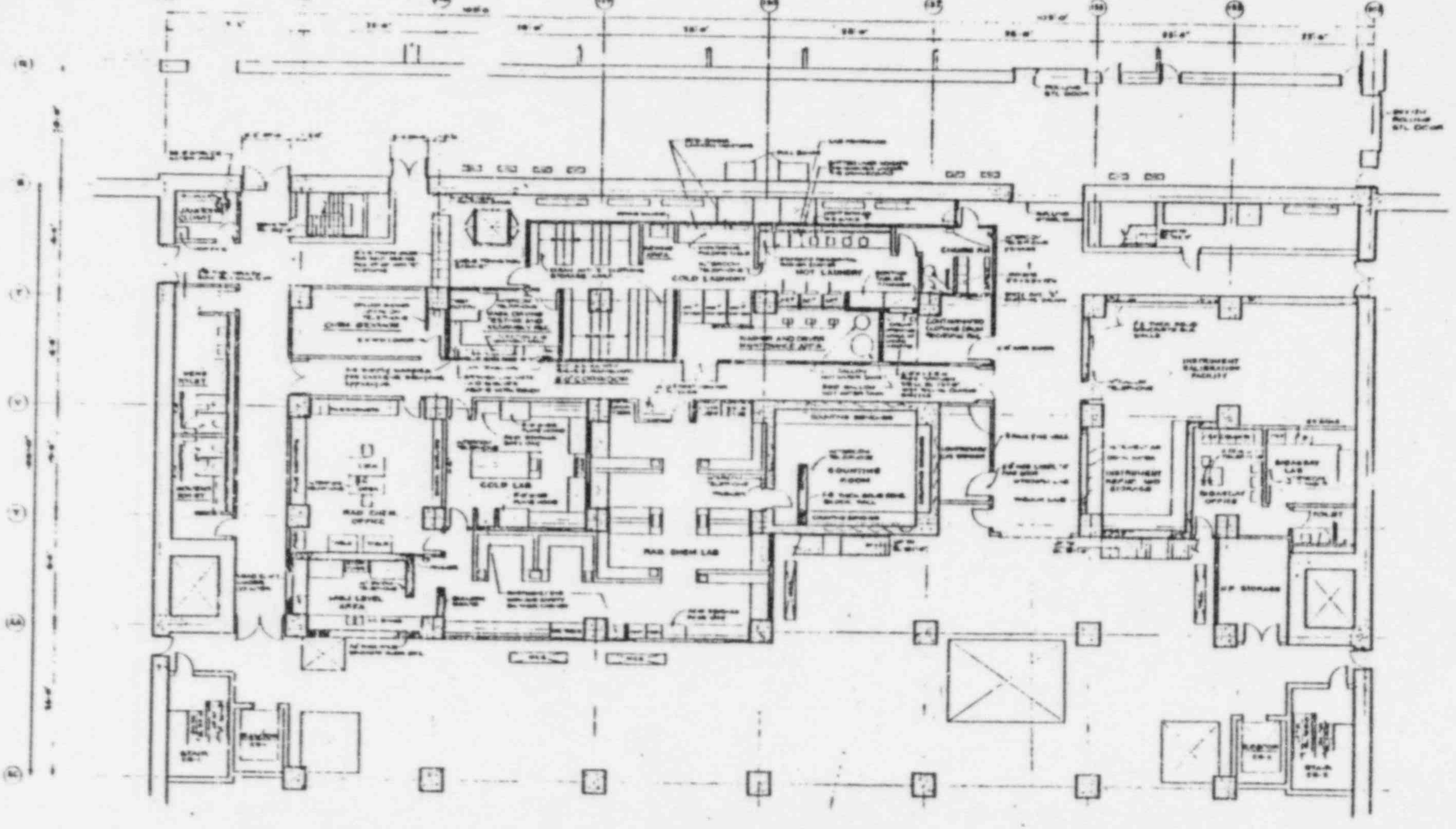
SAMPLE ANALYSIS PANEL GRAPHIC

FIGURE 11



TRANSPORT PATH TO RAD CHEM LABORATORIES

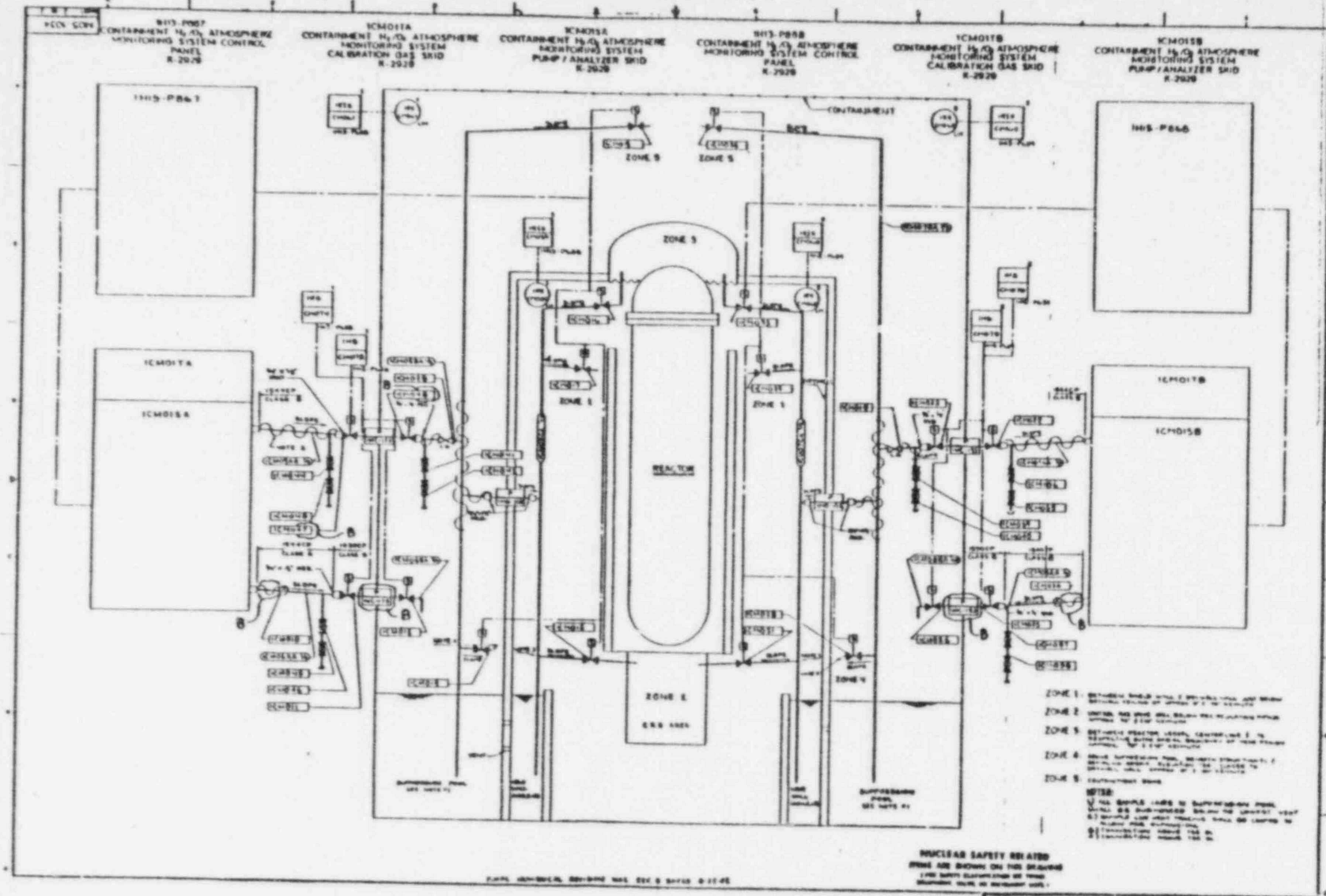
FIGURE 12



CONTROL BUILDING - EL. 737'-0"

LAYOUT OF STATION LABORATORIES COUNTING ROOM AND LAUNDRY

FIGURE 13



AMENDMENT 26
JULY 1983

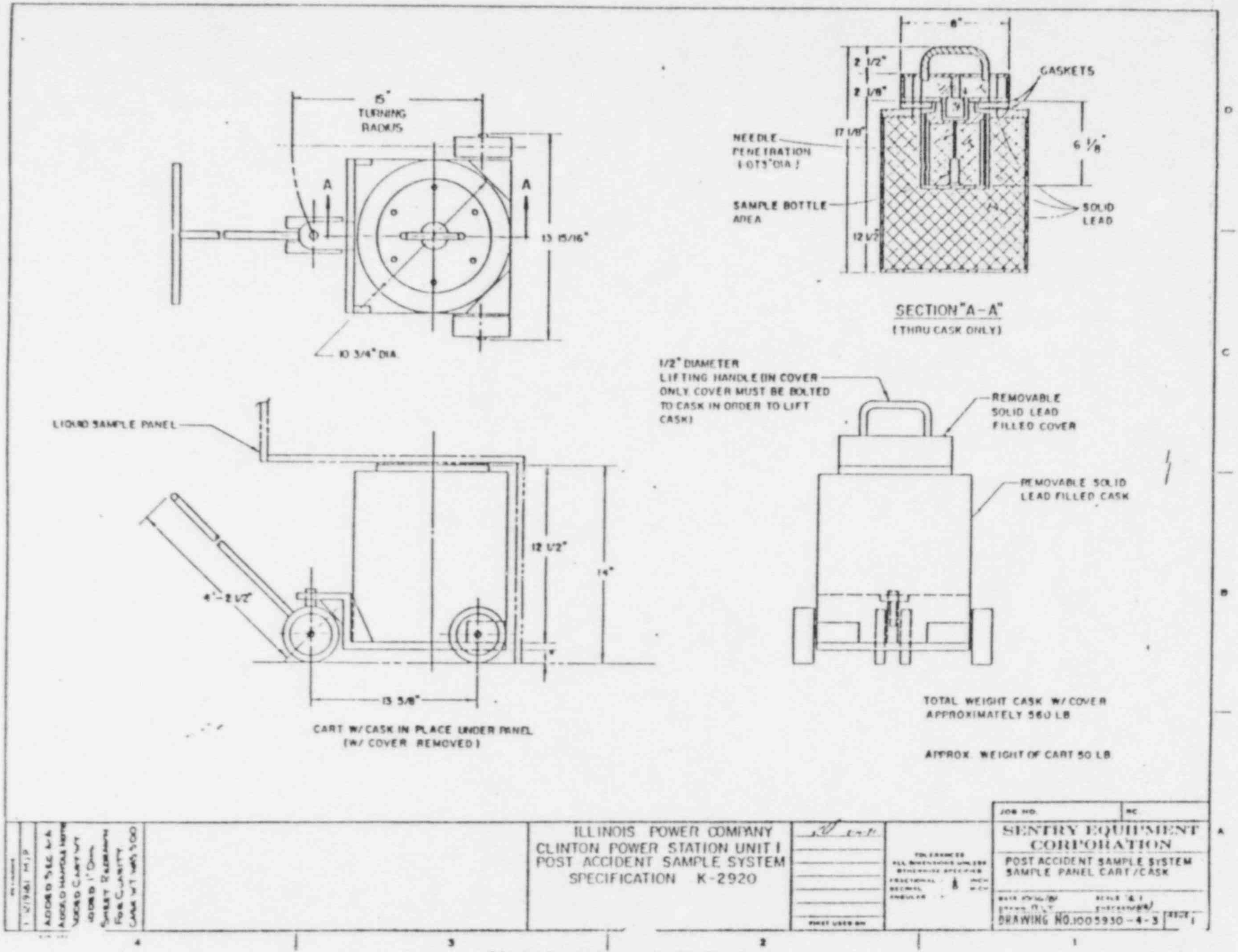
CONTAINMENT ATMOSPHERE MONITORING SYSTEM P&ID

CLINTON POWER STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE 7.6-23

CONTAINMENT ATMOSPHERE
MONITORING SYSTEM P&ID

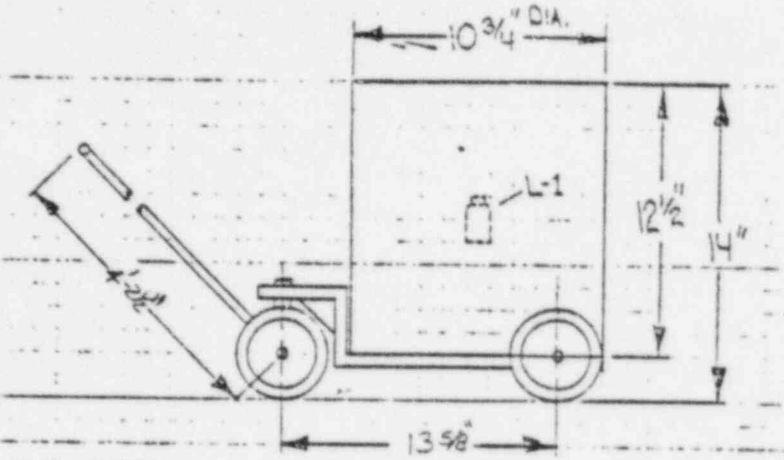
FIGURE 14



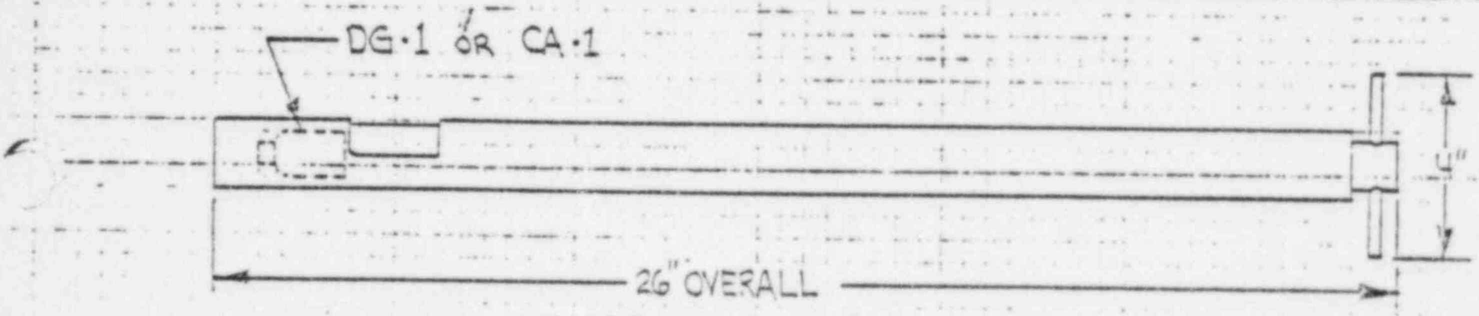
SAMPLE PANEL CART/CASK

FIGURE 15

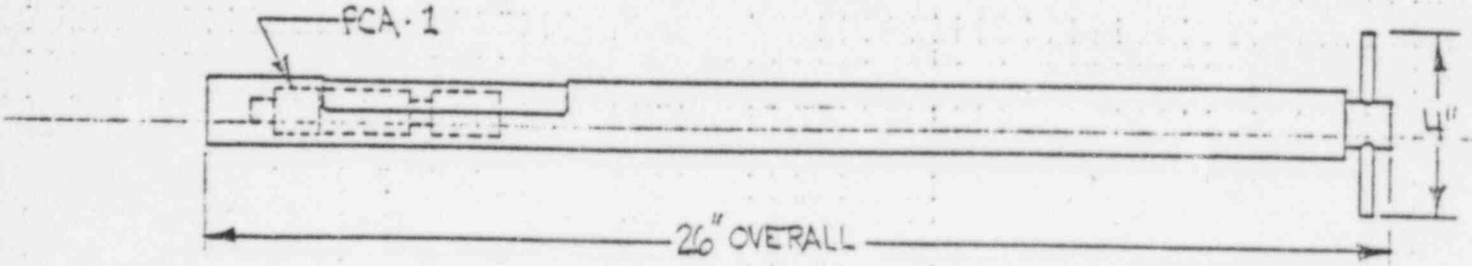
SKETCH: MOBILE PIG FOR UNDILUTED LIQUID SAMPLE



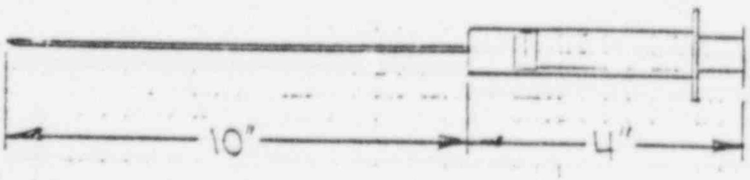
SKETCH: GAS SAMPLE TONGS & LIQUID ALIQUOTER



TONGS T-1, T-2



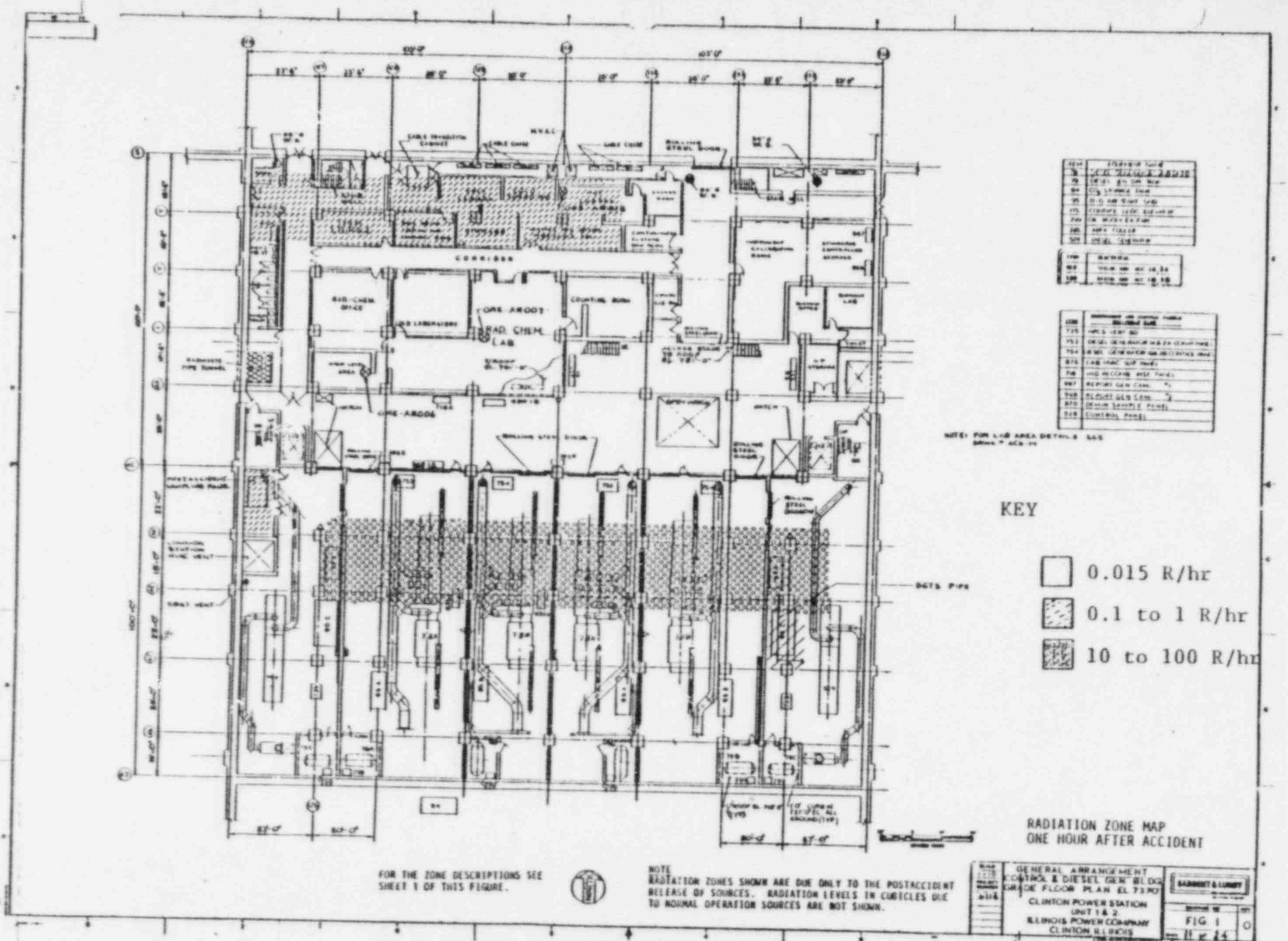
TONG T-3



ALIQUOTER DL-1

MOBILE PIG, GAS SAMPLE TONGS AND LIQUID ALIQUOTER

FIGURE 16



POST ACCIDENT RADIATION MAP

ATTACHMENT 1

Post Accident Sampling System

Person-Motion Study for Radiation Exposure at Clinton Power Station During Post-Accident Exercises

Objectives: Determine that the radiation dose to any one individual involved with Post-Accident exercises (i.e. sampling, transport, and analysis) does not exceed the criteria of GDC 19 (Appendix A, 10CFR, Part 50; i.e., 5 Rem whole body, 75 Rem extremities).

Assumptions/Data:

1) The source terms assumed are per Regulatory Guide 1.3 (100% Noble Gas, 50% Halogens and 1% particulates based on design basis accidents).

2) The sample radiochemistry design parameters (at 1 hour after the accident) are listed on Table 1. The resulting exposure rate (R/h) is listed on Table 2 for each of the samples taken at the Sample Analysis Panel (SAP).

3) The time estimated for transporting samples to the laboratory (from the SAP) is 20 minutes (This is conservative based on walkdown results).

4) The ambient radiation levels for the PASS panel area, Rad Chem laboratory and transport path to it are taken from post-accident radiation maps (Figure 1) at 1 hour (earliest time maps developed) after an accident. These maps were prepared as part of the Post-Accident Shielding evaluation performed for Clinton Power Station (CPS) in response to NUREG-0737, item II.B.2.

   The radiation level in the shielded PASS room is caused by the sources present in the PASS panel itself; thus the only exposure incurred is from the sampling and analysis exercises.

   The dose to an operator during the above exercises has been found to be 1305 mrem, whole body, and 171 mrem for extremities (Ref. 1).

   For exposure received from the PASS Panel room after completion of the sampling exercises, the average radiation level in the area is assumed to be equal to the highest radiation level in the area from post-accident radiation maps.

5) Panel activity is concentrated as a point source located directly behind the SAP shield. This is conservative since it results in the highest calculated doses.

6) The Operator does not move away during flushing. This is conservative since the calculated doses are maximized.

7) The highest dose rate during an exercise step is used to determine the exposure for that step.

8) Post Accident exercises are completed in 3 hours (maximum), as specified in NUREG-0737, item II.B.3.

9) The activity levels for all sources are constant throughout the Post-accident exercises. This is consistent with the assumptions employed in the Post-Accident Shielding Survey performed for TMI item II.B.2.

10) Fig. 2 provides a method for determining Gamma Attenuation Factors (AF) for lead and iron. The average gamma energy releases used with this figure are 1.2 MeV for Liquids and 1.3 MeV for Gases. These are the weighted energy release averages.

Person-Motion Study

1) Sampling and In-line Analyses Exercises

The total radiation dose to an operator during this period is 1305 mrem for the hands and 171 mrem whole body.

2) Transporting Samples

A. Undiluted Reactor Coolant (L-1)

The sample is contained in a mobile shielded cask. The cask has a 4.7" shield of solid lead and is pulled by its 4' (122cm) long handle (See Fig. 4):

Hands and Body distance = 122 cm (4') + 12 cm (4.7") = 134 cm,

Sample Activity (Table 2): whole vial = 55,136 R/hr @ 1 cm

Attenuation Factor (Figure 2) = 0.0008

Transport Time = 3 hours (worst case, to shipping area)

Hands and Body

$D = (55{,}136\ R/hr)(0.0008)(3hr)(1{,}000\ mR/R)/(134\ cm)^2$
$= 7.4\ mR$

B. Diluted (1,000:1) RC (DL-1)

The sample is transported to the laboratory in a shielded (.34" of lead + .22" of iron) syringe. A worker grasps the syringe by its handle and holds it 18" in front of his body with the bore of the syringe positioned horizontally.

Distance from hands = 27 cm (10.5")
Distance from body = 46 cm (18")
Sample activity per cc = 13.784 R/hr @ 1 cm
Sample aliquote size = 10 cc (See Fig. 5)
Attenuation Factor (iron) = 0.82
Attenuation Factor (lead) = 0.6
Combined AF = (0.82)(0.6) = 0.49
Transport time = 20 minutes

Hands

$D = (13.784\ R/hr/cc)(10cc)(0.49)(20/60\ hr)(1{,}000\ mR/R)/(27\ cm)^2$
$= 31.0\ mR$

Body

D = (13.784 R/hr/cc)(10cc)(0.49)(20/60 hr)(1,000 mR/R)/(46 cm)^2
= 10.6 mR

C. GAS (DA-1)

The sample is contained in a vial inside a 26" tong. A worker hand-carries the tong to the laboratory with the tong pointed away from his body:

Distance from hands = 46 cm (18")
Distance from body = 66 cm (26")
Sample Activity, whole vial = 14.2 R/hr @ 1 cm
AF = 1
Transport time = 20 minutes

Hands

D = (14.2 R/hr)(1)(20/60hr)(1,000 mR/R)/(46 cm)^2
= 2.2 mR

Body

D = (14.2 R/hr)(1)(20/60hr)(1,000 mR/R)/(66 cm)^2
= 1.1 mR

D. RC Off-Gas (DG-1)

The sample is contained in a vial inside a tong (26" long). A worker carries the tong to the laboratory and points it away from his body:

Distance from hands = 18 in (46 cm)
Distance from body = 26 in (66 cm)
Sample activity, whole vial = 85.4 R/hr @ 1 cm
AF=1
Transport Time = 20 minutes

Hands

D = (85.4 R/hr)(1)(20/60hr)(1,000 mR/R)/(46 cm)^2
= 13.5 mR

Body

D = (85.4 R/hr)(1)(20/60hr)(1,000 mR/R)/(66 cm)^2
= 6.5 mR

3. Chemical Analysis

A. The analyses for hydrogen, dissolved oxygen, pH and conductivity are performed in-line as part of the PASS sampling and analysis exercises. Thus the total dose for these exercises, (see item 1), includes the dose for the in-line analyses.

B. Boron in Diluted (1,000:1) RC

This analysis is performed in the site laboratory using a Tetrafluoroborate Selective Ion Electrode (TSIE). The time for this analysis is about 15 minutes while the sample size to be analyzed is 5cc (Ref. 5). During the analysis, the sample can be maintained at a 15 cm distance from the hands and 40 cm from the body:

Distance from hands = 15 cm
Distance from body = 40 cm
Sample activity = 13.784 R/hr @ 1 cm
Sample size = 5cc
Time = 15 minutes

Hands

$$D = (13.784 \text{ R/hr cc})(5\text{cc})(15/60 \text{ hr})(1{,}000 \text{ mR/R}) / (15 \text{ cm})^2 = 77 \text{ mR}$$

Body

$$D = (13.784 \text{ R/hr cc})(5\text{cc})(15/60 \text{ hr})(1{,}000 \text{ mR/R}) / (40 \text{ cm})^2 = 11 \text{ mR}$$

C. Chloride in RC

An undiluted reactor coolant sample will be analyzed at an offsite facility within four (4) days with no exposure resulting to Clinton Power Station Personnel. In the event of a minor accident (sample activity is at/or below 1/1,000 of the worst case activity), a liquid sample may be analyzed onsite via an Ion Chromatograph.

C.1 Offsite Analysis

Dose to: Hands = 0
Body = 0

C.2 Onsite Analysis

This analysis will be performed in the site laboratory using a model 2010i Ion chromatograph (IC). Following reactor coolant injection into the IC, the chemist moves away while the analysis is carried out. Thus the only significant dose received is due to the injection process. The sample can be injected into the IC in 120 seconds. The minimum body distance for this process is 15 cm, while the hands will be at 5 cm from the sample:

Distance from hands = 5 cm
Distance from body = 15 cm
Sample activity per cc = 13.784 R/hr @ 1 cm
Sample size = 6cc
AF = 0.49 (See Item 2B)
Time = 2 minutes

Hands

$$D = (13.784\ \text{R/hr cc})(6\text{cc})(0.49)(2/60/\text{hr})(1{,}000\ \text{mR/R})/(5\ \text{cm})^2$$
$$= 54\ \text{mR}$$

Body

$$D = (13.784\ \text{R/hr cc})(6\text{cc})(0.49)(2/60\text{hr})(1{,}000\ \text{mR/R})/(15\ \text{cm})^2$$
$$= 6\ \text{mR}$$

D. Nuclide Analysis (Liquid and Gas Samples)

For this analysis in the on-site laboratory, a diluted liquid sample and a gas sample will be used.

Liquid sample size = 14cc; Activity per cc = 13.78 R/h @ 1 cm

Gas sample size = 0.023cc; Activity per cc = 618 R/h @ 1 cm

Total Sample Activities

Liquid = 193 R/h @ 1cm
Gases = 14.2 R/h @ 1cm
Time = 2 minutes for injecting samples (with a syringe) into a counting bottle.
Body distance = 15cm
Hand distance = 10cm (avg)

Hands

$$D = \frac{(193\ \text{R/h})(2/60\text{h})(1{,}000\ \text{mR/R}) +}{(10\text{cm})^2}$$

$$\frac{+\ (14.2\ \text{R/h})(2/60\text{h})(1{,}000\ \text{mR/h})}{(10\text{cm})^2} = 69.1\text{mR}$$

Body

$$D = \frac{(193\ \mathrm{R/h})(2/60\mathrm{h})(1{,}000\ \mathrm{mR/R})\ +}{(15\mathrm{cm})^2}$$

$$+\ \frac{(14.2\ \mathrm{R/h})(2/60\mathrm{h})(1{,}000\ \mathrm{mR/h})}{(15\mathrm{cm})^2} = 30.7\ \mathrm{mR}$$

E. Ambient Contributions

In addition to the exposure described in A,B,C and D above, further exposure is received from ambient radiation levels in (1) the PASS panel room, (2) Radiological Chemistry (Rad Chem) laboratory and (3) the transport path to it (Fig. 3). The radiation levels in these areas are taken from the post-accident shielding evaluation as described in Assumption 4. Fig. 1 shows the post-accident radiation levels in the areas of concern.

1. PASS Room

The doses for the 1st part of the exercise, due to ambient radioactivity are included in the total dose received during the PASS sampling and analysis exercises. The above is true since the PASS panel along with sample lines, cooler etc., are the only sources for radioactivity in the room.

The occupancy level in the room drops off considerably, following sampling and analysis exercises. 20% occupancy in the PASS room is assumed for the time following sampling procedures. From Fig. 1, the rad level in the room will be 0.1 to 1 R/hr (assuming radioactivity is still present in the SAP).

Ambient Rad level (worst case) = 1 R/hr

Occupancy = 20% for 1½ hours

Hands & Body

$$D = (1\mathrm{R/hr})(0.2)(1\tfrac{1}{2}\ \mathrm{hr}) = 0.3\ \mathrm{R} = 300\ \mathrm{mR}$$

2. Transport path to the Rad Chem laboratory

Fig. 3 shows the transport path to the Rad Chem laboratory. The radiation level in this area is a maximum of 15 mR/hr. The transport time is 20 minutes per sample. A total of four samples may be taken to the laboratory (i.e. Undiluted Liquid, Diluted Liquid, RC Off-Gas and Containment Air).

Ambient level = 15 mR/hr

Total Transport time = (20)(4) = 80 minutes)

Hands & Body

D = (15 mR/hr)(80/60 hr) = 20 mR

3. Rad Chem Laboratory

The radiation level in the lab is a maximum of 15mR/hr. The maximum amount of time an individual will spend in the laboratory is 3 hrs.

Ambient level = 15 mR/hr

Time duration = 3 hrs

Hands & Body

D = (15 mR/hr)(3 hrs) = 45 mR

Summary

Table 3 summarizes the calculated doses. Assuming the worst possible case (one person doing everything), man-rem exposure for the Clinton post-accident sampling, transporting and analyses is substantially less than the integrated dose limits set by the NRC. The total whole body dose is 12% of the limit dose while the extremities dose is about 3% of the limit.

TABLE 1

Sample Radiochemistry Design Parameters
(@ 1 hr after an accident)

1. RC Liquid Not Degassed: Gamma

   D = 17,495 R/hr/cc @ 1cm*

2. RC Liquid Degassed: Gamma

   D = 13,784 R/hr @ 1cm per cc of RC

3. RC Off-Gas: Gamma

   D = 3,711 R/hr/cc @ 1cm

4. Drywell Atmosphere: Gamma

   D = 618 R/hr/cc @ 1cm

* Radiation Level for 1cc of sample @ 1cm.

* The above information is taken from Ref. 4.

TABLE 2

Predicted Activities of Diluted and Undiluted Grab Samples
(@ 1 hour after an accident)

1. Undiluted liquid, degassed (L-1) Sample size = 4cc

   D = 55,136 R/hr @ 1cm (whole bottle)

2. Diluted Liquid (1,000:1) (DL-1) Sample size = 10cc

   D = 137.8 R/hr @ 1cm (whole vial)

3. RC Off-Gas (DG-1) Sample size = 0.023cc

   * D = 85.4 R/hr @ 1cm (whole bottle)

4. Drywell Atmosphere (DA-1) Sample size = 0.023cc

   D = 14.2 R/hr @ 1cm (whole bottle)

---

* Gases are stripped from a 30 cc liquid sample and subsequently they are allowed to expand into a 300 cc evacuated volume. The total activity of the gas remains constant; however, the activity per cc of gas is reduced by a factor of 10. For conservatism, this dilution constant is not taken into account in the analysis.

TABLE 3

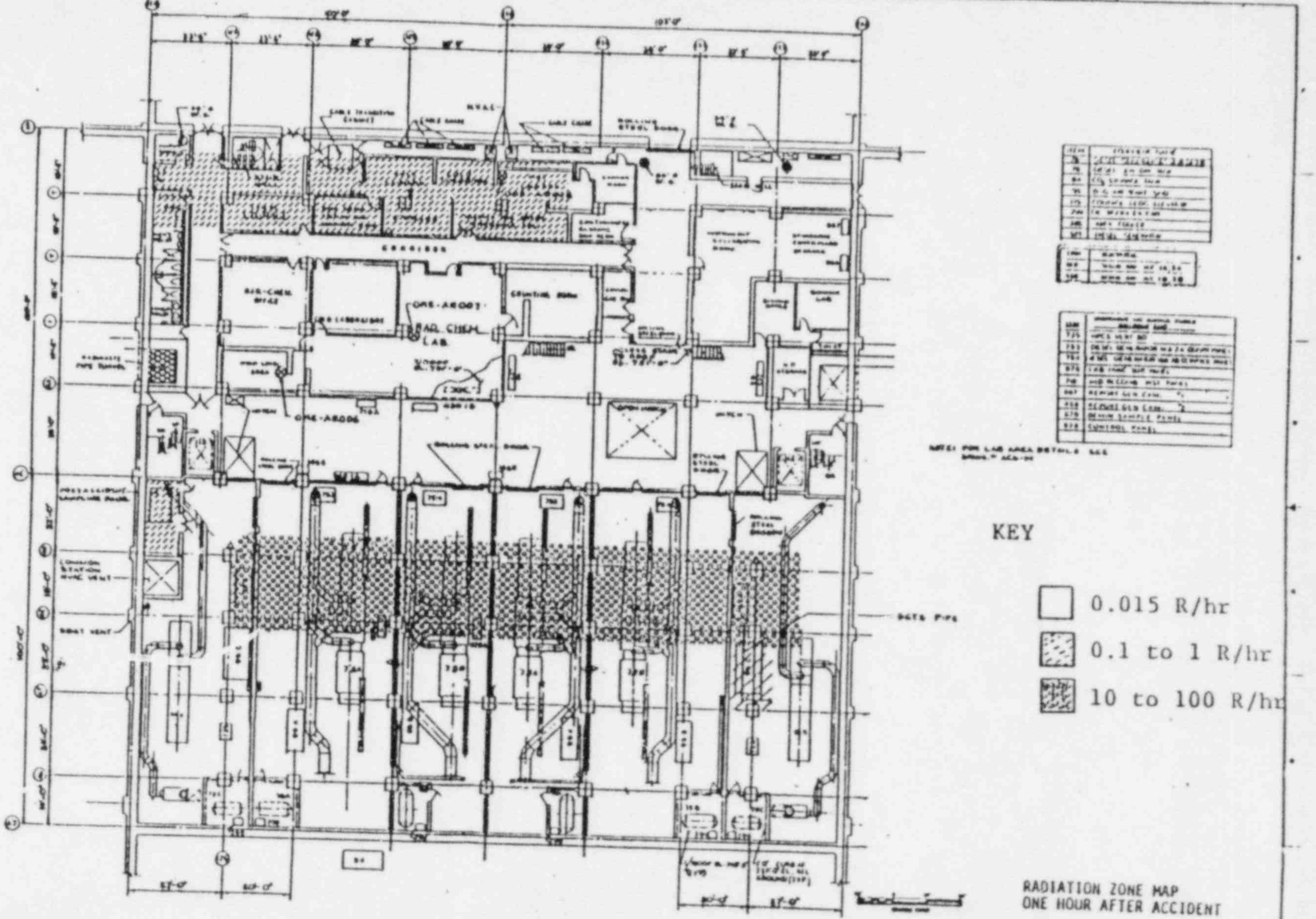
Personnel Radiation Exposure Predictions

ACTION		EXPOSURE (mR)	
		hand	body
PASS Exercises		1305	171
	Ambient contribution (after sampling)	300	300
Transporting Samples	RC off-gas	13.5	6.5
	Undiluted RC	7.4	7.4
	Diluted RC	31.0	10.6
	DA	2.2	1.1
	Ambient contribution	20	20
Analysis of Samples	Chloride (Worst Case)	54	6
	Boron	77	11
	Nuclides	69.1	30.7
	Ambient Contribution	45	45
TOTAL (Worst Case)		1924 mRem	609 mRem
Criterion Maximum		75 Rem	5 Rem

References

1. Nuclear Station Engineering Department (NSED) calculation number 8, Rev. 1 "Operator Dose Report for Sampling Exercises on Model B Sample Analysis Panel."
2. Sentry Equipment Corporation (SEC) Specification B10-01.
3. Appendix C, Figure 2 & 3, page 78 & 79 of "Safe Handling of Radioactive Materials", Handbook 92, U.S. Department of Commerce.
4. Nuclear Station Engineering Department (NSED) calculation number 7, "Post-Accident Design Source Terms Analysis." (Rev. 0)
5. CPS Operating Procedure No. 1890.33 "Post Accident Sample Analysis." (Rev.0)

FiG. 1



RADIATION ZONE MAP
ONE HOUR AFTER ACCIDENT

FOR THE ZONE DESCRIPTIONS SEE SHEET 1 OF THIS FIGURE.

NOTE
RADIATION ZONES SHOWN ARE DUE ONLY TO THE POSTACCIDENT RELEASE OF SOURCES. RADIATION LEVELS IN CUBICLES DUE TO NORMAL OPERATION SOURCES ARE NOT SHOWN.

GENERAL ARRANGEMENT
CONTROL & DIESEL GEN. BLDG
GRADE FLOOR PLAN EL 737'0"
CLINTON POWER STATION
UNIT 1 & 2
ILLINOIS POWER COMPANY
CLINTON ILLINOIS

SARGENT & LUNDY

FIG. 1

0

POST ACCIDENT RADIATION MAP

Appendix C

Nomogram for Shielding of Point Sources

Figures 2 and 3 in this appendix were prepared to simplify the calculation of shield thickness for lead and iron.

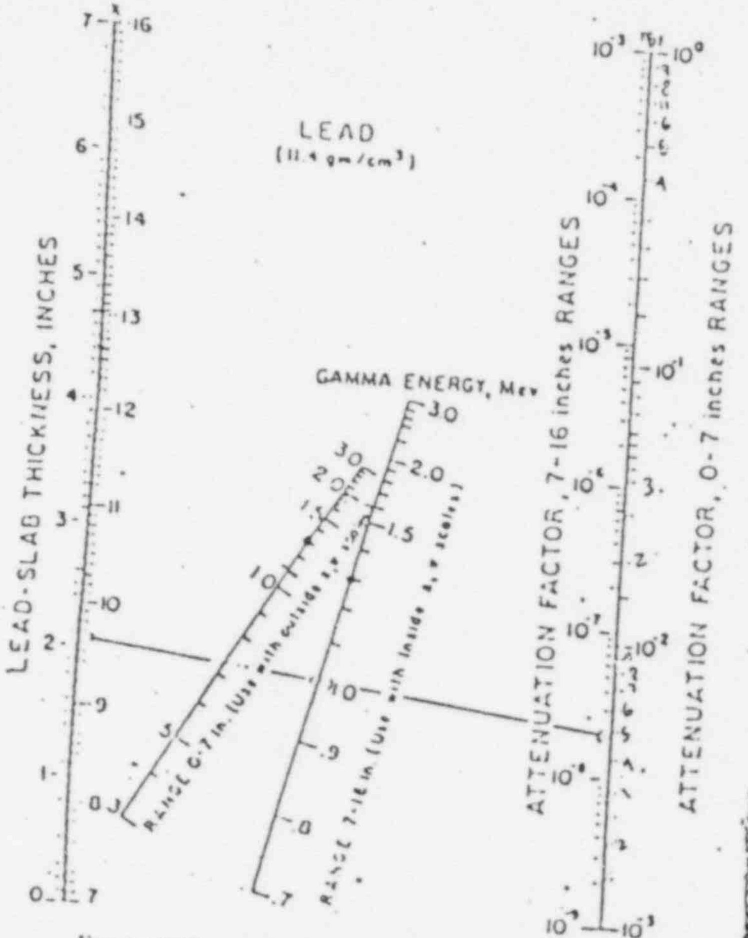


FIGURE 2. *Gamma attenuation with buildup in lead.*

(Chersell, D. G., Gamma attenuation with buildup in lead and iron, Nucleonics 15, No. 1, 42 (1957)).

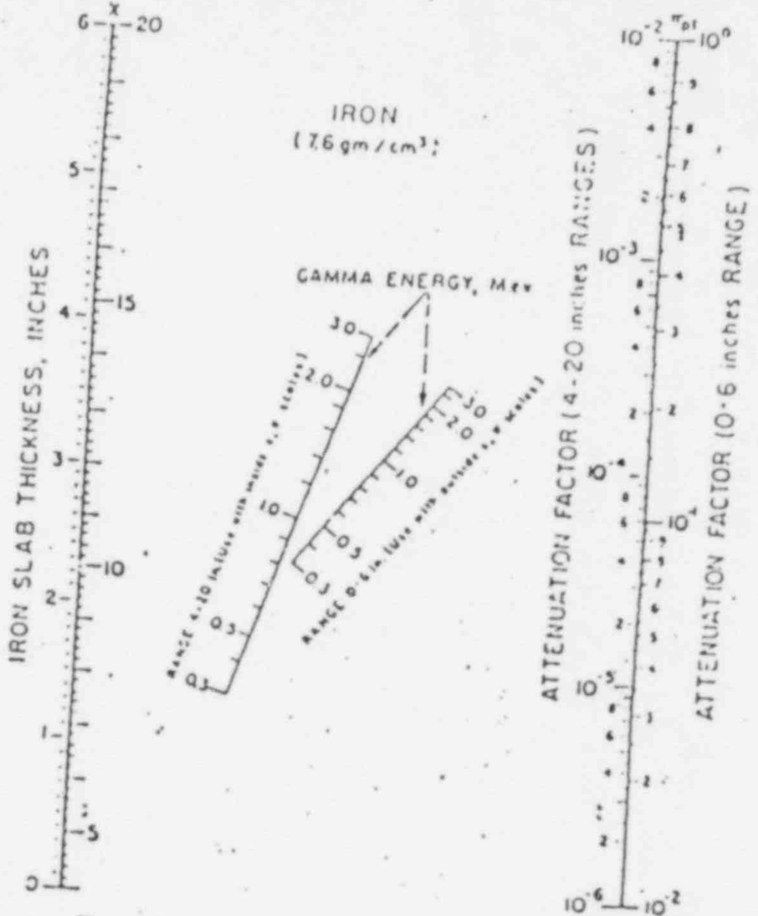


FIGURE 3. *Gamma attenuation with buildup in iron.*

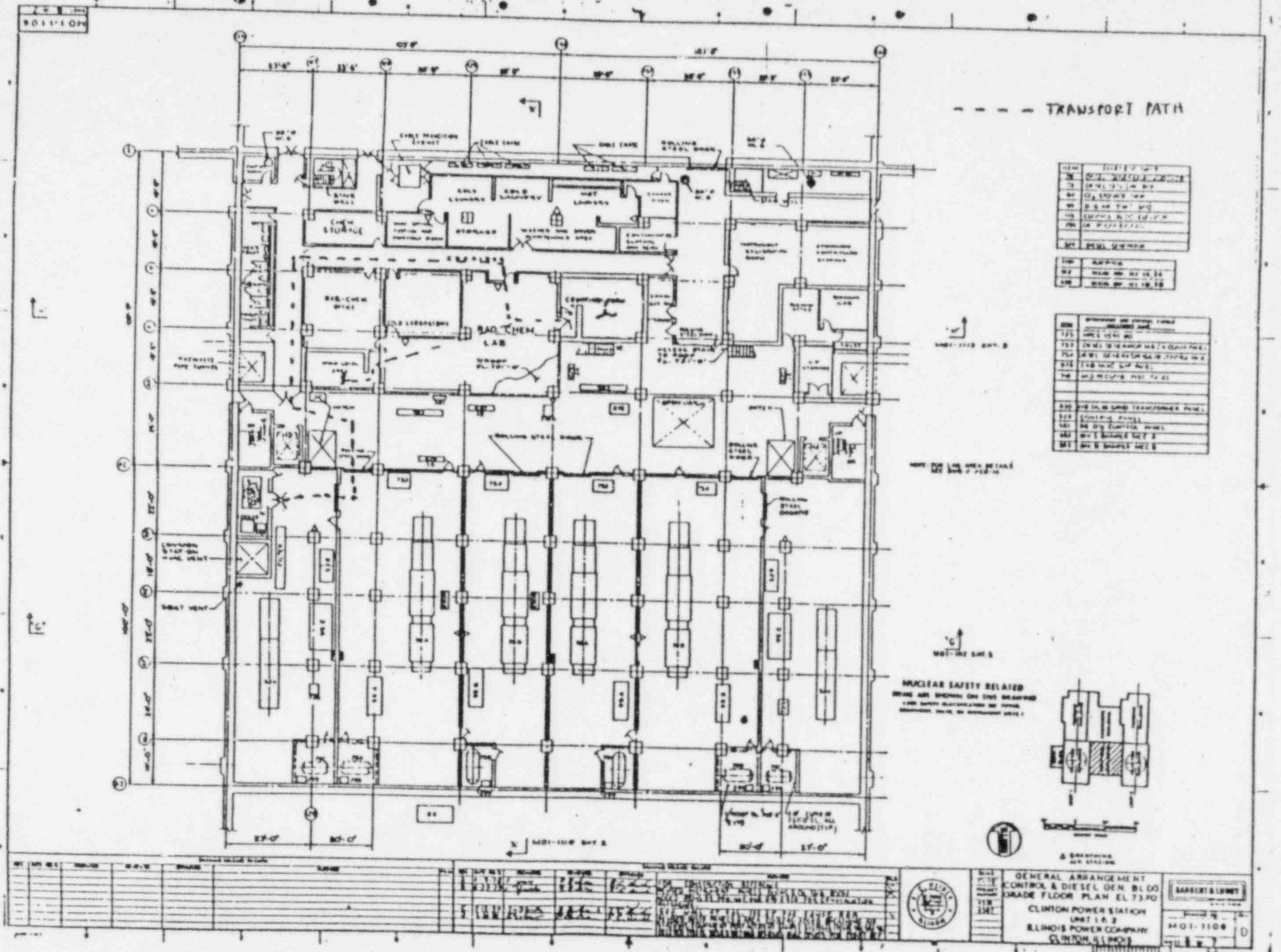
(Chersell, D. G., Gamma attenuation with buildup in lead and iron, Nucleonics 15, No. 1, 42 (1957)).

They represent the attenuation for point sources of gamma emitters of various energies and include the scattered radiation. Over most of the range the accuracy is reported to be 10 percent or better, although at extreme thicknesses the accuracy is about 25 percent.

The example given in figure 2 answers the question as to how thick a lead shield must be to reduce the dose rate

FIG. 2

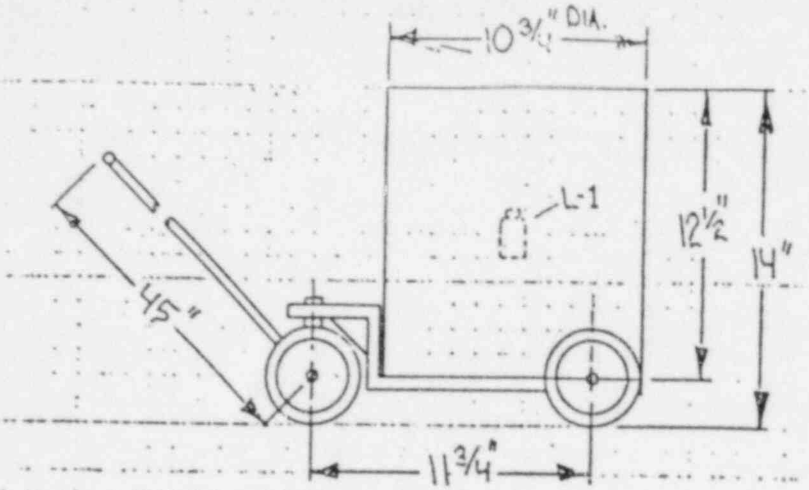
FiG. 3



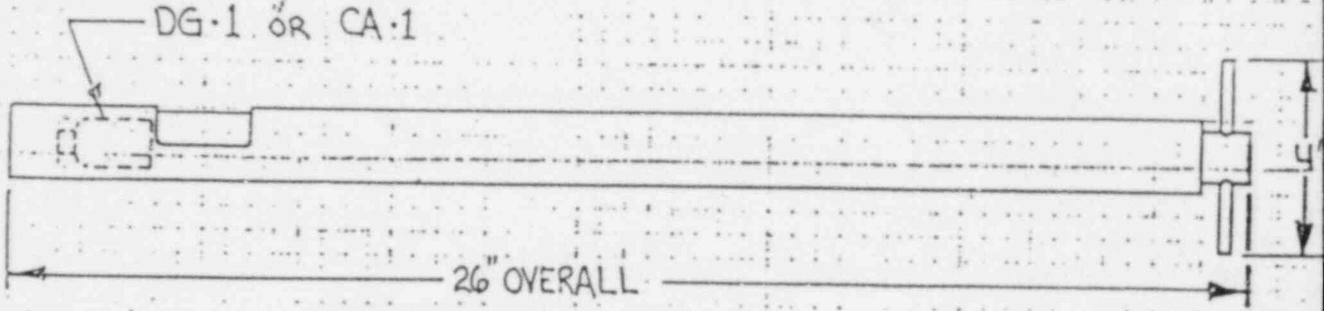
TRANSPORT PATH TO THE RAD CHEM LABORATORY

SKETCH: MOBILE PIG FOR UNDILUTED LIQUID SAMPLE

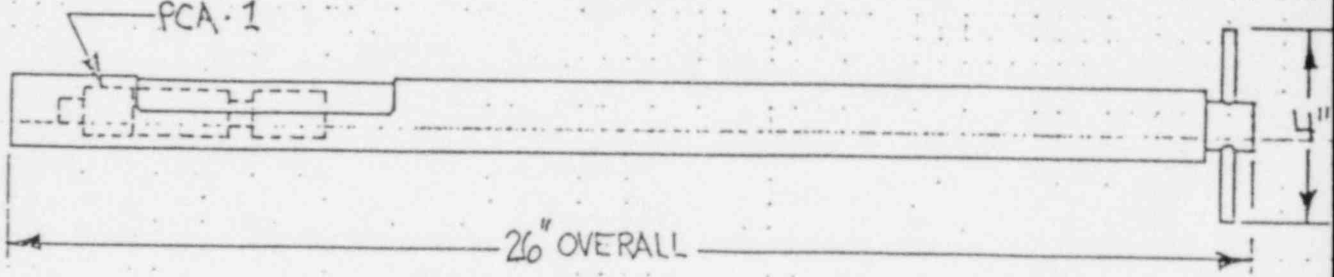
FIG. 4



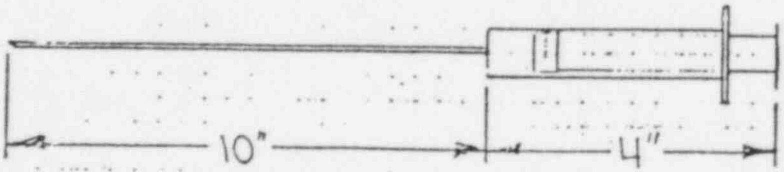
SKETCH: GAS SAMPLE TONGS & LIQUID ALIQUOTER



TONGS T-1, T-2



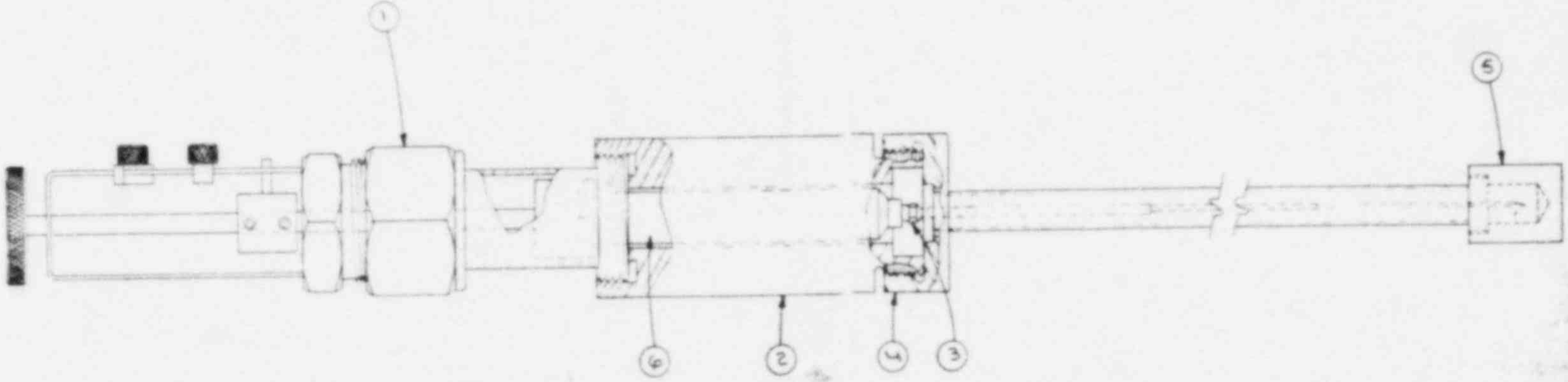
TONG T-3



ALIQUOTER DL-1

FIG. 5

P/N	ITEM	REQD	DESCRIPTION
6-00373A	1	1	MES TON ASSEMBLY, DWG # 190-67-128-3
6-00371A	2	1	SYRINGE SHIELD, DWG # 190-67-082-1
6-00374A	3	1	SAMPLE NEEDLE WELDMENT, DWG # 190-67-084-1
6-00365A	4	1	NEEDLE SHIELD ASS'Y, DWG # 190-67-085-1
2-00946A	5	1	NEEDLE SHIELD END CAP, DWG # 190-67-124-2
4-00773A	6	1	SYRINGE, 10cc



P/N 7-00163A

DO NOT SCALE DWG

ENG REVIEW

BY DATE

TOLERANCES
ALL DIMENSIONS UNLESS
OTHERWISE SPECIFIED
FRACTIONAL
DECIMAL
ANGULAR

DWG # 5656

SENTRY EQUIPMENT CORP.

DILUTED LIQUID SAMPLE
NEEDLE ASSEMBLY

DATE 5-7-81

SCALE FULL