ILLINOIS POWER COMPANY

CLINTON POWER STATION - UNIT #1

POST ACCIDENT SAMPLING

SYSTEM EVALUATION REPORT

M7541/M254/s

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LIST OF ACRONYMS

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

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

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 Requirement". 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

Attachment 2 - 10CFR50.59 SAFETY EVALUATION L&S No. 93-0058

Attachment 3 - 10CFR50.59 SAFETY EVALUATION L&S No. 93-0059

Attachment 4 - 10CFR50.59 SAFETY EVALUATION L&S No. 93-0060

Attachment 5 - 10CFR50.59 SAFETY EVALUATION L&S No. 93-0061

Attachment 6 - 10CFR50.59 SAFF Y EVALUATION L&S No. 93-0062

Attachment 7 - 10CFR50.59 SAFETY EVALUATION L&S No. 93-0063

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 license 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 labcratories 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. RHR pump 1A or 1B effluent (Reactor coolant or Suppression pool water).

* The NRC concluded in CPS Safety Evaluation Report, Supplement 7, section 9.3.5, page 9-5, that CPS meets the requirements of item II.B.3 of NUREG 0737 without sump sampling capability.

Position 1 (Continued)

The gas samples routed to the system consist of:

1. Containment Atmosphere (2 sample locations, elevations 790 and 740 feet at azimuth (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.

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 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. The reading is displayed on an indicator 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.

Containment and Drywell atmosphere gas samples are passed near a radiation detector to inform the operator at the SMP of the radiation level. These gases 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 a 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.

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 two grab samples (undiluted liquid and diluted liquid); performing pH inline; 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 pH

The analysis for liquid pH is via inline analyzer. The total required time for the analysis is approximately 40 minutes.

Position 1 (Continued)

- 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 (Continued)

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) are analyzed in the laboratory by gamma spectroscopy for I-131 & Cs-137. Gas samples (containment and drywell) are analyzed by gamma spectroscopy in the lab for determination of 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_2O_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 Contrcl Room.

   The H_2O_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

Position 2 (Continued)

Containment. The Containment Atmosphere Monitoring H_2O_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 may be 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. The bottle can be transported offsite for analysis.

The PASS also provides approximately a 1,000:1 diluted liquid grab sample. The sample is prepared by mixing a "bite" of undiluted, degassed, depressurized liquid with demineralized H_2O in a behind-shield, septum equipped mixing chamber. Following sample dilution, 10cc of the sample is 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.

Chloride Content

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

Position 2 (Continued)

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 Oxygen and Dissolved Hydrogen

As stated in NUREG-0853, "Safety Evaluation Report Related to the Operation of Clinton Power Station, Unit No. 1," Supplement No. 5, subsequent to the TMI-2 incident, the need was recognized for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable PASS are specified in NUREG-0737, Item II.B.3. Supplement 7 to NUREG 0853 reaffirms that samples obtained from the PASS are intended to determine the extent of core damage.

As referenced previously, the CPS Emergency Plan Implementing Procedure used to perform reactor core damage estimations is based on General Electric NEDO-22215, "Procedures for the Determination of the Extent of Core Damage Under Accident Conditions," dated August, 1982. This procedure provides estimates of the degree of reactor core damage based on primary system water samples, Drywell and Containment atmosphere samples, Drywell/Containment radiation levels, core uncovery time, and Containment atmosphere hydrogen levels. The reactor water samples are analyzed by gamma spectroscopy for Iodine-131 (I-131) and Cesium-137 (Cs-137). The Drywell and Containment atmosphere samples are analyzed by gamma spectroscopy for Krypton-85 (Kr-85) and Xenon-133 (Xe-133).

The CPS core damage estimation procedure does not utilize reactor water dissolved hydrogen or dissolved oxygen. Under accident conditions when the reactor vessel is depressurized, the concentration of dissolved gasses in the reactor water would have little meaning. Containment atmosphere hydrogen levels (obtained from 1E redundant H_2O_2 monitoring systems) can be used to provide a measure of the extent of metal water reaction which, in turn, can be used to estimate the degree of fuel cladding damage, in lieu of reactor water dissolved hydrogen values. Dissolved oxygen (which is an already known parameter) could be useful for ensuring proper steps are taken to minimize corrosion of reactor internals but does not provide useful data for core damage estimation (which is the purpose of the PASS as noted in the first paragraph).*

* NUREG/CR-4330 and SECY-93-087 conclude that primary system water dissolved gas samples are not necessary for post accident core damage estimations.

Position 2 (Continued)

CPS retracts the capability to perform reactor water dissolved oxygen and dissolved hydrogen sampling and analysis from the post-accident sampling system.

pH

CPS will measure pH inline with a Leeds and Northrup (L&N) No. 117489 probe. The probe is sealed and has automatic temperature compensation. The monitor will be an L&N model No. 7075 or better. The range of the pH analysis will be 1 to 13 or better. The 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).

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 System in the Radiological Chemistry Laboratory.

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

   A combination of KCl/AgCl gel-type pH electrode or better will be used to measure the liquid sample pH. The probe will be 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 or better considering advances in technology.

   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 with 1E power supplies 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. Technical Specification 3.3.7.5 requires plant shutdown if either CAM system is inoperable.

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 post accident conditions.

The Spectral Analysis 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 computer system tied in with a high efficiency high purity germanium (HPGe) detector. 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 independent inline H_2O_2 monitoring systems for the Containment and Drywell. The H_2/O_2 monitoring systems 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 (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 USAR 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 chloride 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

See position 2.

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 the 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.

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. 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 560 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.

2. 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.

3. 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.

Position 6 (Continued)

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 chromtograph. 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 5 cm 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; 31mRem 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 15 mRem/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.

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: pH in reactor coolant and hydrogen/oxygen concentration in Containment/Drywell 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.

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-003084 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). The H_2 quantity used in this iteration is taken from the Containment/Drywell Hydrogen/Oxygen Atmosphere Hydrogen Analyzer Systems. See position 2.

Position 8 (Continued)

2. Dissolved Oxygen

   See position 2.

3. Conductivity

   See position 2.

4. pH

   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.

   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.

For transporting samples offsite, CPS maintains a shipping container (cask) onsite with which to send samples to the offsite laboratory. The current shipping container 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 se 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.

Position 9 (continued)

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

The Spectral Analysis system used in the Radiological Chemistry laboratory will be capable of acquiring nuclide spectra and identifying major constituents of the spectrum as well as quantifying the major constituent. The system consists of a computer system tied in with a high efficiency high purity germanium detector. 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 μCi/cc to 10 Ci/cc can be analyzed.

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

b) The background radiation dose rate in the counting room is 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. 3. 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 concentration 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.

(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 Spectra Analysis 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 μCi/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 $\pm$15% in the 0.1 to 1.0 ppm chloride range and is $\pm$25% for higher concentrations. By calibrating at higher concentrations, the accuracy can be maintained at $\pm$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

See position 2.

E. Oxygen in RC

See position 2.

F. pH in RC (Via PASS Inline Analyzer)

pH is quantified by using a sealed probe (Leeds & Northrup (L&N) #117489) with automatic temperature compensation. The monitor is an L&N #7075 (or better). The pH probe is calibrated, in-place, using 2 buffer solutions.

a. Minimum Range = 1 to 13

b. Worst Case Accuracy = ±0.3 pH units (5-9 pH) & ±0.5 pH units (outside 5-9 pH) (Assuming a 10°C difference in temperature between calibration solution and PASS liquid temperature).

Position 10 (Continued)

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).

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 Chemical/Radiochemical Procedure 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 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.

Position 11 (Continued)

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.

* 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	pH	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 <u>and</u> CA)	179
Criterion Maximum (For RC or CA, single sample & analysis)	180

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	1.48

4 DRYWELL ATMOSPHERE: GAMMA

(a)	R/hr @cm/cc	618
(b)	mCi/cc	101

* Radiation dose rate 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

* Not obtained at CPS.

TABLE 5

PERSONNEL RADIATION EXPOSURE PREDICTIONS

ACTION		EXPOSURE (mR) hands	body
PASS Exercises		1305	171
Ambient contribution (after sampling)		300	300
Transporting Samples	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)		1911mR(1.9 Rem)	603mR(0.6 Rem)
Criterion Maximum		75 Rem	5 Rem

TABLE 6

CHEMICAL ANALYSIS CAPABILITY

Analysis	Method/Instrument	Minimum Range	Worst Case Accuracy
Gross Activity	Spectral Analysis System (in the laboratory)	1μCi/cc to 10Ci/cc	± 50%
Gamma Spectrum	Spectral Analysis System (in the laboratory)	1μCi/cc to 10Ci/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)
pH	pH Probe (inline)	1-13	± 0.3 pH units (5-9 pH) ± 0.5 pH units (outside 5-9 pH)

TABLE 7

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

Connection # of 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).

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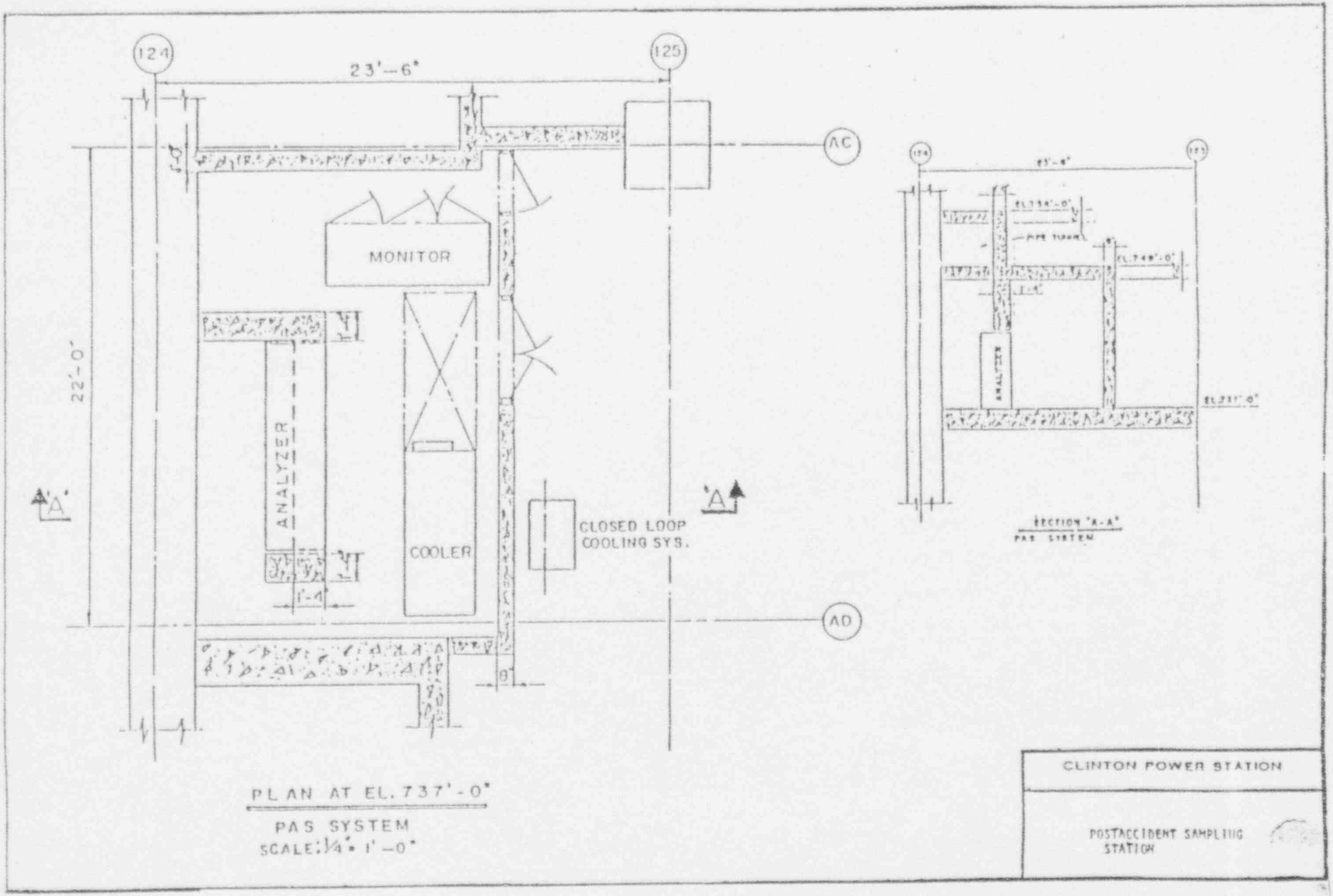
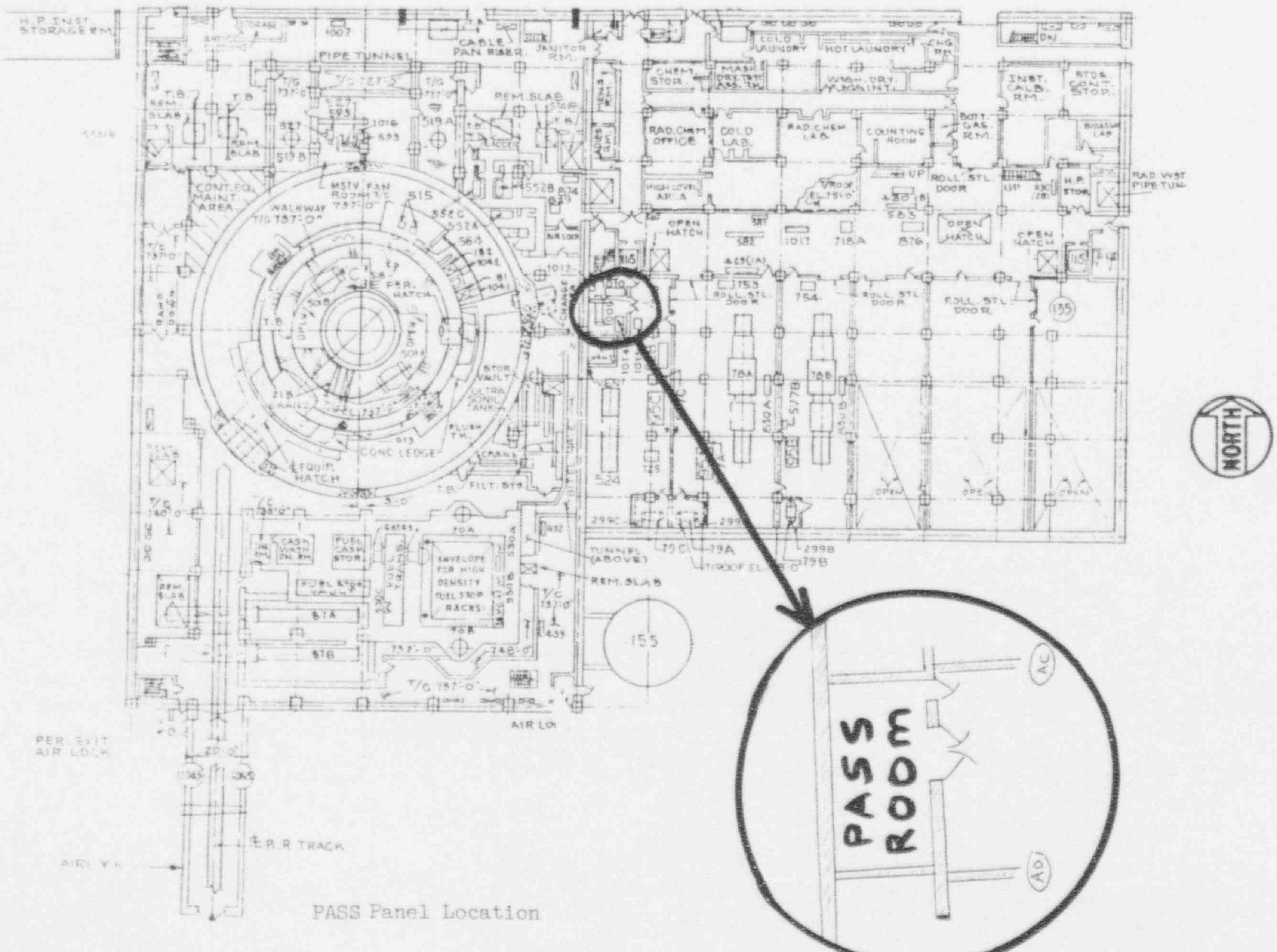
Figure 1



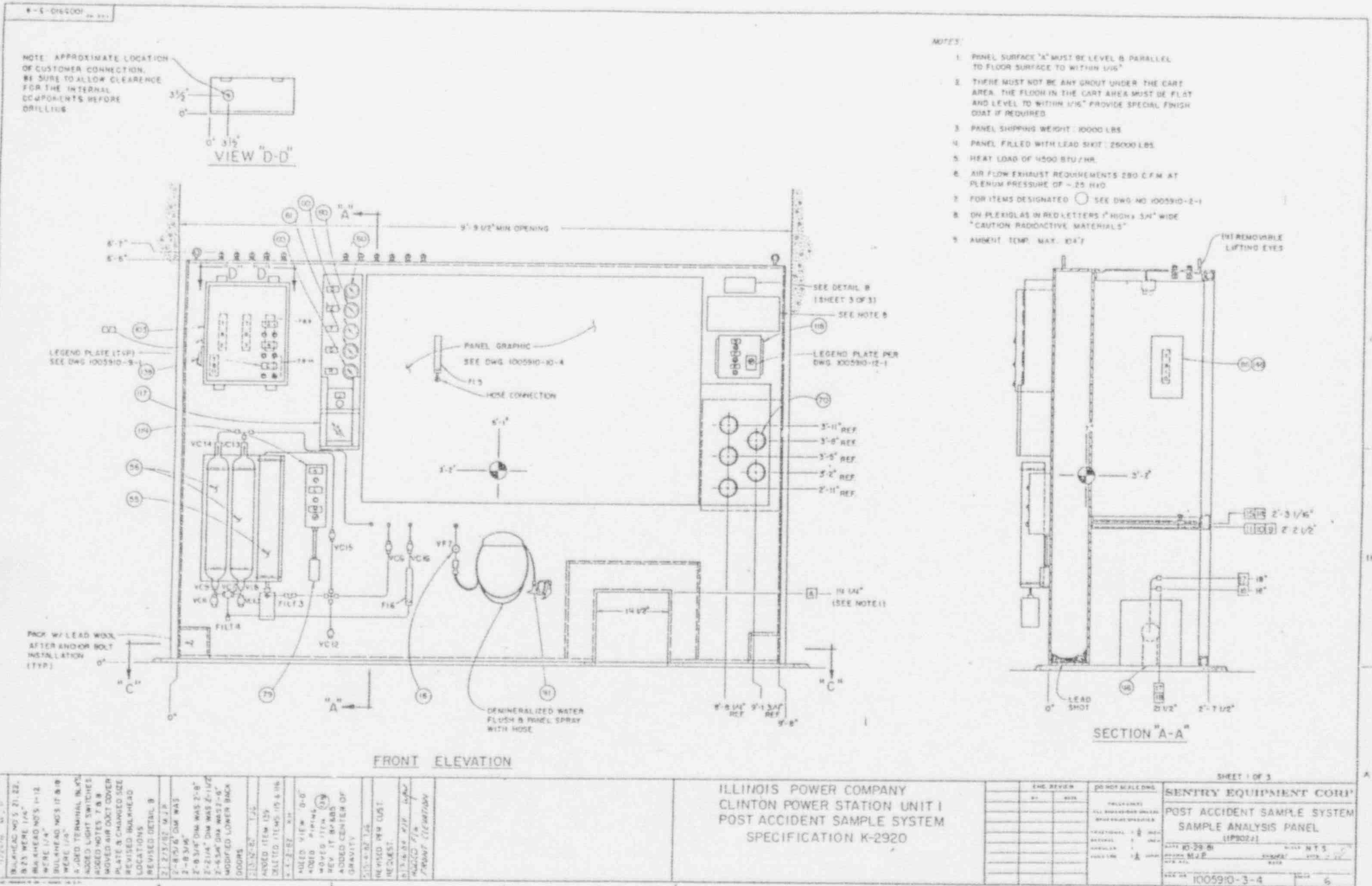
PASS Components Physical Layout

Figure 2



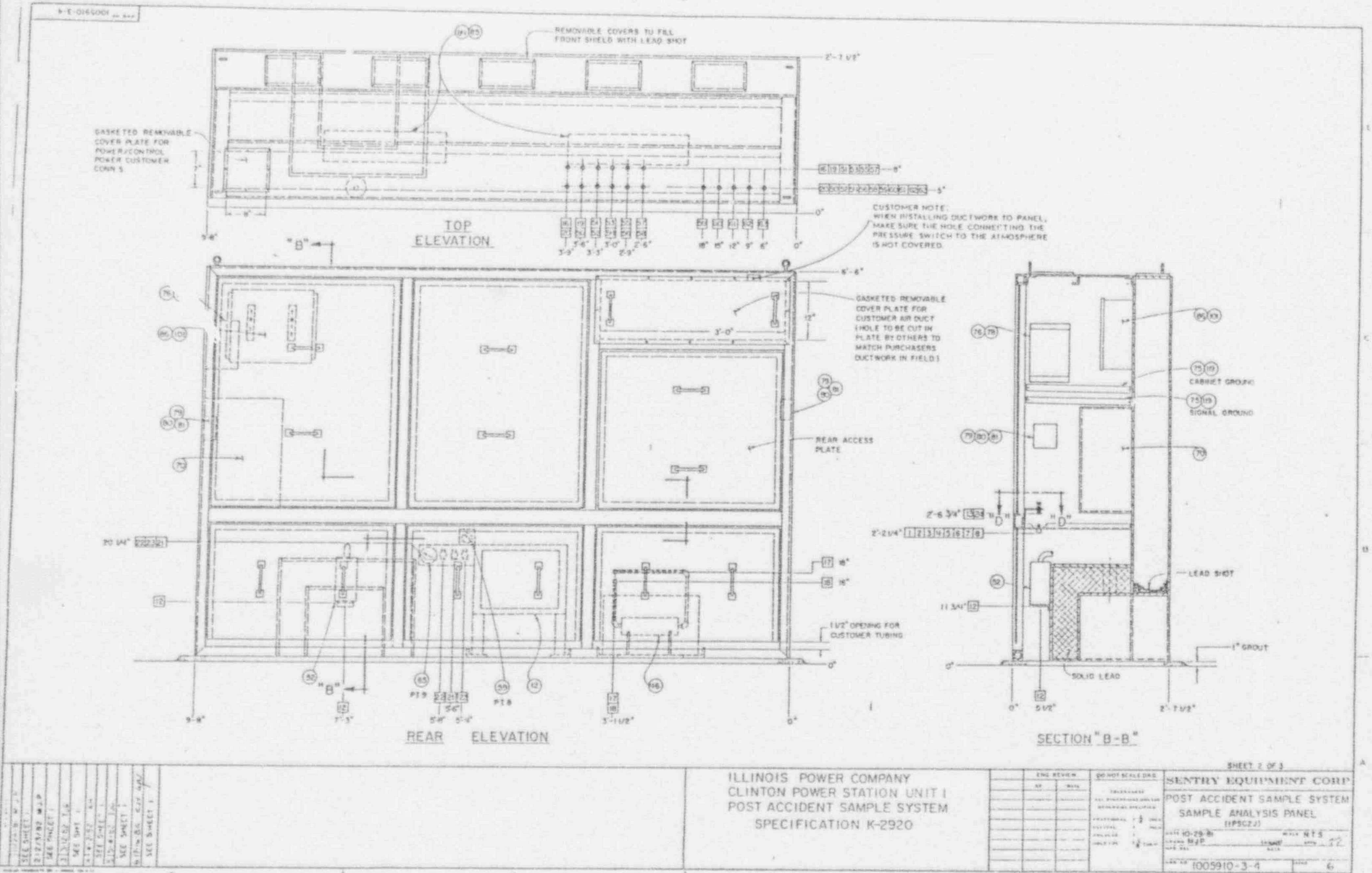
PASS Panel Location

Figure 3



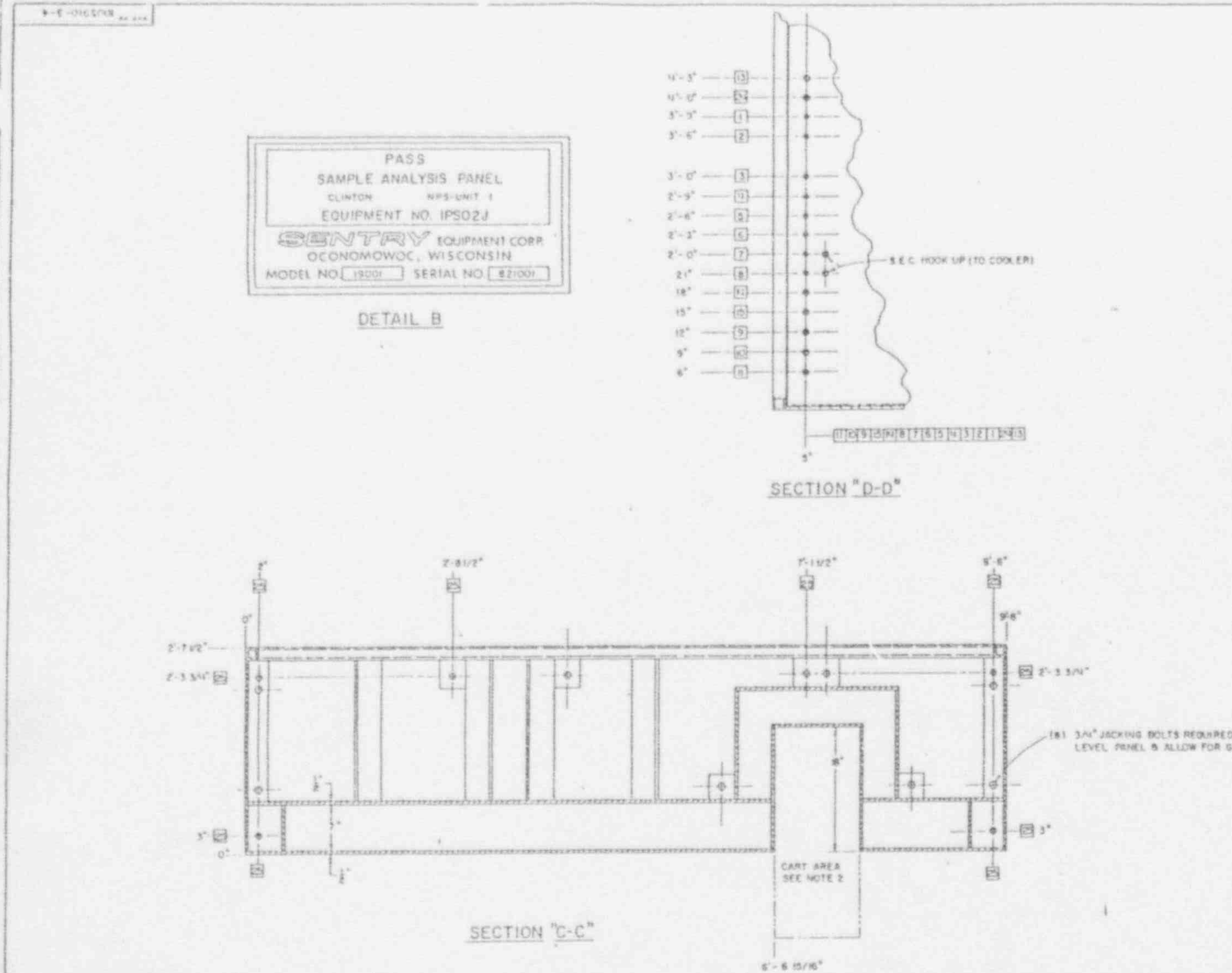
Sample Analysis Panel (Front View)

Figure 4



Sample Analysis Panel (Top and Read View)

Figure 5



LOC.	DESCRIPTION
1	3/8" SOCKET WELD CONN RHR PUMP SAMPLE 1B IN
2	3/8" SOCKET WELD CONN REACTOR WATER CLEAN UP EFFLUENT IN
3	3/8" SOCKET WELD CONN CONTAINMENT FLOOR DRAIN SUMP IN
4	3/8" SOCKET WELD CONN DRYWELL FLOOR DRAIN SUMP IN
5	3/8" SOCKET WELD CONN CONTAINMENT EQUIPMENT DRAIN SUMP IN
6	3/8" SOCKET WELD CONN DRYWELL EQUIPMENT DRAIN SUMP IN
7	3/8" SOCKET WELD CONN PRIMARY COOLANT IN
8	3/8" SOCKET WELD CONN LIQUID SPARE
9	3/8" COMPRESSION BULKHEAD GAS SPARE
10	3/8" COMPRESSION BULKHEAD DRYWELL ATMOSPHERE IN
11	3/8" COMPRESSION BULKHEAD CONTAINMENT ATMOSPHERE IN
12	1/4" NPT SUMP GRAVITY DRAIN OUT
13	1/4" COMPRESSION BULKHEAD WASTE LINE OUT - LIQUID
14	1/2" COMPRESSION BULKHEAD CONTAINMENT WASTE OUT - GAS
15	1/2" COMPRESSION BULKHEAD CONTAINMENT WASTE OUT - GAS
16	1/4" COMPRESSION BULKHEAD DEMINERALIZED WATER IN
17	3/4" NPT COOLING WATER OUT
18	3/4" NPT COOLING WATER IN
19	1/4" COMPRESSION BULKHEAD INSTRUMENT AIR IN
20	1/4" COMPRESSION BULKHEAD NITROGEN IN
21	1/8" COMPRESSION BULKHEAD CHROMATOGRAPH GRADE ARGON
22	1/8" COMPRESSION BULKHEAD CAL GAS 2000ppm H_2 IN
23	1/8" COMPRESSION BULKHEAD CAL GAS 10% H_2 IN
24	1/4" COMPRESSION BULKHEAD PRV 1 WASTE LINE OUT
25	(8) 1 1/8" HOLES FOR 1" ANCHOR BOLTS
50	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 1
51	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 21
52	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 22
53	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 23
54	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 24
55	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 25
56	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 26
57	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 27
58	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 28
59	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 31
60	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 32
61	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 33
62	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 41
63	1/4" COMPRESSION BULKHEAD CONTROL AIR FOR AV 42

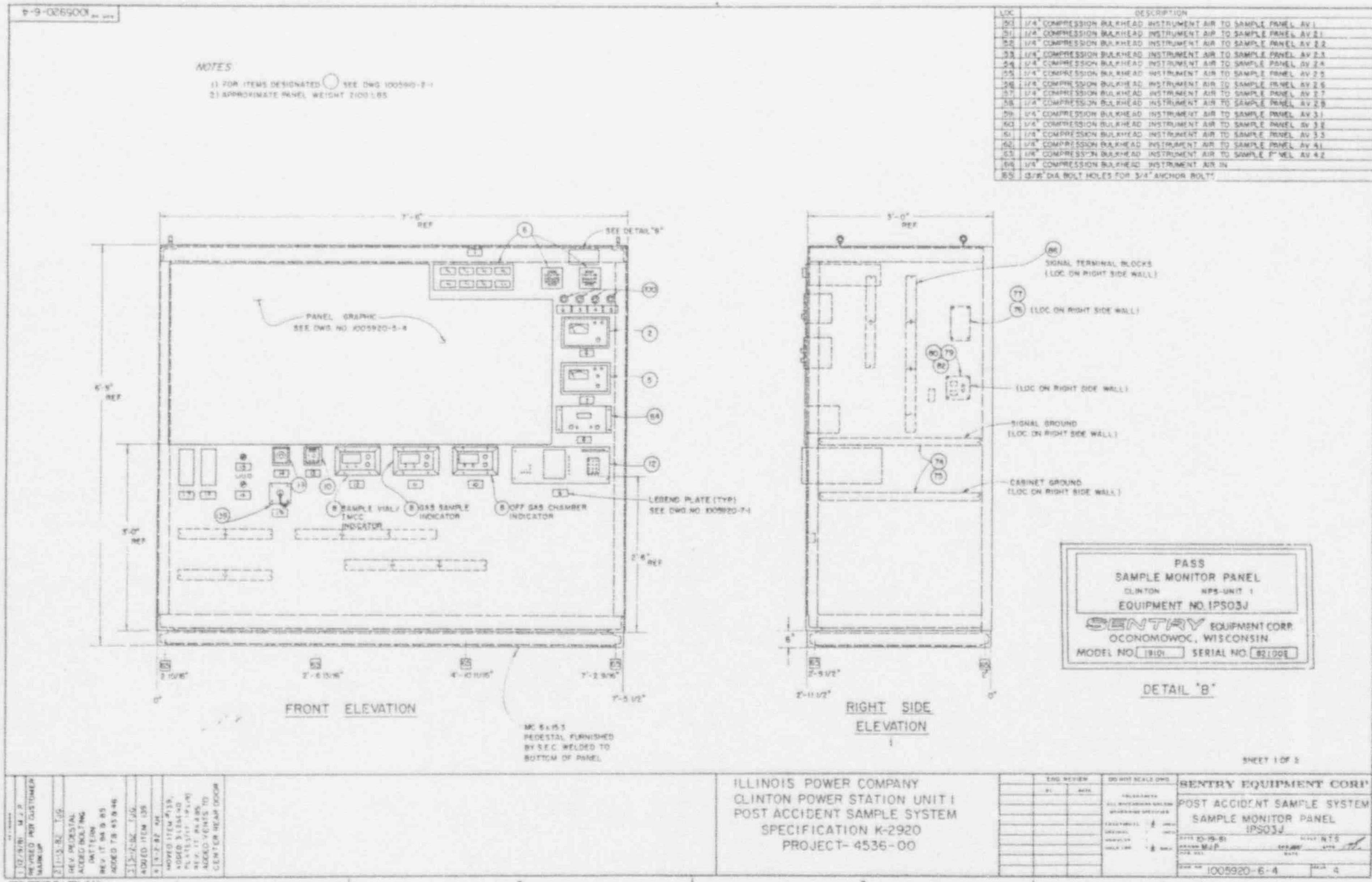
SHEET 3 OF 3

ILLINOIS POWER COMPANY
CLINTON POWER STATION UNIT I
POST ACCIDENT SAMPLE SYSTEM
SPECIFICATION K-2920

SENTRY EQUIPMENT CORP
POST ACCIDENT SAMPLE SYSTEM
SAMPLE ANALYSIS PANEL
(IPS02J)
10-29-81
N.T.S.
1005910-3-4

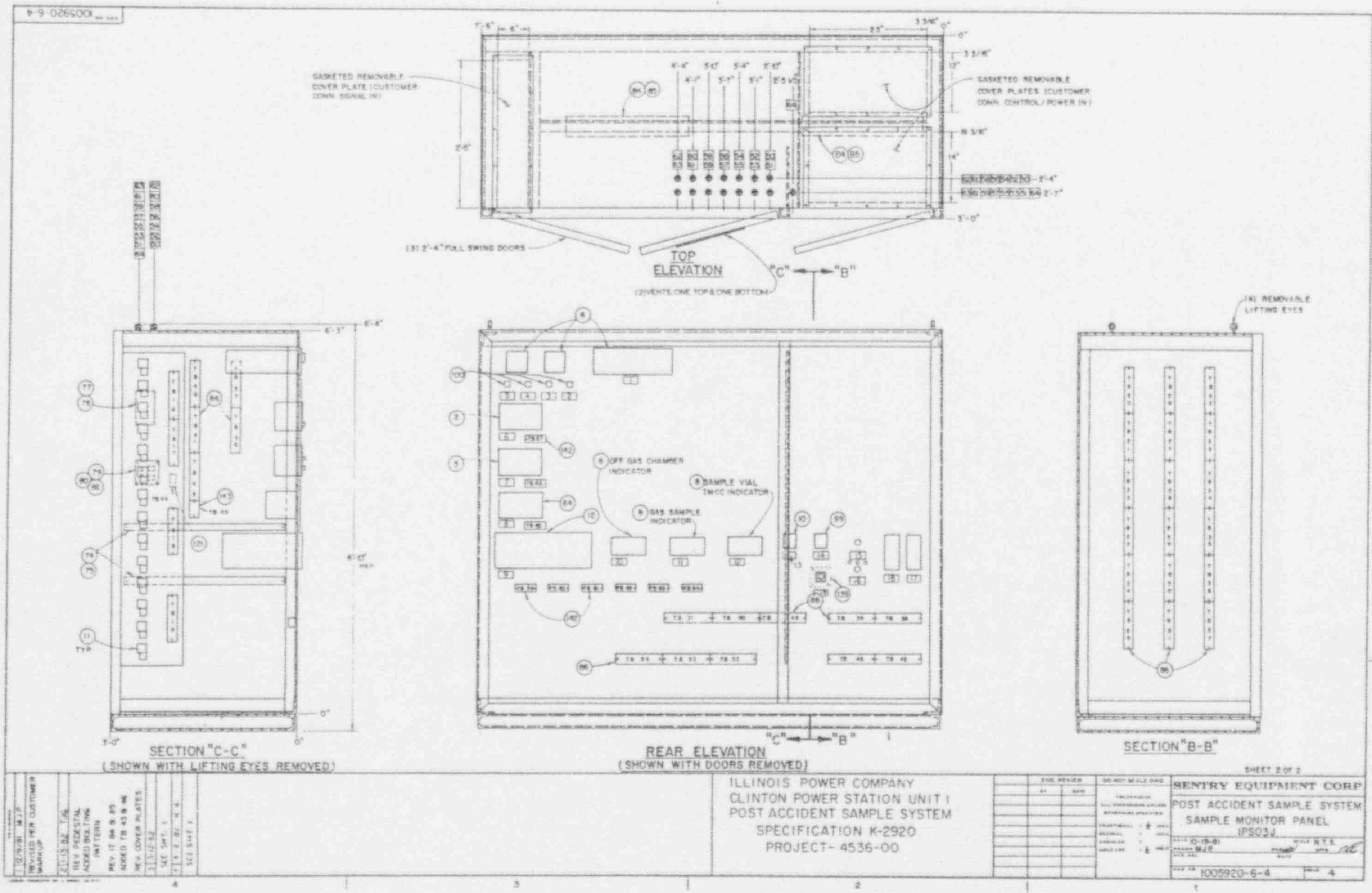
Sample Analysis Panel (Details)

Figure 6



Sample Monitor Panel (Front and Right View)

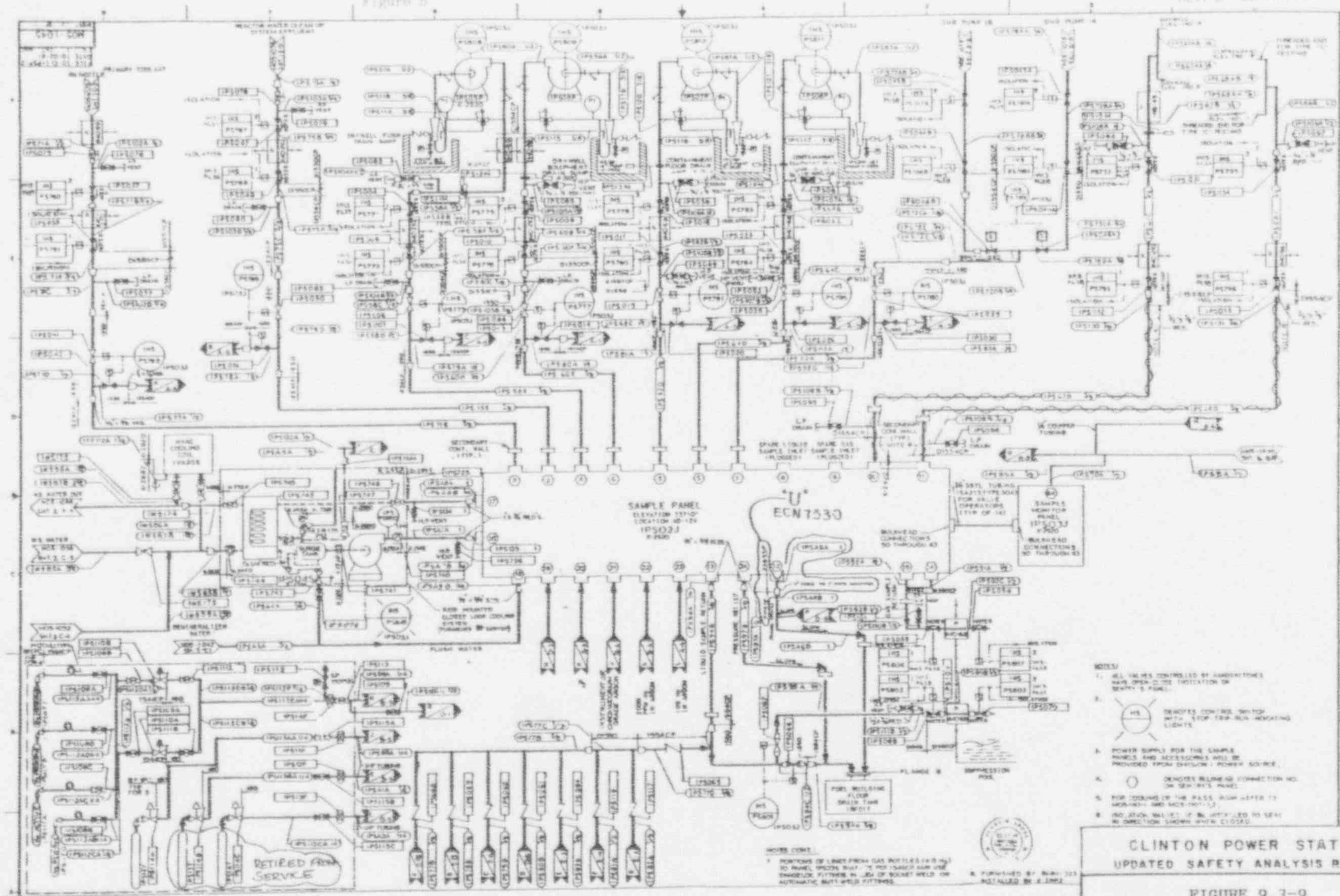
Figure 7



Sample Monitor Panel (Top and Rear View)

Figure 8

REV. 9 SEPT. 1991



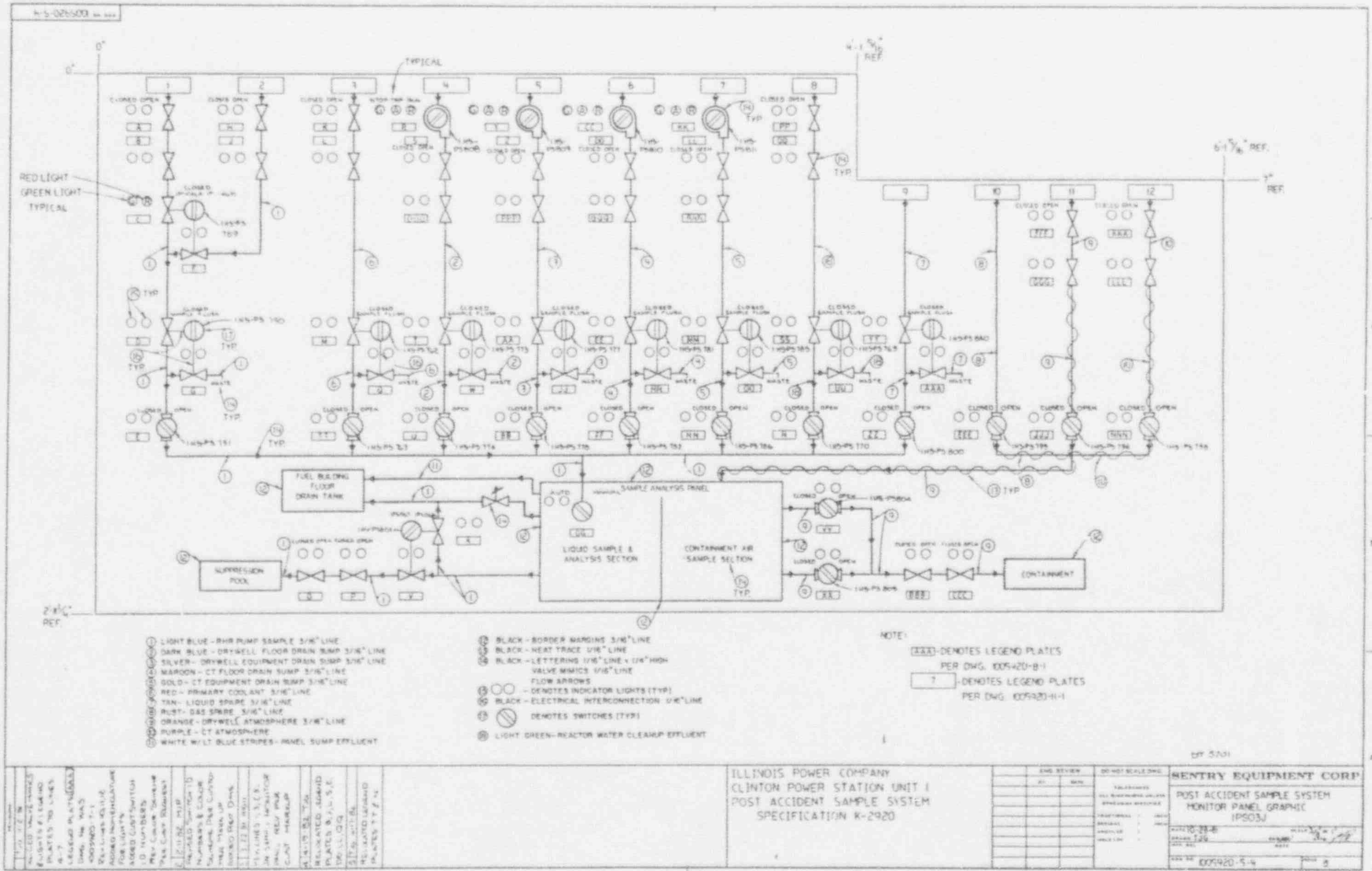
CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 9.3-9

POST ACCIDENT SAMPLE SYSTEM (PASS)

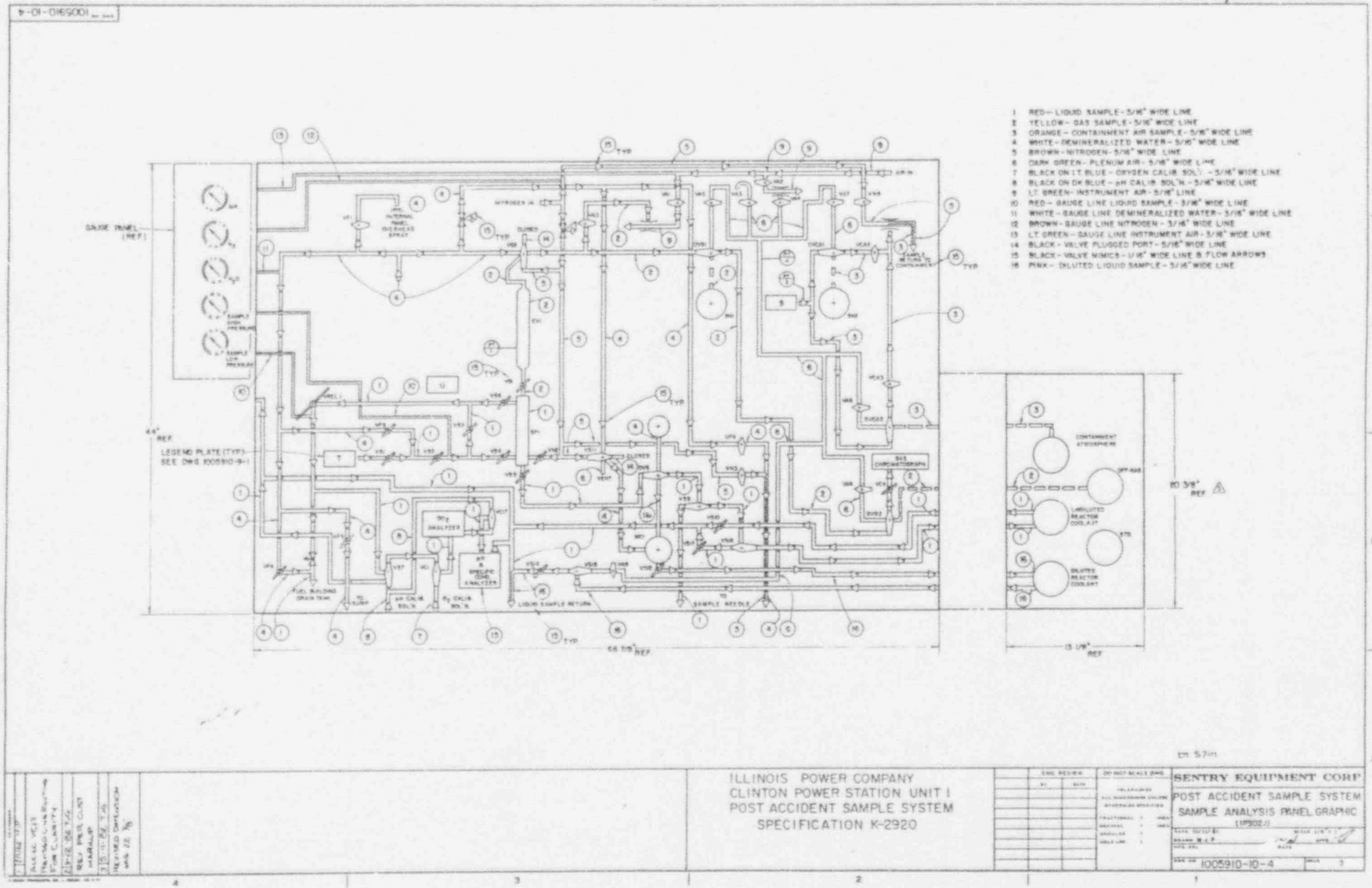
Post Accident Sample System (PASS)

Figure 9



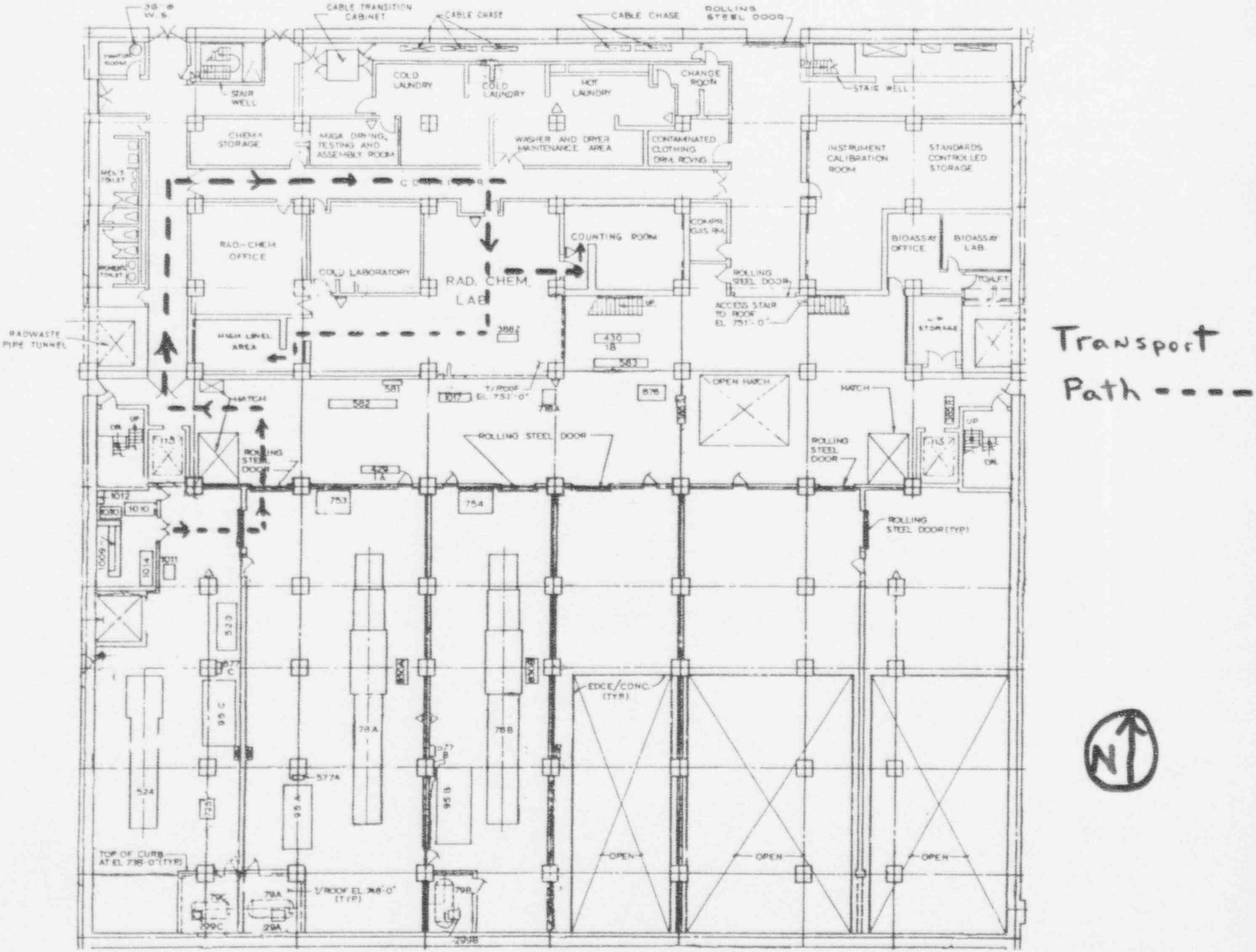
Sample Monitor Panel (Graphic)

Figure 10



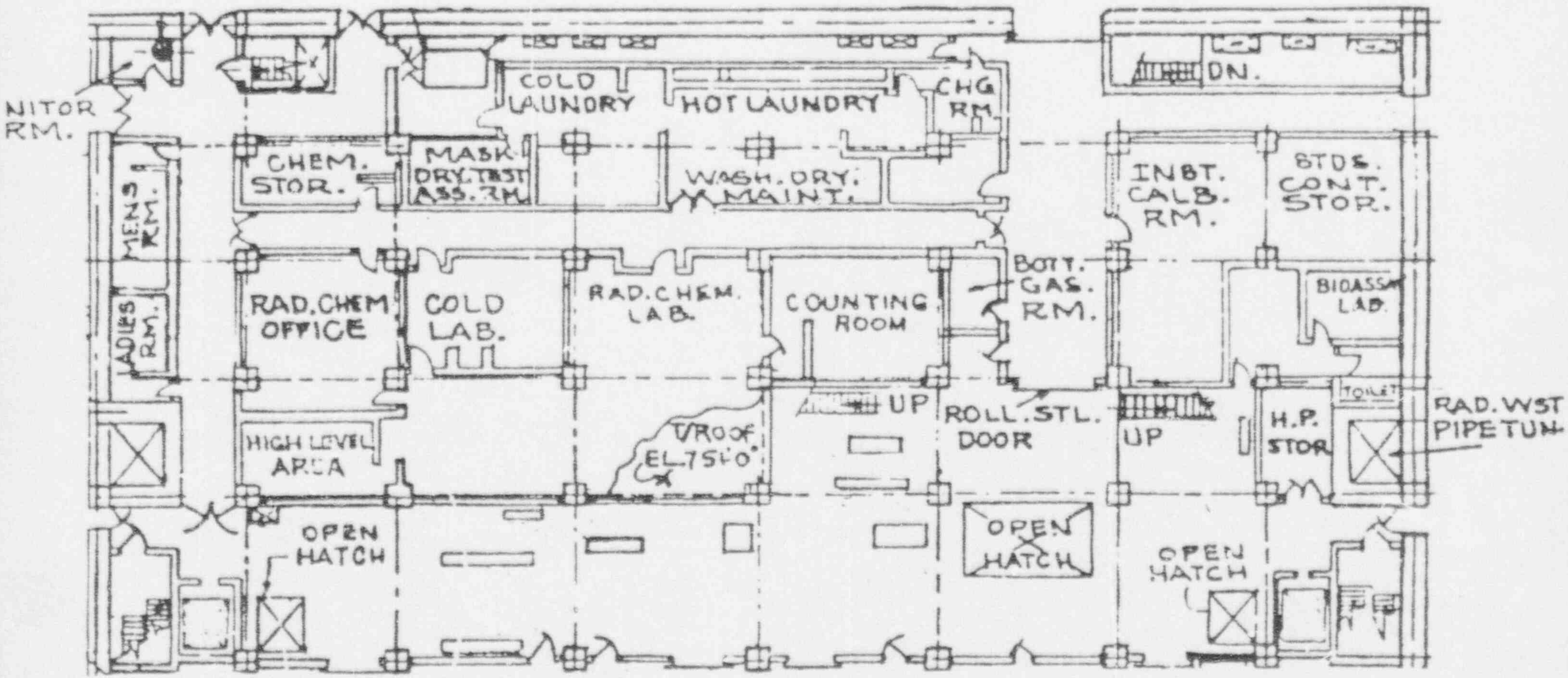
Sample Analysis Panel Graphic

Figure 11



Transport Path To Radchem Laboratories

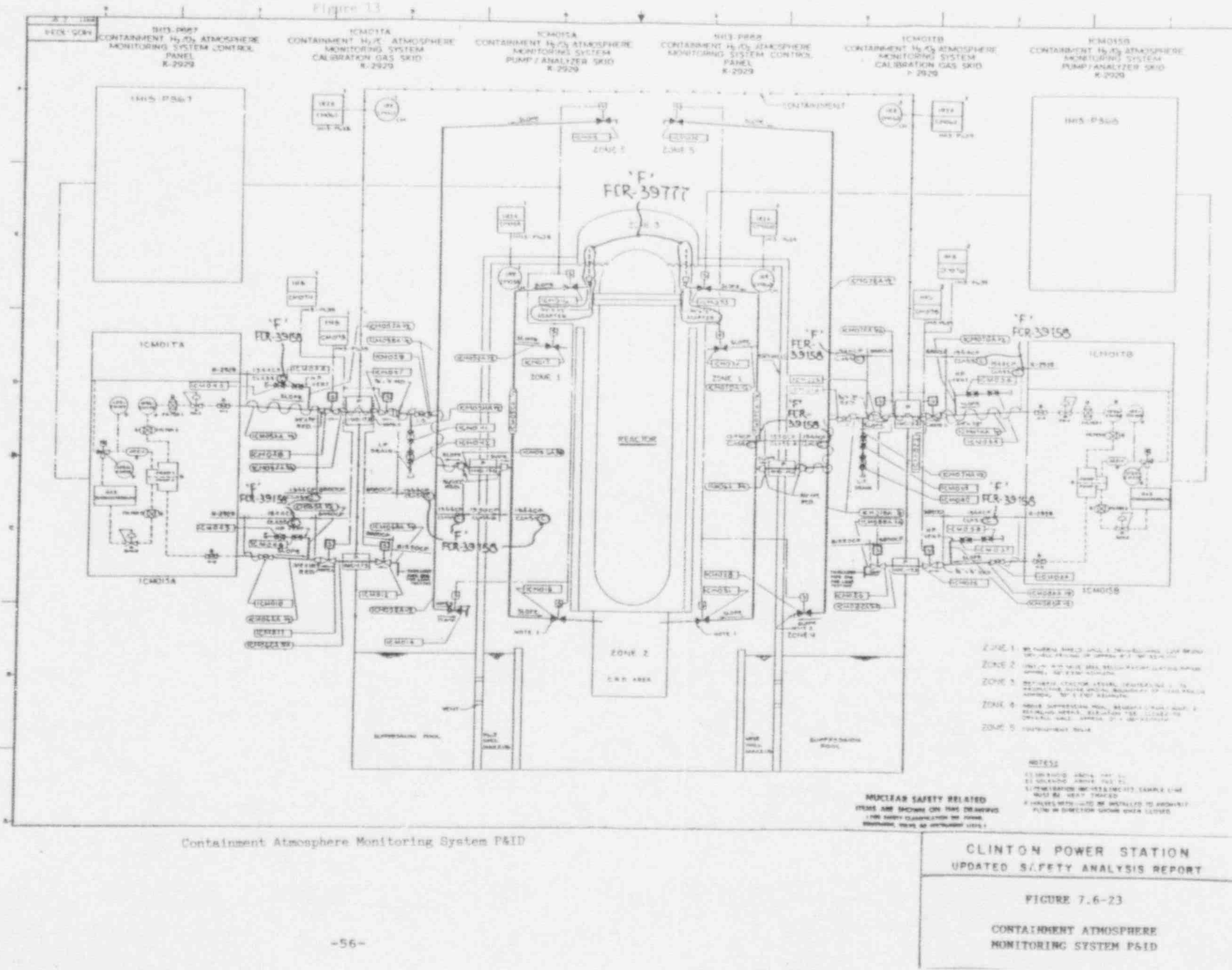
FIGURE 12



Control Building El. 737' 0"

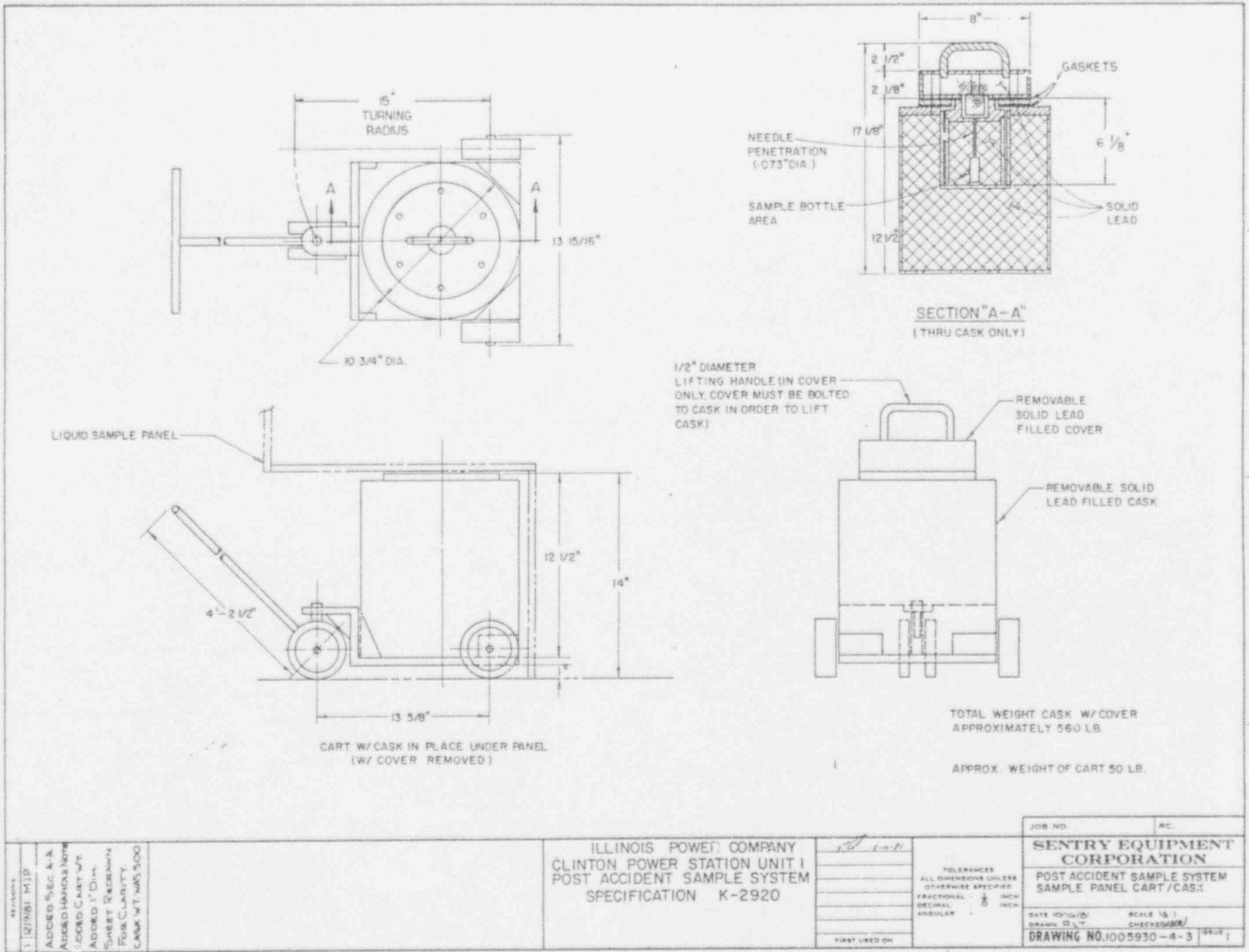
Layout of Station Laboratories, Counting Room, and Laundry

Figure 13



Containment Atmosphere Monitoring System P&ID

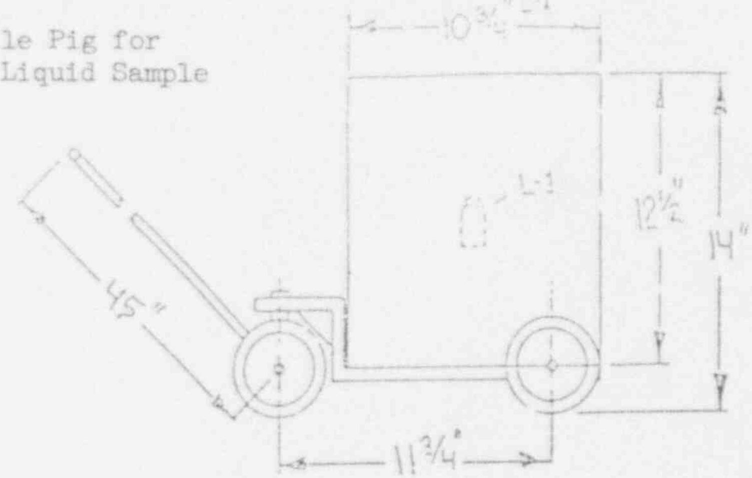
Figure 14



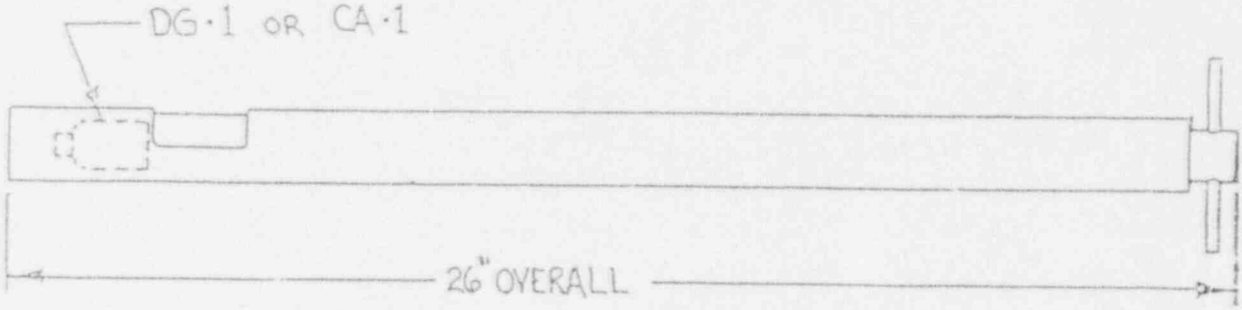
Sample Panel Cart/Cask

Figure 15

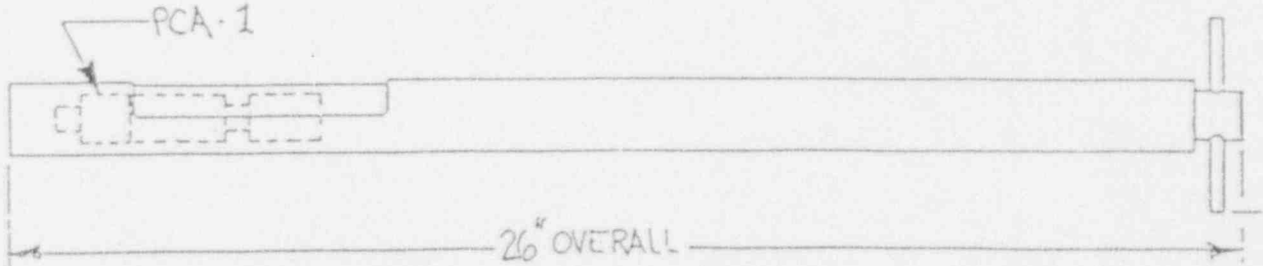
Sketch: Mobile Pig for
Undiluted Liquid Sample



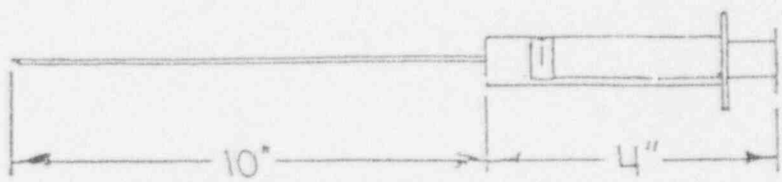
SKETCH: GAS SAMPLE TONGS & LIQUID ALIQUOTER



TONGS T-1, T-2



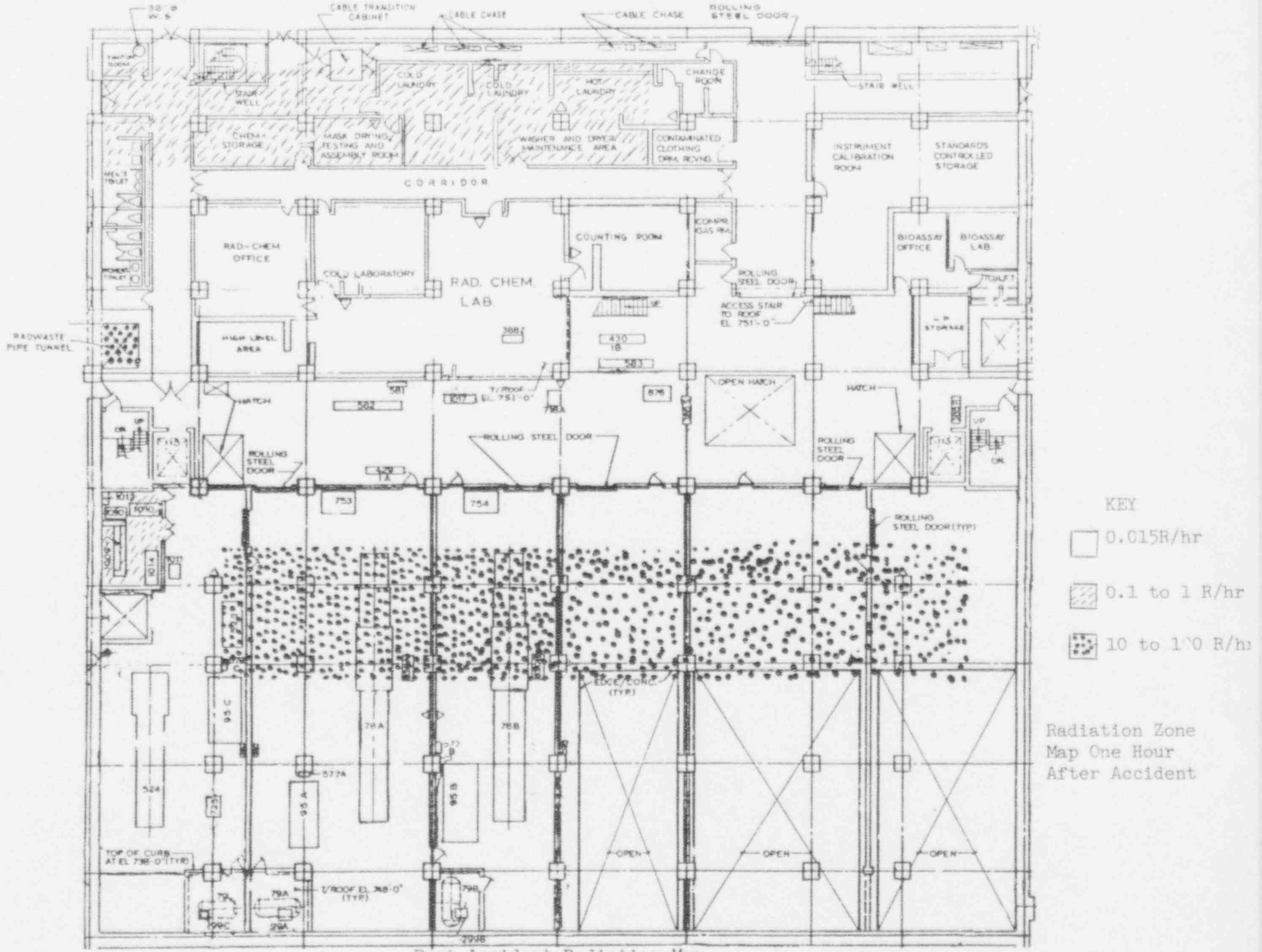
TONG T-3



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' (122 cm) 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

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

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

3. Chemical Analysis

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

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 5 cc (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 R/hr cc)(5cc)(15/60 hr)(1,000 mR/R)/ (15 cm)^2 = 77 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 an 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 = 15cm
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 = 15 cm
Hand distance = 10 cm (avg)

Hands

$$D = \frac{(193\ R/h)(2/60h)(1{,}000\ mR/R)}{(10\ cm)^2} + \frac{(14.2\ R/h)(2/60h)(1{,}000\ mR/h)}{(10\ cm)^2} = 69.1mR$$

Body

$$D = \frac{(193\ R/h)(2/60h)(1{,}000\ mR/R)}{(15\ cm)^2} + \frac{(14.2\ R/h)(2/60h)(1{,}000\ mR/h)}{(15\ cm)^2} = 30.7\ mR$$

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

<u>Hands & Body</u>

D = (1R/hr)(0.2)(1½ hr) = 0.3 R = 300 mR

2. <u>Transport path to the Rad Chem laboratory</u>

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 three samples may be taken to the laboratory (i.e. undiluted liquid, diluted liquid, and Containment air).

Ambient level = 15 mR/hr

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

<u>Hands & Body</u>

D = (15 mR/hr)(80/60 hr) = <u>20 mR</u>

3. <u>Rad Chem Laboratory</u>

The radiation level in the lab is a maximum of 15 mR/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

<u>Hands & Body</u>

D = (15 mR/hr)(3 hrs) = <u>45 mR</u>

<u>Summary</u>

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

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

TABLE 3

Personnel Radiation Exposure Predictions

ACTION		EXPOSURE (mR) hand	EXPOSURE (mR) body
PASS Exercises		1305	171
	Ambient contribution (after sampling)	300	300
Transporting Samples	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)		1911 mRem	603 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."

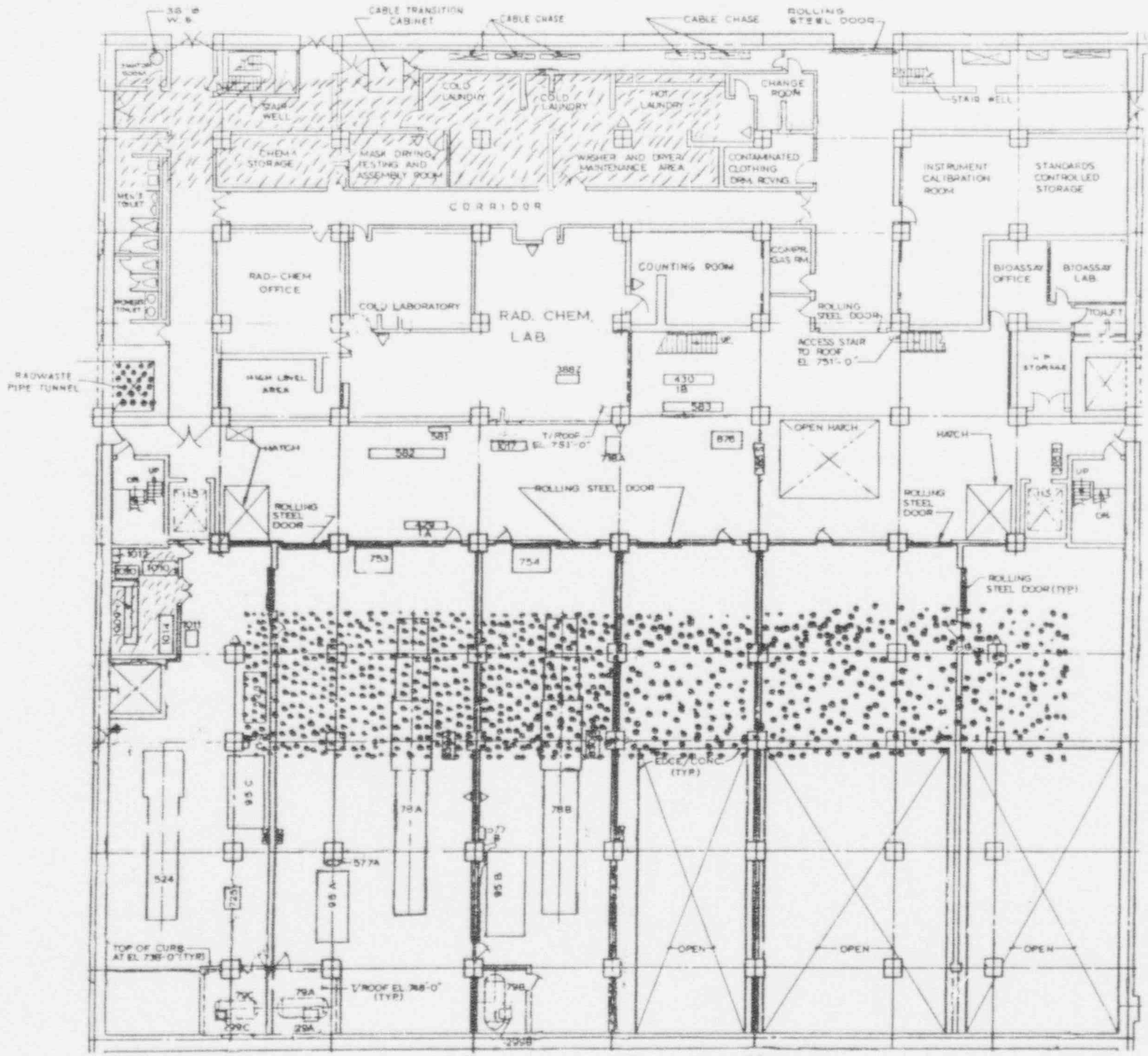
2. Sentry Equipment Corporation (SEC) Specification B10-01.

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)

Figure 1



KEY

0.015 R/hr

0.1 to 1 R/hr

10 to 100 R/hr

Radiation Zone
Map One Hour
After Accident

Post Accident Radiation Map

Appendix C

Nomograms 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.

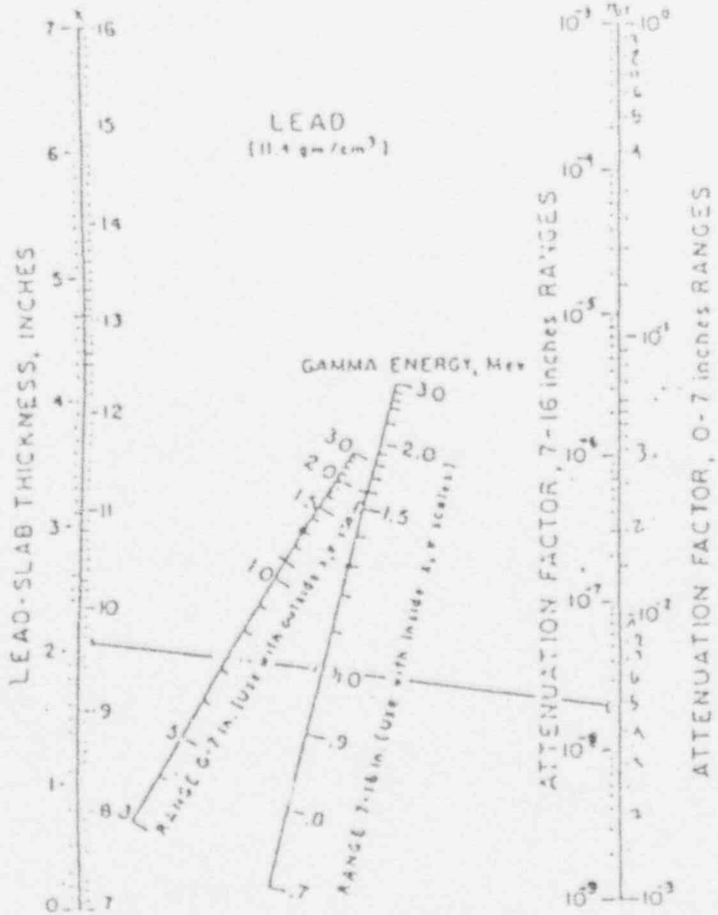


FIGURE 2. *Gamma attenuation with buildup in lead.*

(Chastell, D. G., Gamma attenuation with buildup in lead and iron, Nucleonics 11, No. 1, 41 (1953)).

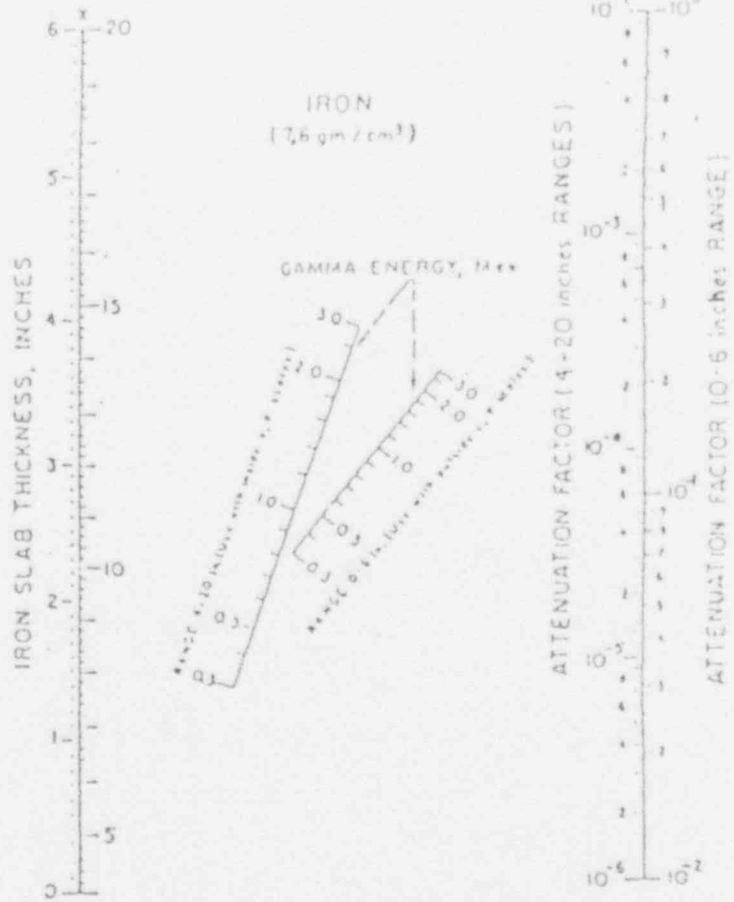


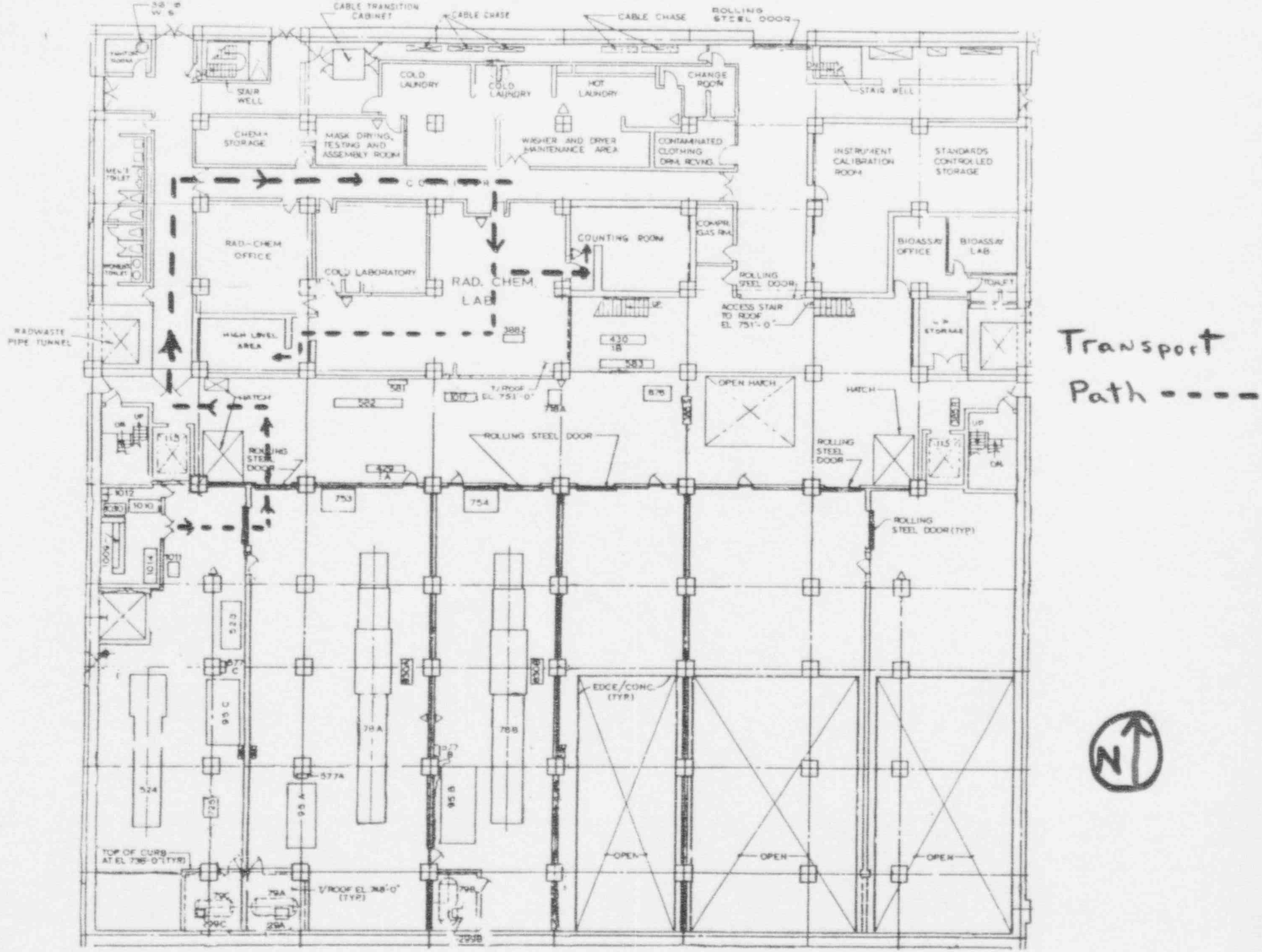
FIGURE 3. *Gamma attenuation with buildup in iron.*

(Chastell, D. G., Gamma attenuation with buildup in lead and iron, Nucleonics 11, No. 1, 41 (1953)).

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.

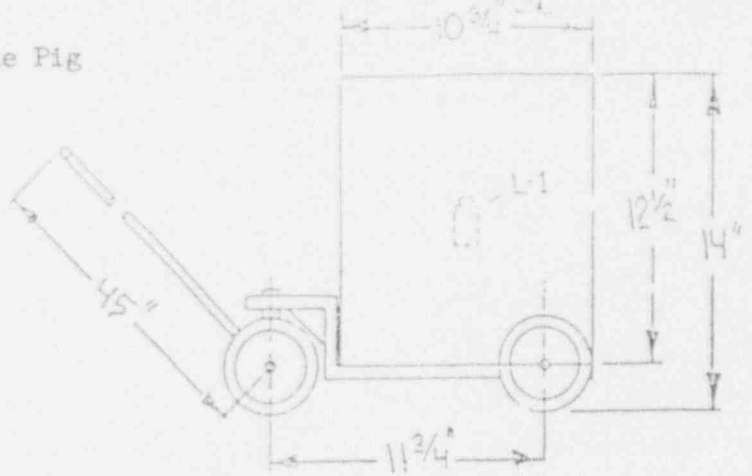
The example given in figure 2 answers the question as to how thick a lead shield must be to reduce the dose rate

Figure 3

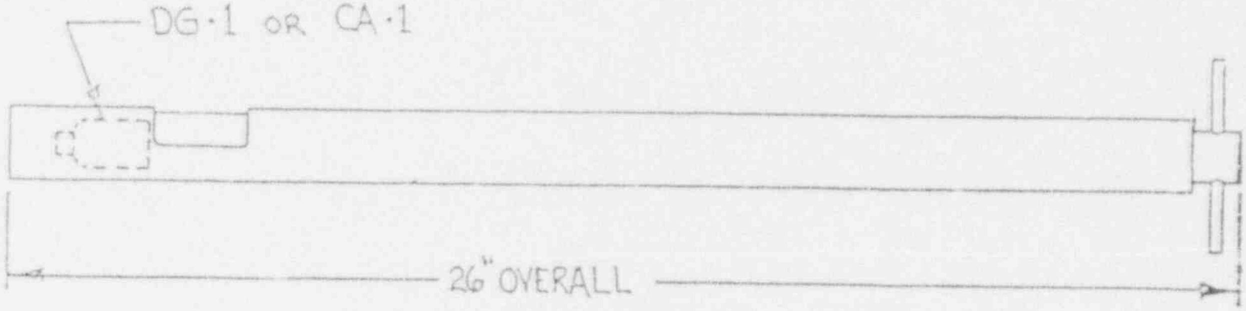


Transport Path To The Radchem Laboratory

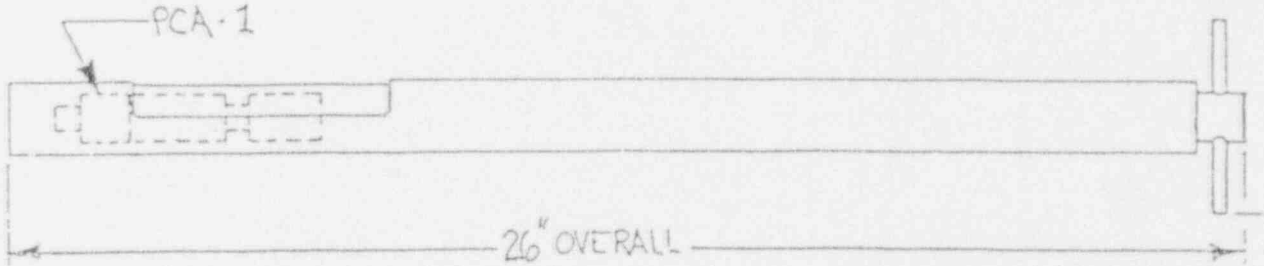
Sketch: Mobile Pig



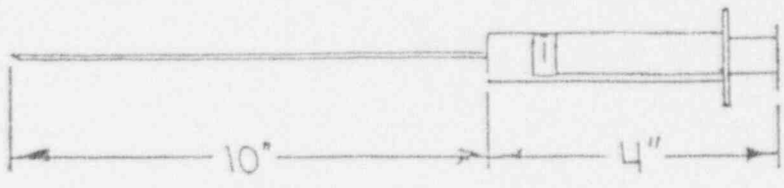
SKETCH: GAS SAMPLE TONGS & LIQUID ALIQUOTER



TONGS T-1, T-2



TONG T-3

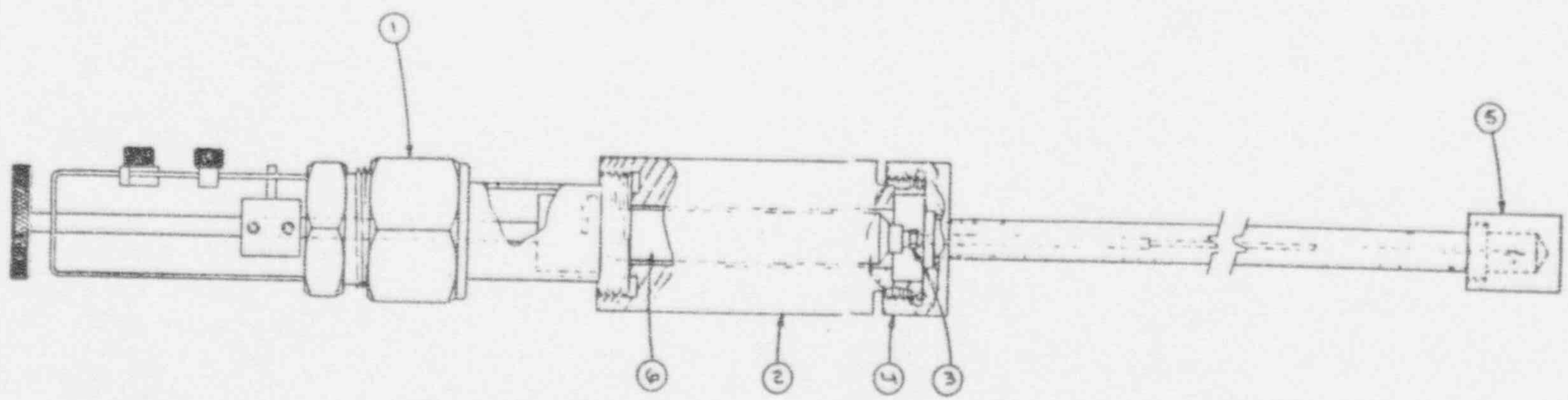


ALIQUOTER DL-1

Figure 5

5-550-59-061

P/N	ITEM	REQD	DESCRIPTION
6-00373A	1	1	MESITON ASSEMBLY, DWG.# 190-67-128-3
6-00371A	2	1	SYRINGE SHIELD, DWG.# 190-67-082
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



ADDED P/N 3

P/N 7-00163A

DO NOT SCALE DWG

ENG REVIEW

SENTRY EQUIPMENT CORP.

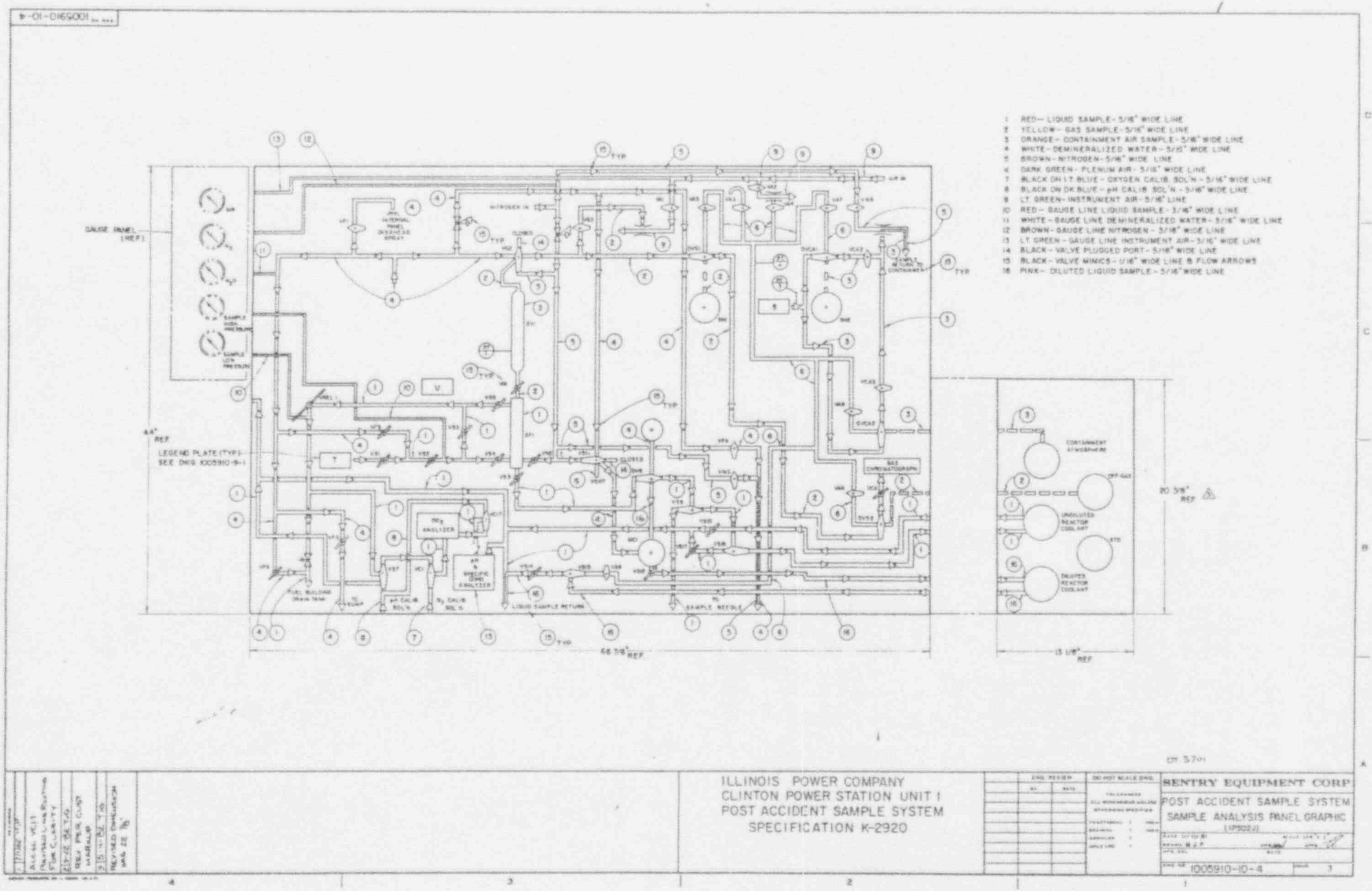
DILUTED LIQUID SAMPLE NEEDLE ASSEMBLY

SCALE FULL

190-63-035-3

1

Figure 6


1005910-10-4
GAUGE PANEL (REF)
LEGEND PLATE (TYP) SEE DWG 1005910-9-1
44" REF
68 7/8" REF
15 1/8" REF
20 3/8" REF
NITROGEN IN
GAS CHROMATOGRAPH
LIQUID SAMPLE RETURN
SAMPLE NEEDLE
FUEL BUILDING DRAIN TANK
TO SUMP
pH CALIB SOL'N
O_2 CALIB SOL'N
DO_2 ANALYZER
CONTAINMENT ATMOSPHERE
OFF-GAS
UNDILUTED REACTOR COOLANT
DILUTED REACTOR COOLANT
1 RED— LIQUID SAMPLE - 5/16" WIDE LINE
2 YELLOW - GAS SAMPLE - 5/16" WIDE LINE
3 ORANGE - CONTAINMENT AIR SAMPLE - 5/16" WIDE LINE
4 WHITE - DEMINERALIZED WATER - 5/16" WIDE LINE
5 BROWN - NITROGEN - 5/16" WIDE LINE
6 DARK GREEN - PLENUM AIR - 5/16" WIDE LINE
7 BLACK ON LT BLUE - OXYGEN CALIB. SOL'N - 5/16" WIDE LINE
8 BLACK ON DK BLUE - pH CALIB SOL'N - 5/16" WIDE LINE
9 LT. GREEN - INSTRUMENT AIR - 5/16" LINE
10 RED - GAUGE LINE LIQUID SAMPLE - 3/16" WIDE LINE
11 WHITE - GAUGE LINE DEMINERALIZED WATER - 3/16" WIDE LINE
12 BROWN - GAUGE LINE NITROGEN - 3/16" WIDE LINE
13 LT GREEN - GAUGE LINE INSTRUMENT AIR - 3/16" WIDE LINE
14 BLACK - VALVE PLUGGED PORT - 5/16" WIDE LINE
15 BLACK - VALVE MIMICS - 1/16" WIDE LINE & FLOW ARROWS
16 PINK - DILUTED LIQUID SAMPLE - 5/16" WIDE LINE
ILLINOIS POWER COMPANY
CLINTON POWER STATION UNIT I
POST ACCIDENT SAMPLE SYSTEM
SPECIFICATION K-2920
SENTRY EQUIPMENT CORP
POST ACCIDENT SAMPLE SYSTEM
SAMPLE ANALYSIS PANEL GRAPHIC
1005910-10-4

SAFETY EVALUATION FORM

1.1 L&S Log # 93-0058

Document Evaluated:

1.2 Number: USAR 1.3 Revision: 4

1.4 Title: Removing Containment/Drywell Atmospheric Hydrogen/Oxygen Sampling Capability with the Post Accident Sampling System (PASS)

1.5 References:

U-600525 (5/5/86)	Tech Spec. 3.3.7.5
NUREG 0853 part 9.3.5	USAR App. D. II.B.3
NUREG 0737 II.B.3	USAR Table 7.1-13
NUREG-0578	USAR Table 9.3.7.E13)&A6)
NUREG-0660	USAR 9.3.7
Reg. Guide 1.97 Rev. 3	USAR Table 9.3-5
BWROG 8324 (6/17/83)	
NEDO-22215 (August 1982)	
EPIP EC 13 R/3	

BLOCK A - DESCRIPTION OF CHANGE
(Use additional pages if required)

A.1 Describe the basic document or system and the changes being made. Discuss how the change affects the SAR description. Discuss the reason for change.

The PASS Evaluation Report (U-600525/Y-206420, 5/5/86) and USAR detail CPS compliance with NUREG 0737, TMI Action Plan Item II.B.3. These documents currently describe CPS as having the capability to monitor Containment/Drywell atmospheric hydrogen and oxygen with the PASS. CPS is retracting from this capability.

USAR 9.3.7; App. D.II.B.3; Table 7.1-13; Table 9.3.7.E13; and Table 9.3-5 describe PASS as having this capability. These portions of the USAR will be revised for the following reasons:

CPS utilizes redundant, physically and electrically separated, 1E powered, Containment/Drywell atmospheric hydrogen/oxygen analyzing systems which require plant shutdown unless both independent systems are operable per Tech. Spec. 3.3.7.5. Reg. Guide 1.97 (R/3) states Containment/Drywell atmospheric hydrogen/oxygen can have multiple use and must meet the more stringent requirements of Category 1. CPS has exceeded Category 1 requirements with fully qualified, redundant, continuous real-time display monitoring systems, with onsite (standby) power. CPS has double the requirement of the most stringent control NUREG 0737/Reg. Guide 1.97 instruments to provide dependable redundant safety-related system back-up capability. As CPS has exceeded the requirements in this area, PASS sampling is unneeded.

Maintaining PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability is an unnecessary added burden on CPS. Removing the unneeded capability allows CPS to more effectively utilize resources for support of useful PASS capabilities. Chemistry personnel radiation exposures during accident mitigation will be lowered and less highly radioactive material will be routed outside primary and secondary Containment.

L&S Log # 1...

(Continued from page 1)

A.2 Identify the equipment, systems and parameters that may be affected by the change:

The Post Accident Sampling System, PASS is a non-safety, non-seismic system. The PASS panel 1PS02J will be unchanged by retracting from sampling the Containment/Drywell atmospheric hydrogen/oxygen. Eliminating this capability will simplify procedures, laboratory equipment, Chemistry Technician training, and Emergency Operating Procedures. No CPS operating parameter will be affected by this change. CPS Containment/Drywell atmospheric hydrogen/oxygen will continue to be monitored by Tech. Spec. 3.3.7.5 redundant 1E powered safety-related monitors which require plant shutdown if either system becomes inoperable.

L&S Log # 93-0258

BLOCK B - RADWASTE TREATMENT SYSTEMS

B.1 The proposed activity involves a modification to a radiological waste treatment system or the way in which it is operated as described in Chapter 11 of the SAR. Yes ____ No X

B.2 Because: The PASS is not a radiological waste treatment system. Removing the PASS Containment/Drywell atmospheric hydrogen/ oxygen sampling capability will have no affect on PASS or any other plant system. Therefore, this activity will not modify any radiological waste treatment system or the way in which it is operated.

If B.1 is yes, complete form NF-003.

BLOCK C - TECHNICAL SPECIFICATION IMPACT

C.1 The proposed activity requires a change to any part of the Technical Specifications. Yes ____ No X

C.2 Justification if "NO", Technical Specification change package identification number if "YES".

The PASS is controlled per Technical Specification 6.8.4.a/c which requires leakage inspection/testing, maintenance, training, and procedures. However, Tech. Specs do not require or imply that PASS be capable of obtaining Containment/Drywell atmospheric hydrogen/oxygen. Therefore, CPS Technical Specifications will remain unchanged.

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

(Attach additional pages with the responses to the block D questions. Identify your answers to Parts I, II, III, and IV.)

Part I - Impact on equipment malfunctions evaluated as the design basis

1. For the equipment and systems identified in A.1 and A.2, identify any failures evaluated in the SAR.

Failure of the PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability is not evaluated in the USAR. NUREG 0737 and Reg. Guide 1.97 Rev. 3 require backup grab sample capability for those PASS capabilities which use in-line instrumentation. CPS utilizes a seismic, safety-related system with a redundant, electrically and physically separated, seismic, safety-related system as back-up Containment/Drywell atmospheric hydrogen/oxygen sampling capability. While this is not a grab sample, this back-up capability far exceeds the Reg. Guide 1.97 requirements for a dependable backup system.

(Continued from page 2)

2. Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

The PASS will be unchanged. Laboratory equipment, support procedures, and personnel training will be simplified. Personnel radiation exposures will be lower during post-accident mitigation and recovery. Less highly radioactive material will be routed outside primary and secondary Containment. Containment/Drywell atmospheric hydrogen/oxygen will continue to be sampled by redundant, safety-related, seismically qualified, Tech. Spec. 3.3.7.5 monitors which require plant shutdown if either monitor is inoperable.

3. Identify what new failure modes could be introduced by the change.

No new failure mode could be introduced by retracting from the PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability.

(See D.I.2 above)

4. Identify any impact of the change on the consequences of the failures evaluated in the SAR.

Removing the PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability will not impact the consequences of the failures evaluated in the USAR. The USAR takes no credit for the PASS in mitigating the accidents evaluated. Additionally, this parameter will remain available as provided by redundant, safety-related, seismically qualified, Tech. Specs 3.3.7.5 monitors which require plant shutdown if either 1E powered system becomes inoperable.

5. Identify any impact of the change on the probabilities of the failures evaluated in the SAR.

The PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability has no impact on the probability of any failure evaluated in the SAR. (See D.I.4 above)

L&S Log # 93-0258

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 4, are the consequences of any malfunction of equipment evaluated in the SAR increased?	YES ___ NO _X_
Based on item 5, is the probability of a malfunction of equipment evaluated in the SAR increased?	YES ___ NO _X_

If the answer to any of the above questions is yes, the change is an unreviewed safety question.

Part II - Impact on the accidents evaluated as the design basis

1. Identify the accidents evaluated in the SAR which could be affected by the change.

   Eliminating the PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability will not affect any accident evaluated in the USAR. The purpose of this parameter will continue to be met by independent, seismic, safety-related, Tech. Spec. 3.3.7.5 monitors which require plant shutdown if either system is inoperable.

. Discuss how the change impacts the consequences of these accidents.

   Eliminating the PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability will not impact the consequences of the accidents evaluated in the USAR. The USAR takes no credit for the PASS in mitigating any evaluated accident.

3. Discuss how the change impacts the probability of these accidents.

   The PASS is a non-safety/non-seismic system. The PASS is not a precursor to and has no impact on the accidents evaluated as the design basis of the USAR. Retracting from the PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability will have no affect on the PASS or the probability of the accidents evaluated as the bases of the USAR.

SUMMARY

Based on item 2, are the consequences of an accident evaluated in the SAR increased?	YES ___ NO _X_
Based on item 3, is the probability of an accident evaluated in the SAR increased?	YES ___ NO _X_

If any of the above questions is answered yes, the change is an unreviewed safety question.

L&S Log #93-0058

Part III - Potential for Creation of a New Unanalyzed Event

1. Based on Part I, items 1 and 3, could this change initiate a new type of accident or equipment malfunction? Discuss the basis for this determination.

   Retracting from PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability will have no affect on the PASS or any other CPS system or parameter. Therefore, this action will not initiate a new type of accident or equipment malfunction.

2. Determine if the new accident or malfunction identified above has sufficient probability or consequences to be considered in the Licensing basis. Discuss the bases for this determination.

   Not Applicable. See D.III.1.

L&S Log #93-0058

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 2, does the change create the possibility for an equipment malfunction or accident of a different type than previously evaluated in the SAR?

YES ____
NO __X__

If the answer if yes, the change represents an unreviewed safety question.

Part IV - Impact on the Margin of Safety

1. Identify how any of the protective barriers are directly affected by the change.

Retracting from PASS Containment/Drywell atmospheric hydrogen/oxygen sampling capability has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the Containment. Containment/Drywell atmospheric hydrogen/oxygen sampling will continue to be accomplished by way of the CPS redundant 1E, safety-related, seismically qualified, Tech. Spec. 3.3.7.5 monitors which require plant shutdown when either system is inoperable. Therefore, removing this capability will have no impact on any of the protective barriers.

2. Discuss the impact of the change on the approach to the acceptance limits for any of the protective barriers.

As CPS will continue to monitor this parameter by use of better equipment, retracting from monitoring with the PASS will have no impact on the approach to the acceptance limits for any of the protective barriers.

3. Discuss the impact of the change on the bases of the Technical Specifications.

PASS Containment/Drywell atmospheric hydrogen/oxygen sampling is not stated or implied in the bases of the Technical Specifications. Technical Specifications do require the Tech. Spec. 3.3.7.5 monitors which will continue to monitor as required. Technical Specification bases will not be impacted by this change.

SUMMARY

Based on item 2, is any parameter in chapter 7 of the Safety Evaluation Manual exceeded?

YES ____
NO __X__

Based on items 2 and 3, does the change reduce the margin of safety provided for the protective barriers?

YES ____
NO __X__

If the first of these two questions is answered yes, the change may be unsafe and requires further justification. If the first question is answered no and the second is answered yes, the change is safe to implement but is an unreviewed safety question and requires prior NRC approval.

L&S Log # 93-0258

SAFETY EVALUATION FORM

BLOCK E - SUMMARY

Based on the evaluation in Block C and Block D, the change

_X_ is safe and is not an unreviewed safety question and requires no Technical Specification change. The change may be implemented in accordance with applicable procedures.

____ is safe but is an unreviewed safety question or requires a Technical Specification change. The change requires NRC approval, prior to implementation.

____ is unsafe and cannot be implemented.

	printed name	signature	date
Preparer	Tom Leffler	Tom Leffler	4/27/93
Director	D.E. Korneman	D.E. Korneman	4/27/93
Manager, NSED	N/A SSG		
Manager, L&S	SSG 4/29/93 J.L. Peterson	J.L. Peterson	5-3-93
FRG	P.D. [illegible]	[illegible]	5/6/93

EVIDENCE OF NRC APPROVAL, IF REQUIRED:

License Amendment No. N/A

printed name	signature	date
N/A	SSG 4/29/93	

The department responsible for vaulting the parent document must vault this completed form with the document evaluated.

NF-002-6 (12/90)
M7521/254/L

SAFETY EVALUATION FORM

1.1 L&S Log # 93-0059

Document Evaluated:

1.2 Number: USAR 1.3 Revision: 4

1.4 Title: Post Accident Sampling System (PASS) Containment and Drywell Atmospheric Iodine Sampling

1.5 References:

USAR App. D Item. II.B.3	NUREG 0853 part 9.3.5
USAR Table 9.3-5	NUREG 0737,II.B.3
USAR Table 7.1-13	NUREG-0578
USAR Table 9.3.7 E13	NUREG-0660
U-600525 (5/5/86)	Reg. Guide 1.97 Rev. 3
BWROG-8324 (6/17/83)	Tech Spec 3.3.7.5
NEDO-22215 (August 1982)	Tech Spec 6.8.4.a, c
	EPIP EC 13 R/3

BLOCK A - DESCRIPTION OF CHANGE
(Use additional pages if required)

A.1 Describe the basic document or system and the changes being made. Discuss how the change affects the SAR description. Discuss the reason for change.

The Post Accident Sampling System (PASS) Evaluation Report, U-600525, and CPS USAR identify the CPS compliance commitments in order to meet the recommendations of NUREG 0737, TMI Action Plan Item II.B.3. They currently describe CPS as having the capability to sample Iodine in Containment and Drywell air with the PASS. CPS is retracting from the commitment to this capability based on the following reasons:

The USAR will need to be changed to reflect this change in CPS commitments concerning the PASS. SAR section 9.3.7.2 GAS SAMPLING at PASS describes collecting the sample for isotopic analysis, but does not specify Iodine. USAR table 7.1-13 E13) states requirements for Gamma Spectrum of Containment air, but does not specify Iodine. USAR Appendix D II.B.3 is CPS position on radionuclide analysis. It has grouped analyses in such a way as to imply CPS will sample Containment and Drywell atmosphere for Iodine. So, USAR Appendix D will be more clearly defined.

Containment and Drywell atmospheric Iodine has never been required by the NRC at PASS unless it is used to estimate the extent of core damage. NUREG 0737 mentions iodine as an example isotope which may be used to estimate the extent of core damage. R/G 1.97 Rev. 3 states containment air should be sampled at PASS for Gamma Spectrum, but does not specify Iodine.

L&S Log # 93-0059

(Continued from page 1)

CPS uses BWR Owners Group (BWROG 8324, 6/17/83) recommended methods for core damage estimation found in General Electric's NEDO 22215 document of August, 1982. These methods are proceduralized in CPS Emergency Plan Implementing Procedure EC-13. CPS does not use Containment or Drywell atmospheric Iodine for estimation of core damage. CPS Emergency Procedure, EPIP EC-13, ratios PASS reactor water concentrations of iodine, cesium, barium, strontium, lawrencium & ruthenium and PASS Drywell/Containment air concentrations of xenon & krypton to estimate core damage. Additionally, EC-13 utilizes core uncovery time, Drywell/Containment radiation levels and Drywell/Containment air hydrogen concentration to estimate core damage.

Maintaining this capability is an unnecessary burden on CPS. Retracting this unneeded capability allows CPS to utilize resources more effectively toward capabilities which provide useful data. Also, personnel radiation exposures during accident mitigation will be lowered, high level radioactivity will be minimized outside of primary and secondary containment and the potential for release to the environment will be lowered.

A.2 Identify the equipment, systems and parameters that may be affected by the change:

Containment and Drywell atmospheric Iodine sampling requires sample lines to be heat traced. Heat tracing these lines prevents moisture accumulation in the lines. Such moisture would absorb Iodine and prevent CPS from obtaining a representative sample. Heat tracing shall be evaluated toward abandonment if the sole purpose is the post accident sampling of iodine. In any case its need to support representative Containment and Drywell sampling for airborne iodine is eliminated.

No CPS operating parameter will be affected by this change. CPS will maintain normal and accident range process monitors for continuous monitoring of effluent gaseous iodine per the CPS Offsite Dose Calculation Manual and Tech Spec 6.8.4.c including grab sample capability.

Eliminating Drywell and Containment atmospheric iodine sampling capability at the PASS will have no affect on the PASS other than to simplify PASS which facilitates maintenance required by Tech Spec 6.8.4.c.

L&S Log # 93-0259

BLOCK B - RADWASTE TREATMENT SYSTEMS

B.1 The proposed activity involves a modification to a radiological waste treatment system or the way in which it is operated as described in Chapter 11 of the SAR. Yes ____ No X

B.2 Because: The PASS is not a radiological waste treatment system. Removing the Containment and Drywell atmospheric Iodine sampling will only simplify the PASS. Therefore, no radiological waste treatment system or its operation will be involved.

If B.1 is yes, complete form NF-003.

BLOCK C - TECHNICAL SPECIFICATION IMPACT

C.1 The proposed activity requires a change to any part of the Technical Specifications. Yes ____ No X

C.2 Justification if "NO", Technical Specification change package identification number if "YES".

CPS Technical Specifications do not directly address PASS Drywell and Containment atmospheric Iodine sampling. Technical Specification 6.8.4.a requires the PASS be maintained and inspected to reduce leakage and 6.8.4.c requires maintenance of PASS sampling and analysis equipment. CPS will maintain normal and accident range process monitors for continuous monitoring of effluent gaseous iodine per the CPS Offsite Dose Calculation Manual and Tech Spec 6.8.4.c including grab sample capability. These Technical Specifications will not be changed by removing the PASS Containment and Drywell atmospheric Iodine sampling commitment.

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

(Attach additional pages with the responses to the block D questions. Identify your answers to Parts I, II, III, and IV.)

Part I - Impact on equipment malfunctions evaluated as the design basis

1. For the equipment and systems identified in A.1 and A.2, identify any failures evaluated in the SAR.

   The USAR does not identify any failure of the PASS Containment or Drywell atmospheric Iodine sampling capability. NUREG 0737 and Reg. Guide 1.97 rev. 3 require CPS to maintain back-up sampling for those PASS parameters monitored with in-line instruments. Containment and Drywell atmospheric Iodine sampling is not performed with in-line instrumentation. The need for grab sampling is being deleted.

2. Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

L&S Log # 93-0259

(Continued from page 2)

The PASS design parameters and hence the performance of the PASS will be unaffected by removing the Containment and Drywell atmospheric iodine capability. The PASS may be better maintained as the PASS will have fewer subsystems to divert maintenance resources from subsystems providing useful data.

3. Identify what new failure modes could be introduced by the change.

   No new failure mode could be introduced by eliminating Containment and Drywell atmospheric Iodine monitoring capability at the PASS. (See D.I.1, 2)

4. Identify any impact of the change on the consequences of the failures evaluated in the SAR. (See D.I.1,2)

   Eliminating the Containment and Drywell atmospheric Iodine sampling capability with the PASS has no impact on the consequences of the failures evaluated in the SAR.

5. Identify any impact of the change on the probabilities of the failures evaluated in the SAR.

   Eliminating the Containment and Drywell atmospheric Iodine monitoring capability at PASS will simplify the PASS. Simplifying the PASS will have no affect on any CPS operating parameter. Therefore, removing this capability will have no impact on the probability of the failures evaluated in the SAR.

L&S Log # 93-0059

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 4, are the consequences of any malfunction of equipment evaluated in the SAR increased? YES ____ NO X

Based on item 5, is the probability of a malfunction of equipment evaluated in the SAR increased? YES ____ NO X

If the answer to any of the above questions is yes, the change is an unreviewed safety question.

Part II - Impact on the accidents evaluated as the design basis

1. Identify the accidents evaluated in the SAR which could be affected by the change.

   No accident evaluated in the USAR will be affected by eliminating the Containment and Drywell atmospheric Iodine sampling capability at the PASS panel. This sampling capability has no relationship with any parameter which can affect any USAR evaluated accident. Chapter 15 of the USAR does not utilize PASS for mitigating any accident.

2. Discuss how the change impacts the consequences of these accidents.

   As this sampling capability has no relationship with any parameter which can affect USAR evaluated accidents, removing this sampling capability will not impact any consequence of any USAR evaluated accident.

3. Discuss how the change impacts the probability of these accidents.

   As this sampling capability has no relationship with any parameter which can affect SAR evaluated accidents, removing this sampling capability will not impact the probability of any accidents.

SUMMARY

Based on item 2, are the consequences of an accident evaluated in the SAR increased? YES ____ NO X

Based on item 3, is the probability of an accident evaluated in the SAR increased? YES ____ NO X

If any of the above questions is answered yes, the change is an unreviewed safety question.

L&S Log # 93-0259

(Continued from page 3)

Part III - Potential for Creation of a New Unanalyzed Event

1. Based on Part I, items 1 and 3, could this change initiate a new type of accident or equipment malfunction? Discuss the basis for this determination.

   No accident evaluated in the SAR will be affected by removing the Containment and Drywell atmospheric Iodine sampling capability at the PASS panel. This sampling capability has no relationship with any parameter which can affect any USAR evaluated accident.

2. Determine if the new accident or malfunction identified above has sufficient probability or consequences to be considered in the Licensing basis. Discuss the bases for this determination.

   As this sampling capability has no relationship with any parameter which can affect SAR evaluated accidents, removing this sampling capability will not impact any consequence of any USAR evaluated accident.

L&S Log # 93-0059

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 2, does the change create the possibility for an equipment malfunction or accident of a different type than previously evaluated in the SAR?	YES ____ NO X

If the answer if yes, the change represents an unreviewed safety question.

Part IV - Impact on the Margin of Safety

1. Identify how any of the protective barriers are directly affected by the change.

   Removing the CPS commitment to maintain Containment and Drywell atmospheric Iodine monitoring capability with the PASS will have no affect on the fuel cladding, the reactor coolant pressure boundary, or the Containment. CPS EPIP EC-13 does not use this parameter has no effect on any of these boundaries. This parameter, as measured by the PASS after a postulated accident in the detection of fuel cladding or Containment breach.

   This parameter can be used to detect reactor coolant pressure boundary leakage. Technical Specification 3/4.4.3.1. requires redundant systems to detect this parameter for the purpose of leak detection. The PASS capability is not needed for this purpose. Sump levels and Cooler drain flows backup this parameter for leak detection. Technical Specifications will continue to require plant shutdown when leak detection capability is lost.

   In summary, removal of PASS monitoring capability for atmospheric Iodine in Containment and Drywell will have no effect on any protective barrier.

2. Discuss the impact of the change on the approach to the acceptance limits for any of the protective barriers.

   Retracting from the capability to measure this parameter with the PASS has no impact on the approach to the acceptance limits for any of the protective barriers. CPS EPIP EC-13 utilities alternate parameters for estimates of protective barrier degradation.

3. Discuss the impact of the change on the bases of the Technical Specifications.

   The bases of the Technical Specifications do not address Containment or Drywell atmospheric Iodine at PASS. Removing the CPS capability to obtain this parameter with PASS will have no impact on the bases of Technical Specifications.

L&S Log # 93-0259

(Continued from page 4)

SUMMARY

Based on item 2, is any parameter in chapter 7 of the Safety Evaluation Manual exceeded? YES ____ NO __X__

Based on items 2 and 3, does the change reduce the margin of safety provided for the protective barriers? YES ____ NO __X__

If the first of these two questions is answered yes, the change may be unsafe and requires further justification. If the first question is answered no and the second is answered yes, the change is safe to implement but is an unreviewed safety question and requires prior NRC approval.

L&S Log # 93-0059

SAFETY EVALUATION FORM

BLOCK E - SUMMARY

Based on the evaluation in Block C and Block D, the change

_X_ is safe and is not an unreviewed safety question and requires no Technical Specification change. The change may be implemented in accordance with applicable procedures.

____ is safe but is an unreviewed safety question or requires a Technical Specification change. The change requires NRC approval, prior to implementation.

____ is unsafe and cannot be implemented.

	printed name	signature	date
Preparer	T. I. Leffler	Tom Leffler	4/27/93
Director	D.E. Korneman	DE Korneman	4/27/93
Manager, NSED	NA		
Manager, L&S	SSG 4/29/93 J.L. Peterson	J L Peterson	5-3-93
FRG	P.A. Yocum	[illegible]	5/6/93

EVIDENCE OF NRC APPROVAL, IF REQUIRED:

License Amendment No. N/A

printed name	signature	date
N/A	SSG 4/29/93	

The department responsible for vaulting the parent document must vault this completed form with the document evaluated.

NF-002-5(12/90)
M6420/316

SAFETY EVALUATION FORM

1.1 L&S Log # 93-0060

Document Evaluated:

1.2 Number: USAR 1.3 Revision: 4

1.4 Title: Removing Post Accident Sampling System (PASS) Reactor Coolant Dissolved Noble Gas Monitoring Capability.

1.5 References:

USAR Table 9.3-5	BWROG-8324
USAR Appendix D Item II.B.3	NEDO-22215
USAR Section 9.3.7.2	EPIP EC-13 (R/3)
NUREG 0737,II.B.3	U600525 (5/5/86)
NUREG 0853 Part 9.3.5	Tech Spec, 3.3.7.5
NUREG-0578	Tech Spec 6.8.4 a/c
NUREG-0660	
Reg. Guide 1.97 Rev. 3	

BLOCK A - DESCRIPTION OF CHANGE
(Use additional pages if required)

A.1 Describe the basic document or system and the changes being made. Discuss how the change affects the SAR description. Discuss the reason for change.

The PASS Evaluation Report (U600525/Y206420) and USAR detail CPS compliance with NUREG 0737, TMI Action Plan, Item II.B.3. They describe that CPS has the capability to monitor reactor coolant dissolved noble gas at the PASS panel. CPS is retracting from this capability.

USAR Appendix D item II.B.3 and Section 9.3.7.2 describe PASS as determining reactor coolant dissolved noble gas. These portions of the USAR will be revised to reflect CPS' position to not sample for reactor coolant dissolved noble gas with the PASS for the following reasons:

Reactor coolant dissolved noble gas is not required by NUREG 0737 or Reg. Guide 1.97. The NRC only requires dissolved noble gas measurements if they are used to estimate core damage. Since CPS uses BWR Owners Group recommended methods for core damage estimation which are NRC approved, and these methods do not use reactor coolant dissolved noble gas, there is no need for monitoring reactor coolant dissolved noble gas with the PASS.

Maintaining reactor coolant dissolved noble gas sampling capability with the PASS unnecessarily burdens CPS. Removing the unneeded sampling capability allows CPS to more effectively utilize resources for support of useful PASS capabilities. Personnel radiation exposures during accident mitigation will be lowered and less highly radioactive material will be outside primary and secondary containment.

L&S Log# 93-0060

(Continued from page 1)

A.2 Identify the equipment, systems and parameters that may be affected by the change:

The Post Accident Sampling System is a non-safety, non-seismic system. The Post Accident Sampling System Panel, 1PS02J, stripped gas section will no longer be required for post accident sampling of reactor coolant dissolved noble gas The elimination of this commitment would simplify the PASS. Simplifying the PASS will allow CPS to more effectively maintain needed analysis capabilities.

No CPS operating parameter will be affected by this change. CPS reactor coolant dissolved noble gas boils free of the reactor coolant. Less than 5% of the noble gas inventory is expected in BWR reactor coolant. Noble gas in reactor coolant is not a normally monitored BWR parameter. CPS does not attempt to control reactor coolant dissolved noble gas. Deleting the CPS capability to monitor reactor coolant dissolved noble gas will have no affect on CPS chemistry control or any CPS operating parameter.

L&S Log # 93-0060

BLOCK B - RADWASTE TREATMENT SYSTEMS

B.1 The proposed activity involves a modification to a radiological waste treatment system or the way in which it is operated as described in Chapter 11 of the SAR. Yes ____ No X

B.2 Because: The PASS is not a radiological waste treatment system. Removing the PASS reactor coolant dissolved noble gas sampling capability will simplify the PASS. Simplifying the PASS will not modify any radiological waste treatment system or the way it is operated.

If B.1 is yes, complete form NF-003.

BLOCK C - TECHNICAL SPECIFICATION IMPACT

C.1 The proposed activity requires a change to any part of the Technical Specifications. Yes ____ No X

C.2 Justification if "NO", Technical Specification change package identification number if "YES".

PASS reactor coolant dissolved noble gas is not a Technical Specification requirement. It is not mentioned in Technical Specification Bases. Technical Specification 6.8.4.a/c require PASS leakage inspection/testing, maintenance, training, and procedures, but does not require PASS reactor coolant dissolved noble gas sampling. Retracting from the CPS monitoring capability of PASS reactor coolant dissolved noble gas will not require a change to any part of Technical Specifications.

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

(Attach additional pages with the responses to the block D questions. Identify your answers to Parts I, II, III, and IV.)

Part I - Impact on equipment malfunctions evaluated as the design basis

1. For the equipment and systems identified in A.1 and A.2, identify any failures evaluated in the SAR.

   Failure of the PASS reactor coolant dissolved noble gas sampling capability is not evaluated in the SAR. NUREG 0737 and Reg. Guide 1.97 Rev. 3 require backup grab sample capability for those PASS capabilities which use in-line monitors. CPS does not use an in-line monitor for PASS determination of reactor coolant dissolved noble gas. The backup capability does not apply to CPS.

L&S Log # 93-0260

(Continued from page 2)

2. Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

   The PASS reactor coolant dissolved noble gas capability would be eliminated. The PASS design parameters will be unaffected by this. The PASS may be better maintained as the system will have one less subsystem diverting resources from subsystems providing useful data. The PASS will continue to meet the NUREG 0737 and Reg. Guide 1.97 intent of providing useful data to Operations and Emergency Response personnel for estimating/mitigating core damage and long-term corrosion potential.

   NUREG 0737 and Reg. Guide 1.97 present reactor coolant noble gas as an example for estimating core damage. CPS does not estimate core damage with reactor coolant dissolved noble gas. CPS estimates core damage with NRC approved, BWR Owners Group recommended GE guidelines. CPS emergency Procedure, EPIP EC-13, ratios PASS reactor water iodine, cesium, barium, strontium, lawrencium & ruthenium concentrations and Drywell/Containment air xenon and krypton concentrations to estimate core damage. Additionally, EC-13 utilizes core uncovery time, Drywell/Containment radiation levels and Drywell/Containment air hydrogen concentration to estimate core damage. PASS reactor coolant dissolved noble gas analysis is not needed and sampling for this parameter is not required.

3. Identify what new failure modes could be introduced by the change.

   No new failure mode could be introduced by retracting from PASS reactor coolant dissolved noble gas sampling capability, because retracting this equipment has no effect on the operation of PASS.

4. Identify any impact of the change on the consequences of the failures evaluated in the SAR.

   Removing the PASS reactor coolant dissolved noble gas sampling capability will not impact the consequences of the failures evaluated in the SAR. Dissolved noble gas is not useful to Chemistry personnel in determining post-accident corrosion potential. As dissolved noble gas is not useful in meeting the intent of the PASS, deleting it will have no affect on the accidents evaluated in the USAR.

5. Identify any impact of the change on the probabilities of the failures evaluated in the SAR.

   The PASS reactor coolant dissolved noble gas sampling capability has no impact on the probability of any failure evaluated in the USAR. (See D.I.3. above)

L&S Log# 93-0060

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 4, are the consequences of any malfunction of equipment evaluated in the SAR increased?	YES ____ NO _X_
Based on item 5, is the probability of a malfunction of equipment evaluated in the SAR increased?	YES ____ NO _X_

If the answer to any of the above questions is yes, the change is an unreviewed safety question.

Part II - Impact on the accidents evaluated as the design basis

1. Identify the accidents evaluated in the SAR whic: could be affected by the change.

   Retracting from the PASS reactor coolant dissolved noble gas sampling capability will not affect any accident evaluated in the USAR. The purpose of the PASS is to estimate/mitigate core damage and determine long-term post-accident corrosion potential. CPS Emergency Procedure, EPIP EC-13, ratios PASS reactor water concentrations of iodine, cesium, barium, strontium, lawrencium, & ruthenium and PASS Drywell/Containment air concentrations of xenon & krypton to estimate core damage. Additionally, EC-13 utilizes core uncovery time, Drywell/Containment radiation levels and Drywell/Containment air hydrogen concentration to estimate core damage. PASS reactor coolant dissolved noble gas sampling is not useful in meeting any of these PASS design purposes. As the PASS will continue to meet its design intent, removing this sampling capability will not affect any accident in the USAR. USAR chapter 15 makes no use of the PASS in mitigating USAR evaluated accidents.

2. Discuss how the change impacts the consequences of these accidents.

   PASS reactor coolant dissolved noble gas sampling is not useful in meeting PASS design intents delineated in NUREG 0737. CPS Emergency Procedure EPIP EC-13, ratios PASS reactor water concentrations to iodine cesium, barium, strontium, lawrencium, & ruthenium and PASS Drywell/Containment air concentrations of xenon & krypton to estimate core damage. Additionally, EC-13 utilizes core uncovery time, Drywell/Containment radiation levels and Drywell/Containment air hydrogen concentration to estimate core damage. Deleting this sampling capability will not impact the PASS. It will not impact the consequences of any accident evaluated as the design basis of the USAR.

(Continued from page 3)

3. Discuss how the change impacts the probability of these accidents.

   The PASS is a non-safety/non-seismic system. The PASS is not a precursor to and has no impact on the accidents evaluated as the design bases of the USAR. Removing the PASS reactor coolant dissolved noble gas sampling capability will simplify the PASS. Simplifying the PASS will have no impact on the probability of the accidents evaluated as the basis of the USAR.

SUMMARY

Based on item 2, are the consequences of an accident evaluated in the SAR increased?	YES ____ NO __X__
Based on item 3, is the probability of an accident evaluated in the SAR increased?	YES ____ NO __X__

If any of the above questions is answered yes, the change is an unreviewed safety question.

Part III - Potential for Creation of a New Unanalyzed Event

1. Based on Part I, items 1 and 3, could this change initiate a new type of accident or equipment malfunction? Discuss the basis for this determination.

   No. Removing the PASS reactor coolant dissolved noble gas sampling capability will simplify the PASS. CPS Emergency Procedure, EPIP EC-13, ratios PASS reactor water concentrations of iodine cesium, barium, strontium, lawrencium, & ruthenium and PASS Drywell/Containment air concentrations of xenon & krypton to estimate core damage. Additionally, EC-13 utilizes core uncovery time, Drywell/Containment radiation levels and Drywell/Containment air hydrogen concentration to estimate core damage. Simplifying the PASS will have no impact on the potential for initiating accidents and equipment malfunctions.

2. Determine if the new accident or malfunction identified above has sufficient probability or consequences to be considered in the Licensing basis. Discuss the bases for this determination.

   As no new accident or malfunction is identified above, the question of sufficient probability or consequences to be considered in the Licensing basis is not applicable.

L&S Log # 93-C260

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 2, does the change create the possibility for an equipment malfunction or accident of a different type than previously evaluated in the SAR?

YES ____
NO X

If the answer if yes, the change represents an unreviewed safety question.

Part IV - Impact on the Margin of Safety

1. Identify how any of the protective barriers are directly affected by the change.

   Removing the PASS reactor coolant dissolved noble gas sampling capability has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the Containment. CPS Emergency Procedure, EPIP EC-13, ratios PASS reactor water concentrations of iodine, cesium, barium, strontium, lawrencium, & ruthenium and PASS Drywell/Containment air concentrations of xenon & krypton to estimate core damage. Additionally, EC-13 utilizes core uncovery time, Drywell/Containment radiation levels and Drywell/Containment air hydrogen concentration to estimate core damage. Therefore, removing the capability has no direct or indirect affect on any of the protective barriers.

2. Discuss the impact of the change on the approach to the acceptance limits for any of the protective barriers.

   Removing the PASS reactor coolant dissolved noble gas sampling capability has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the Containment. Therefore, removing this capability will have no impact on the approach to the acceptance limits for any of the protective barriers.

3. Discuss the impact of the change on the bases of the Technical Specifications.

   The PASS reactor coolant dissolved noble gas sampling capability is not stated or implied in the Bases of the Technical Specifications. Removing this capability will have no impact on the Bases of the Technical Specifications.

SUMMARY

Based on item 2, is any parameter in chapter 7 of the Safety Evaluation Manual exceeded?

YES ____
NO X

Based on items 2 and 3, does the change reduce the margin of safety provided for the protective barriers?

YES ____
NO X

L&S Log #93-0260

(Continued from page 4)

If the first of these two questions is answered yes, the change may be unsafe and requires further justification. If the first question is answered no and the second is answered yes, the change is safe to implement but is an unreviewed safety question and requires prior NRC approval.

L&S Log #93-0260

SAFETY EVALUATION FORM

BLOCK E - SUMMARY

Based on the evaluation in Block C and Block D, the change

_X_ is safe and is not an unreviewed safety question and requires no Technical Specification change. The change may be implemented in accordance with applicable procedures.

____ is safe but is an unreviewed safety question or requires a Technical Specification change. The change requires NRC approval, prior to implementation.

____ is unsafe and cannot be implemented.

	printed name	signature	date
Preparer	Tom Leffler	Tom Leffler	4/27/93
Director	D E Korneman	D E Korneman	4/27/93
Manager, NSED	N/A		
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FRG	[illegible]	[illegible]	5/4/93

EVIDENCE OF NRC APPROVAL, IF REQUIRED:

License Amendment No. N/A

printed name	signature	date
N/A	SSG 4/29/93	

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NF-002-5(2/90)
M6552/319

SAFETY EVALUATION FORM

1.1 L&S Log # 93-0061

Document Evaluated:

1.2 Number: USAR 1.3 Revision: 4

1.4 Title: Removing Post Accident Sampling System (PASS) Reactor Coolant Dissolved Oxygen Monitoring Capability

1.5 References:

USAR App D II.B.3	NUREG-0578
USAR Section 9.3.7.2	NUREG-0660
USAR Section 9.3.7.4	NUREG-0853 part 9.3.5
USAR Section 9.3.7.6	NUREG 0737, II.B.3
USAR Table 9.3-5	Reg. Guide 1.97 Rev. 3
USAR 7.1-13 item E13)F)	GL 88-01 Response
	Tech. Spec. 6.8.4 a/c
	U-600525 (5/5/86)

BLOCK A - DESCRIPTION OF CHANGE
(Use additional pages if required)

A.1 Describe the basic document or system and the changes being made. Discuss how the change affects the SAR description. Discuss the reason for change.

The PASS Evaluation Report (U-600525, 5/5/96) and USAR outline CPS' commitments to meet NUREG 0737, TMI Action Plan Item II.B.3. These documents currently commit CPS to maintain reactor coolant dissolved oxygen monitoring capability at the PASS panel. The following USAR sections describe the PASS as having reactor coolant dissolved oxygen sampling capability:

1. Appendix D item II.B.3.
2. Section 9.3.7.2
3. Section 9.3.7.4
4. Section 9.3.7.6
5. Table 7.1-13 item E 13)F)
6. Table 9.3-5

CPS is retracting from this PASS reactor coolant dissolved oxygen capability. These sections will be amended to delete this requirement.

Reactor Coolant dissolved oxygen is not useful at CPS for post accident situations. Reactor coolant dissolved oxygen is not mandatory per NUREG 0737. Reactor coolant dissolved oxygen is not used to estimate nor mitigate core damage. Reactor coolant dissolved oxygen is presented as a corrosion potential evaluation tool.

L&S Log #93-0061

(Continued from page 1)

NUREG 0737 states that verifying reactor coolant dissolved oxygen below 0.1 ppm reduces concerns that the reactor coolant pressure boundary may be under Intergranular Stress Cracking Corrosion attack if reactor coolant chlorides exceed 0.15 ppm. Although, CPS exceeds the NUREG 0737 requirements for chloride analysis, BWR dissolved oxygen is never expected to be below 0.1 ppm. CPS operates under these normal BWR water chemistry conditions and does not inject hydrogen to mitigate Intergranular Stress Corrosion Cracking. Consequently, CPS reactor coolant dissolved oxygen serves no purpose and will always be greater than 0.1 ppm. Mitigating steps to counter normal BWR chemistry are outlined in CPS responses to GL 88-01 which details increased surveillance, ISI, stress relief plans, and identification of susceptible materials in CPS construction.

Therefore, taking a reactor coolant dissolved oxygen sample is a waste of resources which will be most valuable during a post accident condition. Chemistry personnel will receive lower radiation exposures during accident mitigation. Less highly radioactive material will be routed outside primary and secondary containment during accident mitigation. To ensure optimum support for those parameters which provide useful data, CPS is retracting from reactor coolant dissolved oxygen analysis capability at PASS.

A.2 Identify the equipment, systems and parameters that may be affected by the change:

The Post Accident Sampling System (PASS) will be simplified as the dissolved oxygen analyzer will not be required for post accident analysis. Simplifying the PASS will allow CPS to more effectively maintain analysis capabilities which provide useful data for post-accident core damage estimation/mitigation and corrosion control.

No CPS operating parameter will be affected by this change. CPS reactor coolant dissolved oxygen concentration is a dynamic function of: the radiolytic decomposition and reformation of water; degasification of oxygen into the steam; oxygen additions to condensate for corrosion control; minor condenser oxygen in-leakage; and depletion due to general and accelerated corrosions. Reactor coolant dissolved oxygen varies from 8 ppm when air-saturated cold shut-down conditions exist to 0.1 ppm when heated to $\geq$ 389°F and back to about 0.25 ppm with increase to 100% reactor power.

The purpose of PASS reactor coolant dissolved oxygen sampling is as a possible method of verifying that only a very low potential exists for Intergranular Stress Corrosion Cracking due to dissolved oxygen being below 0.1 ppm. As discussed above, CPS does not expect to ever be below 0.1 ppm. The reactor coolant dissolved oxygen concentration is not a parameter used to determine or mitigate core damage. Therefore, since this parameter serves no function, the only impact on equipment, systems and parameters will be a simplification of these systems and equipment.

L&S Log # 93-0261

BLOCK B - RADWASTE TREATMENT SYSTEMS

B.1 The proposed activity involves a modification to a radiological waste treatment system or the way in which it is operated as described in Chapter 11 of the SAR. Yes ____ No X

B.2 Because: The PASS dissolved oxygen analyzer and the PASS system are not a part of, any radiological waste treatment system. Removing the CPS commitment to maintain reactor coolant dissolved oxygen sampling capability will have no affect on any radiological waste treatment system.

If B.1 is yes, complete form NF-003.

BLOCK C - TECHNICAL SPECIFICATION IMPACT

C.1 The proposed activity requires a change to any part of the Technical Specifications. Yes ____ No X

C.2 Justification if "NO", Technical Specification change package identification number if "YES".

PASS reactor coolant dissolved oxygen sampling capability is not a Technical Specification requirement. It is not mentioned in Technical Specification Bases. Tech Spec 6.8.4 a/c require PASS leakage inspection/testing, maintenance, training, and procedures, but does not require PASS reactor coolant dissolved oxygen sampling. Removing this CPS commitment will not require a change to any part of Technical Specifications.

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

(Attach additional pages with the responses to the block D questions. Identify your answers to Parts I, II, III, and IV.)

Part I - Impact on equipment malfunctions evaluated as the design basis

1. For the equipment and systems identified in A.1 and A.2, identify any failures evaluated in the SAR.

   Failure of the PASS reactor coolant dissolved oxygen sampling capability is not evaluated in the USAR. NUREG 0737 and Reg. Guide 1.97 rev. 3 require backup grab sample capability for those PASS capabilities which use in-line monitors. The requirement for backup sampling is removed when obtaining the parameter is not required.

2. Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

   Maintaining the PASS reactor coolant dissolved oxygen sampling capability would no longer be required. The PASS design parameters will be unaffected by this. the PASS may be better maintained as the system will have one less subsystem diverting resources from those subsystems providing useful data.

L&S Log # 93-0261

(Continued from page 2)

3. Identify what new failure modes could be introduced by the change.

No new failure mode could be introduced by deleting the PASS reactor coolant dissolved oxygen sampling capability. This equipment is independent of the other PASS analysis equipment and has no impact on other PASS equipment operation. Removing it will not create any new failure mode, nor will it change failure modes for remaining PASS equipment.

4. Identify any impact of the change on the consequences of the failures evaluated in the SAR.

Removing the reactor coolant dissolved oxygen sampling capability at PASS will not impact the consequences of the failures evaluated in the USAR. Dissolved oxygen is not useful to operators in mitigating accidents. Dissolved oxygen is not useful to the Emergency Response Organization in estimating core damage. Unless hydrogen water chemistry is implemented, dissolved oxygen will always exceed the levels of Intergranular Stress Corrosion Cracking concern which are addressed in CPS response to Generic Letter 88-01. Dissolved oxygen is not mandatory per NUREG 0737. Removing this capability will not prevent the PASS from providing the support needed during post accident conditions. As the PASS can meet its design intent without reactor coolant dissolved oxygen, deleting it will not impact the PASS. As the PASS will continue to meet its intent, deleting the commitment to PASS reactor coolant dissolved oxygen capability will have no impact on the consequences of the failures evaluated in the USAR.

5. Identify any impact of the change on the probabilities of the failures evaluated in the SAR.

The PASS reactor coolant dissolved oxygen sampling capability has no impact on the probability of any failure evaluated in the USAR. (See D.I.4)

L&S Log # 93-0261

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 4, are the consequences of any malfunction of equipment evaluated in the SAR increased? YES ___ NO _X_

Based on item 5, is the probability of a malfunction of equipment evaluated in the SAR increased? YES ___ NO _X_

If the answer to any of the above questions is yes, the change is an unreviewed safety question.

Part II - Impact on the accidents evaluated as the design basis

1. Identify the accidents evaluated in the SAR which could be affected by the change.

   Removing the PASS reactor coolant dissolved oxygen sampling capability will not affect any accident evaluated as the design basis of the USAR. The PASS is not taken credit for mitigating any of the USAR Chapter 15 accidents.

2. Discuss how the change impacts the consequences of these accidents.

   No accident evaluated as the design basis of the USAR is affected by removing the PASS reactor coolant dissolved oxygen sampling capability. Therefore, removing this capability will have no impact on the consequences of these accidents.

   Furthermore, reactor coolant dissolved oxygen is not used to estimate or mitigate core damage nor will it be used to determine long-term post-accident corrosion potential. As estimating/mitigating core damage and determining long-term corrosion potential are the intent of the PASS, the PASS support of accidents evaluated as the design basis of the USAR will not be impacted by removing the PASS reactor coolant dissolved oxygen sampling capability. As removing the commitment does not impact PASS support of SAR evaluated accidents, removing it will not impact the consequences of these accidents.

3. Discuss how the change impacts the probability of these accidents.

   The PASS is non-safety/non-seismic. The PASS is not a precursor to and has no impact on the accidents evaluated as the design basis of the USAR. Therefore, removing the PASS reactor coolant dissolved oxygen sampling capability will have no impact on the probability of accidents evaluated in the SAR.

SUMMARY

Based on item 2, are the consequences of an accident evaluated in the SAR increased? YES ___ NO _X_

Based on item 3, is the probability of an accident evaluated in the SAR increased? YES ___ NO _X_

L&S Log # 93-0261

(Continued from page 3)

If any of the above questions is answered yes, the change is an unreviewed safety question.

Part III - Potential for Creation of a New Unanalyzed Event

1. Based on Part I, items 1 and 3, could this change initiate a new type of accident or equipment malfunction? Discuss the basis for this determination.

   No. Removing the PASS reactor coolant dissolved oxygen sampling capability will not change the way the PASS operates and will facilitate maintenance of the PASS. A higher state of readiness for the PASS will not impact the potential for initiating accidents and equipment malfunctions.

2. Determine if the new accident or malfunction identified above has sufficient probability or consequences to be considered in the Licensing basis. Discuss the bases for this determination.

   As no new malfunction is identified above, the question of sufficient probability or consequences to be considered in the Licensing basis is not applicable.

L&S Log # 93-0061

SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 2, does the change create the possibility for an equipment malfunction or accident of a different type than previously evaluated in the SAR?

YES _____
NO __X__

If the answer if yes, the change represents an unreviewed safety question.

Part IV - Impact on the Margin of Safety

1. Identify how any of the protective barriers are directly affected by the change.

   Removing the PASS reactor coolant dissolved oxygen sampling capability commitment has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the Containment. Therefore, removing the capability has no direct affect on any of the protective barriers.

2. Discuss the impact of the change on the approach to the acceptance limits for any of the protective barriers.

   Removing the PASS reactor coolant dissolved oxygen sampling capability has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the Containment. CPS response to GL 88-01 addresses Intergranular Stress Corrosion Cracking concerns. Therefore, removing this capability will have no impact on the approach to the acceptance limits for any of the protective barriers.

3. Discuss the impact of the change on the bases of the Technical Specifications.

   The PASS reactor coolant dissolved oxygen sampling capability is not stated or implied in the bases of the Technical Specifications. Removing this capability will have no impact on the bases of the Technical Specifications.

SUMMARY

Based on item 2, is any parameter in chapter 7 of the Safety Evaluation Manual exceeded?

YES _____
NO __X__

Based on items 2 and 3, does the change reduce the margin of safety provided for the protective barriers?

YES _____
NO __X__

If the first of these two questions is answered yes, the change may be unsafe and requires further justification. If the first question is answered no and the second is answered yes, the change is safe to implement but is an unreviewed safety question and requires prior NRC approval.

L&S Log # 93-0261

SAFETY EVALUATION FORM

BLOCK E - SUMMARY

Based on the evaluation in Block C and Block D, the change

_X_ is safe and is not an unreviewed safety question and requires no Technical Specification change. The change may be implemented in accordance with applicable procedures.

____ is safe but is an unreviewed safety question or requires a Technical Specification change. The change requires NRC approval, prior to implementation.

____ is unsafe and cannot be implemented.

	printed name	signature	date
Preparer	Tom Leffler	Tom Leffler	4/27/93
Director	D.E. Korneman	D.E. Korneman	4/27/93
Manager, NSED	N/A		
Manager, L&S	SSG 4/29/93 J.L. Peterson	J.L. Peterson	5-3-93
FRG			5/6/93

EVIDENCE OF NRC APPROVAL, IF REQUIRED:

License Amendment No. N/A

printed name	signature	date
N/A	SSG 4/29/93	

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NF-002-5(12/90)
M6421/316/L

SAFETY EVALUATION FORM

1.1 L&S Log # 93-0062

Document Evaluated:

1.2 Number: USAR 1.3 Revision: 4

1.4 Title: Post Accident Sampling System (PASS)
Reactor Coolant Dissolved Hydrogen Sampling

1.5 References:

USAR Appendix D II.B.3	NUREG 0853 5 part 9.3.5
USAR Section 9.3.7.2	NUREG 0737, TMI. Action Plan II.B.3
USAR Table 7.1-13	NUREG-0578
USAR Table 9.3-5	NUREG-0660
CPS PASS Compliance Report (11/8/85)	GL 88.01 Response
L&S ROC Y-206420	Tech. Spec. 3.3.7.5
CPS PASS Evaluation Report (5/5/86)	Tech. Spec. 6.8.4.a, c
	Reg Guide 1.97 Rev. 3.

BLOCK A - DESCRIPTION OF CHANGE
(Use additional pages if required)

A.1 Describe the basic document or system and the changes being made. Discuss how the change affects the SAR description. Discuss the reason for change.

The PASS Evaluation Report (U-600525, Y-206420) and USAR detail CPS compliance with NUREG 0737, TMI Action Plan, Item II.B.3. They currently describe that CPS has and maintains reactor coolant dissolved hydrogen monitoring capability at the PASS panel. CPS is retracting from this capability.

Appendix D item II.B.3. and Section 9.3.7.2 of the USAR describe PASS as determining reactor coolant dissolved hydrogen. USAR Table 7.1-13 item E13) E) presents range requirements for Dissolved Hydrogen. USAR Table 9.3-5 describe the PASS as obtaining reactor coolant dissolved hydrogen. These sections will be revised to reflect CPS position to not sample for reactor coolant dissolved hydrogen with the PASS. CPS is taking exception to the NUREG 0737 and Reg. Guide 1.97 rev. 3 requirements based on the intent of these documents to provide useful data as detailed below:

Reactor coolant dissolved hydrogen is not useful at CPS. Reactor coolant dissolved hydrogen is not used to determine core damage. Reactor coolant dissolved hydrogen is mentioned in NUREG 0737 as a verification that reactor coolant dissolved oxygen is at low levels not experienced in BWRs. Reactor coolant dissolved hydrogen does not support NUREG 0737 or Reg Guide 1.97 rev. 3 intent to support CPS reactor operator in accident mitigation, Emergency Response Organization personnel in core damage estimation, or Chemistry personnel in corrosion potential determination. Therefore, reactor coolant dissolved hydrogen is not needed.

L&S Log #93-0062

Maintaining reactor coolant dissolved hydrogen sampling capability is an unnecessary burden on CPS. Removing the unneeded sampling capability allows CPS to utilize resources more effectively for support of needed safety equipment. Chemistry personnel radiation exposures during accident mitigation will be lowered. Less highly radioactive material will be routed outside primary and secondary Containment during post accident conditions.

A.2 Identify the equipment, systems and parameters that may be affected by the change:

The Post Accident Sampling System Panel, 1PS02J, stripped gas section will no longer be required for post accident sampling of reactor coolant dissolved hydrogen. The elimination of this capability would simplify the PASS which is a non-safety, non-seismic system. Simplifying the PASS will allow CPS to more effectively maintain needed analysis capabilities.

No CPS operating parameter will be affected by this change. CPS reactor coolant dissolved hydrogen concentration is a function of radiolytic decomposition and reformation of water, degasification of hydrogen into the steam envelope, and formation due to metal-water/accelerated zircaloy corrosions. Reactor coolant dissolved hydrogen is not a normally obtained parameter. CPS does not attempt to control reactor coolant dissolved hydrogen.

CPS does not use reactor coolant dissolved hydrogen for core damage estimation. CPS does not use reactor coolant dissolved hydrogen as a backup for reactor coolant dissolved oxygen. Reactor coolant dissolved oxygen is a function of temperature, partial pressures, and reactor power. At CPS, reactor coolant dissolved oxygen is continuously monitored and never exists below the levels dissolved hydrogen back up sampling is intended to verify per Reg. Guide 1.97 R/3. Therefore, sampling oxygen is unnecessary. It is known that reactor coolant dissolved oxygen will not be below intergranular stress corrosion cracking concern levels. So, sampling dissolved hydrogen as a backup to dissolved oxygen is unnecessary.

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BLOCK B - RADWASTE TREATMENT SYSTEMS

B.1 The proposed activity involves a modification to a radiological waste treatment system or the way in which it is operated as described in Chapter 11 of the SAR. Yes ____ No X

B.2 Because: The PASS reactor coolant dissolved hydro n sampling equipment is not part of a radiological waste treatment system. Removing the CPS reactor coolant dissolved hydrogen sampling capability with the PASS will have no effect on any radiological waste treatment system.

If B.1 is yes, complete form NF-003.

BLOCK C - TECHNICAL SPECIFICATION IMPACT

C.1 The proposed activity requires a change to any part of the Technical Specifications. Yes ____ No X

C.2 Justification if "NO", Technical Specification change package identification number if "YES".

PASS Reactor coolant dissolved hydrogen sampling capability is not a Technical Specification requirement. It is not mentioned in Technical Specification Bases. Tech. Spec. 6.8.4 a/c require PASS leakage inspection/testing, maintenance, training, and procedures, but does not require PASS reactor coolant dissolved hydrogen sampling. Removing this PASS sampling capability will not require a change to any part of Technical Specifications. Letter U-600525/Y-206420 describes our commitment to PASS reactor coolant dissolved hydrogen sampling capability. The USAR and U-600525 will be revised accordingly.

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

(Attach additional pages with the responses to the block D questions. Identify your answers to Parts I, II, III, and IV.)

Part I - Impact on equipment malfunctions evaluated as the design basis

1. For the equipment and systems identified in A.1 and A.2, identify any failures evaluated in the SAR.

Failure of the PASS reactor coolant dissolved hydrogen sampling capability is not evaluated in the USAR. NUREG 0737 and Reg Guide 1.97 rev. 3 require backup grab sample capability for those PASS capabilities which use in-line monitors. CPS does not utilize an in line monitor for determination of reactor coolant dissolved hydrogen at PASS. The requirement for back up capability does not apply to CPS.

2. Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

   The PASS reactor coolant dissolved hydrogen sampling capability would be eliminated. The PASS design parameters will be unaffected by this change. The PASS may be better maintained as the system will have one less subsystem diverting resources from subsystems providing useful data. The PASS will continue to meet the NUREG 0737 and Reg. Guide 1.97 intent of supplying relevant data to Operations and Emergency personnel for estimating/
   mitigating core damage and long term corrosion potential.

3. Identify what new failure modes could be introduced by the change.

   No new failure modes could be introduced by deleting the PASS reactor coolant dissolved hydrogen sampling capability. Dissolved hydrogen does not pose a corrosion hazard in BWR water chemistry. It's only purpose is to verify dissolved oxygen levels below the levels which BWRs can attain. So, deselecting this parameter would not increase the chances of corrosion failure.

4. Identify any impact of the change on the consequences of the failures evaluated in the SAR.

   Removing the reactor coolant dissolved hydrogen sampling capability at PASS will not impact the consequences of the failures evaluated in the USAR. Dissolved hydrogen is not useful to operators in mitigating accidents. Dissolved hydrogen is not useful to the Emergency Response Organization in estimating core damage. Dissolved Hydrogen does not imply any useful data when determining post-accident corrosion potential. As dissolved hydrogen is not a useful parameter in meeting the intent of the PASS, deleting it will have no affect on the accidents evaluated in the USAR.

5. Identify any impact of the change on the probabilities of the failures evaluated in the SAR.

   The elimination of PASS reactor coolant dissolved hydrogen sampling capability will have no impact on the probability of any failure evaluated in the USAR.
   (See D.I.4)

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SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 4, are the consequences of any malfunction of equipment evaluated in the SAR increased?	YES ___ NO _X_
Based on item 5, is the probability of a malfunction of equipment evaluated in the SAR increased?	YES ___ NO _X_

If the answer to any of the above questions is yes, the change is an unreviewed safety question.

Part II - Impact on the accidents evaluated as the design basis

1. Identify the accidents evaluated in the SAR which could be affected by the change.

Removing the PASS reactor coolant dissolved hydrogen sampling capability will not affect any accident evaluated as the design basis of the USAR. The purpose of the PASS is to estimate/mitigate core damage and determine long-term post-accident corrosion potential. PASS reactor coolant dissolved hydrogen sampling is not useful in meeting any of these PASS design purposes. As the PASS will continue to meet its design intent, removing this sampling capability will not affect any analysis in the USAR. Additionally, USAR Chapter 15 accidents do not utilize PASS data.

2. Discuss how the change impacts the consequences of these accidents.

Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

No accident evaluated as the design basis of the USAR is affected by removing the PASS reactor coolant dissolved hydrogen sampling capability. Therefore, removing this capability will have no impact on the consequences of these accidents.

Furthermore, reactor coolant dissolved hydrogen results are not used by operators to mitigate core damage, by emergency response organization to determine core damage, or by chemistry to determine long-term post accident corrosion potential. As these are the intent of the PASS, the PASS support of accidents evaluated in the design basis of the USAR will not be impacted by removing the PASS reactor coolant dissolved hydrogen sampling capability. As removing this capability does not impact PASS support of USAR evaluated accidents, removing it will not impact the consequences of these accidents.

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3. Discuss how the change impacts the probability of these accidents.

   The PASS is non-safety/non seismic. The PASS is not a precursor to and has no impact on the accidents evaluated in the design bases of the SAR. Removing the PASS reactor coolant dissolved hydrogen sampling capability will simplify the PASS. Simplifying the PASS will have no impact on the probability of the accidents evaluated in the USAR. Therefore, removing the PASS reactor coolant dissolved hydrogen sampling capability will have no impact on the probability of accidents evaluated in the USAR.

SUMMARY

Based on item 2, are the consequences of an accident evaluated in the SAR increased?	YES ____ NO __X__
Based on item 3, is the probability of an accident evaluated in the SAR increased?	YES ____ NO __X__

If any of the above questions is answered yes, the change is an unreviewed safety question.

Part III - Potential for Creation of a New Unanalyzed Event

1. Based on Part I, items 1 and 3, could this change initiate a new type of accident or equipment malfunction? Discuss the basis for this determination.

   Removing the PASS reactor coolant dissolved hydrogen sampling capability will simplify the PASS. Simplifying the PASS will have no impact on the potential for initiating accidents and equipment malfunctions.

2. Determine if the new accident or malfunction identified above has sufficient probability or consequences to be considered in the Licensing basis. Discuss the bases for this determination.

   As no new accident or malfunction is identified above, the question of sufficient probability or consequences to be considered in the Licensing basis is not applicable.

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SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 2, does the change create the possibility for an equipment malfunction or accident of a different type than previously evaluated in the SAR? YES ____ NO __X__

If the answer if yes, the change represents an unreviewed safety question.

Part IV - Impact on the Margin of Safety

1. Identify how any of the protective barriers are directly affected by the change.

   Removing the PASS reactor coolant dissolved hydrogen sampling capability has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the Containment. Therefore, removing the capability has no direct or indirect affect on any of the protective barriers.

2. Discuss the impact of the change on the approach to the acceptance limits for any of the protective barriers.

   Removing the PASS reactor coolant dissolved hydrogen sampling capability has no affect on the parameters affecting the reactor fuel clad, reactor pressure vessel, or the containment. Therefore, removing this capability will have no impact on the approach to the acceptance limits for any of the protective barriers.

3. Discuss the impact of the change on the bases of the Technical Specifications.

   The PASS reactor coolant dissolved hydrogen sampling capability is not stated or implied in the Bases of the Technical Specifications. Removing this capability will have no impact on the Bases of the Technical Specifications.

SUMMARY

Based on item 2, is any parameter in chapter 7 of the Safety Evaluation Manual exceeded? YES ____ NO __X__

Based on items 2 and 3, does the change reduce the margin of safety provided for the protective barriers? YES ____ NO __X__

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If the first of these two questions is answered yes, the change may be unsafe and requires further justification. If the first question is answered no and the second is answered yes, the change is safe to implement but is an unreviewed safety question and requires prior NRC approval.

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SAFETY EVALUATION FORM

BLOCK E - SUMMARY

Based on the evaluation in Block C and Block D, the change

_X_ is safe and is not an unreviewed safety question and requires no Technical Specification change. The change may be implemented in accordance with applicable procedures.

____ is safe but is an unreviewed safety question or requires a Technical Specification change. The change requires NRC approval, prior to implementation.

____ is unsafe and cannot be implemented.

	printed name	signature	date
Preparer	Tom Leffler	Tom Leffler	4/27/93
Director	D.E. Korneman	D E Korneman	4/27/93
Manager, NSED	N/A		
Manager, L&S (SSG 4/29/93)	J.L. Peterson	J L Peterson	5-3-93
FRG	P.D. Yocum		5/6/93

EVIDENCE OF NRC APPROVAL, IF REQUIRED:

License Amendment No. N/A

printed name	signature	date
N/A	SSG 4/29/93	

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NF-002-5 (12/90)
M6274/314/L

SAFETY EVALUATION FORM

1.1 L&S Log # 93-0063

Document Evaluated:

1.2 Number: USAR 1.3 Revision: 4

1.4 Title: Removing Post Accident Sampling System (PASS) Conductivity Monitoring Capability

1.5 References:

USAR 9.3.7	Reg. Guide 1.97 Rev. 3
USAR App. D.II.B.3	Tech Spec 6.8.4.a. c.
USAR Table 7.1-13/9.3-5	Tech Spec 3.4.4
NUREG 0853 part 9.3.5	U-600525 (5/5/86)
NUREG 0737, II.B.3	
NUREG-0578	
NUREG-0660	

BLOCK A - DESCRIPTION OF CHANGE
(Use additional pages if required)

A.1 Describe the basic document or system and the changes being made. Discuss how the change affects the SAR description. Discuss the reason for change.

The PASS Evaluation Report (U-600525 on 5/5/86) and USAR describe CPS' compliance with NUREG 0737, TMI Action Plan II.B.3. The report currently describes that CPS has conductivity monitoring capability at the PASS panel for reactor coolant. CPS is retracting from conductivity monitoring capability at the PASS panel for reactor coolant based on the following reasons:

The USAR states in section 9.3.7 and in appendix D.II.B.3 that CPS will maintain conductivity monitoring capability at the PASS panel. The SAR also presents the NRC position which does not require CPS to maintain conductivity monitoring capability at the PASS panel. Conductivity monitoring capability for reactor coolant at the PASS panel is not required by NUREG 0737 or Reg. Guide 1.97 rev. 3.

Maintaining this capability is an unnecessary burden on CPS. CPS already maintains Tech Spec instruments for reactor coolant conductivity which require plant shutdown and depressurization for out-of-specification conductivity. Removing the unneeded capability allows CPS to utilize more reasonable controls and maintenance practices. Chemistry personnel mitigating accidents will receive less radiation exposure collecting unneeded data. Less highly radioactive material will be routed outside primary and secondary containment during post accident conditions.

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(Continued from page 1)

A.2 Identify the equipment, systems and parameters that may be affected by the change:

The conductivity meter, 1CI-PS825, in the Post Accident Sampling System Panel, 1PS03J, will no longer be required to be maintained for post accident sampling. The PASS conductivity monitor for reactor coolant will be used as a back-up monitor to Tech Spec 3.4.4 monitors. CPS takes no credit for this monitor as a Tech Spec or PASS monitor. It will be a for information only monitor.

No CPS operating parameter will be affected by this change. CPS will maintain continuous monitoring of reactor coolant conductivity as required by Technical Specification 3/4.4.4. CPS will maintain conductivity within Tech Spec 3/4.4.4 limits or shutdown, cooldown, and depressurize the reactor as required by Tech Spec 3/4.4.4.

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BLOCK B - RADWASTE TREATMENT SYSTEMS

B.1 The proposed activity involves a modification to a radiological waste treatment system or the way in which t is operated as described in Chapter 11 of the SAR. Yes ____ No X

B.2 Because: The PASS conductivity meter is not a part of any radiological waste treatment system. The PASS conductivity meter is not described or implied in chapter 11 of the SAR and has no affect on radwaste treatment systems.

If B.1 is yes, complete form NF-003.

BLOCK C - TECHNICAL SPECIFICATION IMPACT

C.1 The proposed activity requires a change to any part of the Technical Specifications. Yes ____ No X

C.2 Justification if "NO", Technical Specification change package identification number if "YES".

Technical Specifications do not directly address the PASS conductivity meter. Technical Specification 6.8.4.a requires the PASS be maintained and inspected to reduce leakage and 6.8.4.c requires maintenance of PASS sampling and analysis equipment. These Technical Specifications will not change.

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

(Attach additional pages with the responses to the block D questions. Identify your answers to Parts I, II, III, and IV.)

Part I - Impact on equipment malfunctions evaluated as the design basis

1. For the equipment and systems identified in A.1 and A.2, identify any failures evaluated in the SAR.

   The SAR does not identify any failure of the PASS conductivity monitor. NUREG 0737 and Reg. Guide 1.97 rev. 3 require CPS to maintain back-up sampling capability for those parameters monitored with in-line instruments. The requirement to have back-up capability for conductivity is eliminated with removal of the inline conductivity monitoring capability.

2. Discuss the impact of the change on the performance of the equipment and systems identified in A.1 and A.2.

   The PASS design parameters will be unaffected by removing the reactor coolant conductivity capability. This change has no impact on the performance of the PASS.

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(Continued from page 2)

3. Identify what new failure modes could be introduced by the change.

   No new failure mode could be introduced by retracting PASS reactor coolant conductivity monitoring capability. This equipment does not impact other equipment operation. Removing this equipment will not change other equipment failure modes. Therefore, no new failure modes can be created by removing this equipment.

4. Identify any impact of the change on the consequences of the failures evaluated in the SAR.

   Retracting the CPS capability of monitoring the reactor coolant conductivity at PASS has no impact on the consequences of the failures evaluated in the USAR.

5. Identify any impact of the change on the probabilities of the failures evaluated in the SAR.

   Retracting the CPS capability of monitoring the reactor coolant conductivity at PASS has no impact on the probabilities of the failures evaluated in the USAR. (See D.I.3 above)

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SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 4, are the consequences of any malfunction of equipment evaluated in the SAR increased? YES ____ NO X

Based on item 5, is the probability of a malfunction of equipment evaluated in the SAR increased? YES ____ NO X

If the answer to any of the above questions is yes, the change is an unreviewed safety question.

Part II - Impact on the accidents evaluated as the design basis

1. Identify the accidents evaluated in the SAR which could be affected by the change.

   No accident evaluated in the USAR will be affected by retracting the PASS reactor coolant conductivity monitor. The PASS is not taken credit for mitigating the USAR chapter 15 accidents.

2. Discuss how the change impacts the consequences of these accidents.

   No consequence of any accident evaluated in the USAR will be impacted by retracting the reactor coolant conductivity monitor at the PASS panel. (See D.II.1 above)

3. Discuss how the change impacts the probability of these accidents.

   No impact on the probability of any accident evaluated in the USAR will result from retracting the reactor coolant conductivity monitor at the PASS panel. (See D.II.1 above)

SUMMARY

Based on item 2, are the consequences of an accident evaluated in the SAR increased? YES ____ NO X

Based on item 3, is the probability of an accident evaluated in the SAR increased? YES ____ NO X

If any of the above questions is answered yes, the change is an unreviewed safety question.

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(Continued from page 3)

Part III - Potential for Creation of a New Unanalyzed Event

1. Based on Part I, items 1 and 3, could this change initiate a new type of accident or equipment malfunction? Discuss the basis for this determination.

   Retracting the reactor coolant conductivity monitor at the PASS panel will not initiate any new type of accident or equipment malfunction. The PASS conductivity meter has no support function for any other equipment. Therefore, eliminating CPS conductivity monitoring capability with PASS will not compromise any other piece of equipment. The elimination of this capability would simplify the PASS which is a non-safety, non-seismic system. Simplifying the PASS will not initiate a new type of accident or equipment malfunction.

2. Determine if the new accident or malfunction identified above has sufficient probability or consequences to be considered in the Licensing basis. Discuss the bases for this determination.

   Not applicable. See D.III.1.

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SAFETY EVALUATION FORM

BLOCK D - UNREVIEWED SAFETY QUESTION DETERMINATION

SUMMARY

Based on item 2, does the change create the possibility for an equipment malfunction or accident of a different type than previously evaluated in the SAR?

YES ____
NO __X__

If the answer if yes, the change represents an unreviewed safety question.

Part IV - Impact on the Margin of Safety

1. Identify how any of the protective barriers are directly affected by the change.

   Retracting the reactor coolant conductivity monitor capability at the PASS panel will have no affect on the fuel cladding, the reactor coolant pressure boundary, or the Containment. Reactor coolant conductivity monitoring required by Technical Specification 3/4.4.4 will still continuously verify reactor coolant conductivity within parameter boundaries and require reactor shutdown and depressurization if reactor coolant conductivity poses any threat to fuel clad or reactor coolant pressure boundary integrity.

2. Discuss the impact of the change on the approach to the acceptance limits for any of the protective barriers.

   Retracting the reactor coolant conductivity monitor capability at the PASS panel will not impact the approach to the acceptance limits for any of the protective barriers. As noted above, Technical Specification 3/4.4.4 Instrumentation will continue to monitor conductivity for fuel clad and reactor coolant pressure boundary integrity.

3. Discuss the impact of the change on the bases of the Technical Specifications.

   Reactor coolant conductivity monitoring at the PASS panel is not addressed in the bases of the Technical Specifications. Retracting the reactor coolant conductivity monitor at the PASS panel will have no impact on the bases of the Technical Specifications.

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(Continued from page 4)

SUMMARY

Based on item 2, is any parameter in chapter 7 of the Safety Evaluation Manual exceeded?	YES ____ NO X
Based on items 2 and 3, does the change reduce the margin of safety provided for the protective barriers?	YES ____ NO X

If the first of these two questions is answered yes, the change may be unsafe and requires further justification. If the first question is answered no and the second is answered yes, the change is safe to implement but is an unreviewed safety question and requires prior NRC approval.

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SAFETY EVALUATION FORM

BLOCK E - SUMMARY

Based on the evaluation in Block C and Block D, the change

_X_ is safe and is not an unreviewed safety question and requires no Technical Specification change. The change may be implemented in accordance with applicable procedures.

____ is safe but is an unreviewed safety question or requires a Technical Specification change. The change requires NRC approval, prior to implementation.

____ is unsafe and cannot be implemented.

	printed name	signature	date
Preparer	Tom Leffler	[signature]	4/27/93
Director	D.E.Korneman	D.E.Korneman	4/27/93
Manager, NSED	NA		
Manager, L&S (SSG 4/29/93)	J.L. Peterson	J.L.Peterson	5-3-93
FRG	[signature]	[signature]	5/4/93

EVIDENCE OF NRC APPROVAL, IF REQUIRED:

License Amendment No. N/A

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