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HUNTER,R.S. Indiana & Michigan Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards detailed info re NUREG-0737,Item II.K.3, "Post-Accident Sampling Sys," requested in NRC 820630 ltr. Postponement of committed date for sys to be operational requested.New date will be provided by 821231.

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November 5, 1982
AEP:NRC:0716A

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NUREG-0737, ITEM II.B.3 - POST-ACCIDENT SAMPLING SYSTEM

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D., C. 20555

Dear Mr. Denton:

This letter and its Attachment provide the detailed information pertaining to Item II.B.3 of NUREG-0737, "Post-Accident Sampling System" requested by Mr. S. A. Varga's letter of June 30, 1982. As stated in our letter dated August 4, 1982, AEP:NRC:0716, Indiana & Michigan Electric Company (I&MECo.) believes that several of the clarifications provided in the Attachment to Mr. Varga's letter represent significant expansions of the requirements for the Post-Accident Sampling System beyond those requirements set forth in NUREG-0737 and, henceforth, utilized for system design.

As detailed in the Attachment to this letter, our Post-Accident Sampling system provides, in several areas, alternatives to your clarifications which we consider technically justified. There are some areas, however, where further work is still needed either to meet your requirements or to provide a valid alternative. Because of this situation we request a postponement of the committed date for the system to be operational in Unit 1 of the Cook Plant. This date would have been November 9, 1982. We cannot provide you a new date as of this writing, but will do so prior to December 31, 1982. We note however, that, to a large extent, the system is essentially complete in Unit 1. The Unit 2 portion of the system is currently scheduled for completion by approximately February 10, 1983.

A046

This document has been prepared following Corporate Procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

R. S. Hunter
Vice President

/os

cc: John E. Dolan - Columbus
M. P. Alexich
R. W. Jurgensen
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
Joe Williams, Jr.
NRC Resident Inspector at Cook Plant

ATTACHMENT TO AEP:NRC:0716A
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2
NUREG-0737, ITEM II.B.3-POST-ACCIDENT SAMPLING SYSTEM

POST-ACCIDENT SAMPLING SYSTEM

GENERAL DESCRIPTION

The Donald C. Cook Nuclear Plant Post-Accident Sampling (PAS) System is designed to provide representative samples of reactor coolant and containment atmosphere following a loss-of-coolant accident. The system has the capability to promptly obtain samples under accident conditions without incurring a radiation exposure to any individual in excess of 5 and 75 rem to the whole body and extremities, respectively.

As shown in Figure 1, the PAS System is shared by both Units and is capable of taking samples from the following locations in either Unit under different accident scenarios:

1. Reactor Coolant System Loop 1 and 3 Hotlegs
2. Reactor Coolant System Pressurizer Steam Space
3. Residual Heat Removal System (RHR) - East and West RHR Heat Exchanger Outlets
4. Lower Containment Sump

To facilitate installation of the PAS within the original schedular requirements, connections were made into the existing Nuclear Sampling and Radiation Monitoring System sample piping where possible. A new installation was required for the containment sump sample. The sample lines are run to the PAS Liquid and Gas Sample Panel, and are shown on Figures 1 through 5.

The System includes five panels, namely; PAS Liquid and Gas Sample Panel, PAS Instrument Panel, PAS Control Panel, Unit 1 PAS Valve Panel, and Unit 2 PAS Valve Panel.

Figure 6 shows the approximate location of each of these panels within the spray additive tank room (SATR) which is located at the east end of the Auxiliary Building on elevation 587'.

The PAS Liquid and Gas Sample Panel is designed for in-line analysis of undiluted samples and for collection of diluted samples for transfer to the hot laboratory/counting room for analysis. For undiluted liquid samples, in-line analyses are provided for pH and dissolved oxygen. Gas chromatography is used for dissolved hydrogen analysis of undiluted reactor coolant and containment atmosphere samples. Diluted liquid and containment atmosphere grab samples will be analyzed for radionuclides, boron concentration and other parameters as required. A schematic of the PAS Liquid and Gas Sample Panel is shown on Figure 7. This shielded panel is vented through a HEPA-charcoal

filtration unit which discharges to the auxiliary building ventilation system.

The PAS Instrument Panel provides the PAS Liquid and Gas Sample Panel with calibration and carrier gases for the gas chromatograph, nitrogen for purging, and demineralized water for dilution and flushing. The panel also contains the pH and O_2 calibrators needed for calibration and checkout of the in-line pH and O_2 analyzers in the PAS Liquid and Gas Sample Panel.

The PAS Control Panel contains the controls needed for remotely operating the PAS Liquid and Gas Sample and Panel and is used in conjunction with the PAS Valve Panels for lining up the proper flow path for each sample.

The PAS Valve Panels contain the switches for operating containment isolation valves and remote-operated valves needed to lineup and divert a sample from the required system or component to the PAS Liquid and Gas Sample Panel.

RESPONSES TO CRITERIA

Criterion 1- Response:

The D. C. Cook Plant's hot laboratory and counting room, the primary analysis facilities, are located on elevation 609' of the auxiliary building in the Controlled Access Area, approximately 220' from the PAS station in the SATR on the 587' level. This arrangement is shown on Figures 8 and 9. The sample from the PAS Liquid and Gas Sample Panel is removed by a syringe and transported in a shielded container to the hot laboratory/counting room.

In the unlikely event that the onsite hot laboratory/counting room facilities are inaccessible or unusable, the Palisades Nuclear Plant, located approximately 35 miles north of Cook Plant in South Haven, Michigan, also has these facilities. Arrangements are being made for this back-up support.

There are three categories of specific time intervals to consider; sampling, transport, and analysis. The sampling time interval includes the time it takes to purge (recirculate) the sample and manipulate the associated controls at the PAS system panels for either a direct monitor reading or for a diluted grab sample. The maximum purge time is approximately 40 minutes based on length of the sample tubing and flow rate. The maximum time required for aligning the sample path is approximately 10 minutes based on the equipment design and preoperational testing.

The maximum transport time interval is approximately 15 minutes via the longest possible route between the PAS station and the onsite hot laboratory. The radionuclide counting procedure is the

longest analysis procedure performed for PAS, and requires approximately 60 minutes.

Therefore, the expected total time required for sampling, transport, and analysis of the required samples at the D. C. Cook Plant is less than three hours.

If the offsite laboratory must be used, sample transport will be affected by many additional factors such as sample packaging, weather, and traffic. It is estimated that these operations will add a minimum of 1-2 hours to the time required for sample transport.

The PAS was not designed to operate following a loss of off-site power. This new design consideration is thought to be unnecessary given the time scales involved in collecting and analyzing a sample and the very high reliability of the Cook Plant off-site power sources. Details of the Cook Plant's off-site power connections and a discussion of the loss of off-site power events (none a total loss) are contained in our letter No. AEP:NRC:0292, dated January 3, 1980. Note, however, that the five PAS panels are either powered from an emergency power supply in the event of a loss of off-site power, or from the 250 V.D.C. station battery system in which case they would be unaffected by a loss off-site power.

<u>Criterion 2 - Response:</u>

2(a) The clarification asks for "provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters." The problem is being reviewed; other utilities are being contacted to find out how they are addressing the problem. Current information and procedures from other utilities will be reviewed to assist in the development of an interim procedure for use at the Cook Plant until a final method can be implemented. We currently anticipate having the interim procedure in place by February 28, 1983.

2(b) A diluted grab sample for containment hydrogen analysis can be taken with the PAS Liquid and Gas Sample Panel. The sample is extracted into a syringe, inserted in a lead cask and transported to the on-site laboratory for hydrogen analysis using a gas chromatograph.

2(c) The capability to sample and to analyze for the dissolved gases, chloride, boron concentration and accident sample species is discussed in the responses to Criteria 1, 4, 5, 7 and 9.

2(d) There is in-line capability for monitoring hydrogen in the containment atmosphere, and dissolved hydrogen and oxygen and pH in the reactor coolant. Containment hydrogen and dissolved

hydrogen concentrations are determined by gas chromatography using a Baseline Model 1030A analyzer. The dissolved oxygen concentration is determined by the polarographic method using a Yellow Springs Instrument Co. Model 56 analyzer. The pH analyzer uses a Cole Palmer combination electrode assembly and Model 5650 monitor. The NUS Corporation, our PAS system consultant, has performed in-depth testing of these analyzers to demonstrate their applicability for this service.

These instruments were selected based on:

1. simplicity of operation
2. dependability
3. accuracy of technique
4. wide measurement range
5. limited operator manipulation during analyses sequence
6. applicability to accident conditions.

Criterion 3-Response:

Process auxiliary systems carrying reactor coolant or containment atmosphere gases which are isolated post-accident, i.e., the letdown system or the reactor water cleanup system, are not required to be placed in operation, but portions of the existing Nuclear Sampling System must be placed in service to obtain post-accident samples.

The PAS System has been designed with provisions for purging (recirculating) both liquid and gas samples without the use of those isolated auxiliary systems. This is discussed in the General Description and in the Criteria 11 Response.

The only valves considered inaccessible inside containment are part of the normal sampling system for sampling of the reactor coolant hot legs and the pressurizer steam space. These valves are air operated and fail closed upon loss of air. These valves are under investigation for possible improvement or replacement. The accessibility and/or qualification of PAS System valves located outside of containment is currently under review in an attempt to identify those valves which are required to function for PAS operation and the time dependent radiation levels in the areas where non-qualified valves are used. I&MECo. requests an extension to the time for PASS operability pending resolution of this matter.

Criterion 4-Response

The amount of dissolved hydrogen in the reactor coolant is determined by stripping the hydrogen gas from a pressurized sample and measuring the hydrogen concentration using the gas chromatograph. The dissolved oxygen concentration is determined from a pressurized reactor coolant sample using an in-line oxygen probe/analyzer.

These methods measure the total dissolved hydrogen and oxygen concentrations. Therefore the measured concentrations are directly related to reactor coolant concentrations.

The range of the dissolved hydrogen analyzer is 0.5 to 2000 cc/Kg. The range of the dissolved oxygen analyzer is 0.1 to 20.0 ppm. Therefore either analyses can be used to verify low oxygen levels.

Criterion 5-Response:

Since the cooling water supply for D. C. Cook Nuclear Plant is from Lake Michigan, which is a fresh water supply source, the chloride analysis will be performed within the required 96 hour time limit.

The PAS Liquid and Gas Sample Panel provides a 1000:1 diluted grab sample. The mercuric thiocynate method, having a range of 5 ppb to 1 ppm, will be used to determine chlorides in the diluted samples. Therefore the lower limit of detectability in the reactor coolant is 5 ppm.

As discussed in Criterion 4, total dissolved gases can be determined with the in-line analyzers and the results used to verify if oxygen levels are below 0.1 ppm should chlorides exceed 0.15 ppm.

The PAS System does not have the capability to take an undiluted sample, since it was designed on the basis of NUREG-0578 and NUREG-0737 which emphasized ALARA considerations, and did not specify the requirement for undiluted grab samples.

Since the chloride concentration in the cooling water source (Lake Michigan) is in the range of 5 to 20 ppm, we expect that any chloride inleakage would result in insignificantly low concentrations in the reactor coolant.

Criterion 6-Response:

Criterion 6 states that the design basis for taking and analyzing a sample of reactor coolant or containment air must assume that any individual would not receive a dose that exceeds 5 rems to the whole body or 75 rems to the extremities. Calculations were made of the dose received during each part of taking, transporting and analzying a sample of reactor coolant.

The worst case accident, with a 401 ml/min leak rate of reactor coolant water into the auxiliary building and the plant vent not operating, makes the auxiliary building inaccessible. A more realistic case, the one used to calculate dose, considers a 1 ml/min leak rate with the plant vent not in operation. Other assumptions used in the analysis are that the sample is 5 ml of reactor coolant diluted by a factor of 1000 and that the letdown system was not isolated.

The dose in the hot lab and counting room was calculated for two cases. In the first case, the air supply system into the hot lab and counting room area was not operating; this would allow airborne radiation into the hot lab and counting room. In the second case, the air supply system is working; this provides a positive pressure from the hot lab to the auxiliary building and prevents airborne radioactivity from entering the hot lab or counting room.

The dose received by one individual taking, transporting and anlyzing the sample for the first case is 2.1 rems to the whole body and 55.5 rems to the hands. In the second case, the dose to the hands is the same, but the dose to the whole body is 1.1 rems. Both cases are within the GDC-19 criterion requirements.

Criterion 7-Response:

Boron analysis using the curcumin method is performed on a 1000:1 diluted grab sample taken from the PAS Liquid and Gas Sample Panel.

Criterion 8-Response:

In-line monitoring is used for dissolved hydrogen, dissolved oxygen and pH analyses. Diluted back-up grab samples for these parameters can be taken with the PAS Liquid and Gas Sample Panel. As discussed in the Criterion (7) response, the panel is not capable of taking undiluted samples.

The samples for dissolved hydrogen and oxygen are extracted into a glass syringe which is inserted in a lead cask and transported to the on-site laboratory for analyses by gas chromatography. Employing the chromatographic procedure eliminates the concern regarding use of a moderately diluted sample.

The PAS System is designed to flush the in-line pH monitor with demineralized water to facilitate access for checking and maintenance. The panel would be removed from service, flushed and checked should this be necessary, rather than to determine pH on a grab sample which would only provide questionable results. The dissolved hydrogen and oxygen monitors are also provided with flushing capabilities.

Criterion 9-Response:

The predicted isotopes and activites are listed in Table I and II for the reactor coolant and containment atmosphere samples, respectively. The tables also show the correspondence with the source terms given in Regulatory Guides 1.4 and 1.7.

Provisions to reduce personnel radiation exposure, and to permit sample handling include:

1. Shielding of those sides of the PAS Liquid and Gas Sample Panel to which the operator would be exposed during sample collection.

2. Use of a shielded syringe to remove the diluted grab samples from the PAS Liquid and Gas Sample Panel and a lead cask to transport them to the appropriate analysis location.
To ensure that sample activity will be reduced sufficiently to perform the required analysis, the dose rate of the sample will be determined and appropriate supplementary dilutions made as required.

Range of measurement for nuclides will be within the 1 uCi/g to 10 Ci/g limits. The dilutions will reduce the activity content to the level of the normal sampling capabilities.

Criterion 9(b) concerns the predicted background radiation levels in the counting room and the effect of the background radiation on the counter. The ventilation system provides higher pressure in the hot lab and counting room than in the auxiliary building. This will prevent airborne radiation from entering the counting room. If the ventilation system is not working, the background radiation would be 981 mR/hr. If the system is working but the letdown is not isolated, the background radiation is 50 mR/hr.

The counter is shielded by a 4 inch lead cave. The radiation field inside the cave with a 50 mR/hr background would be 0.01 mR/hr. This field would not prevent proper counting since the counter operates properly in a background of up to 10 mR/hr. If the air supply system was not operating, the airborne radioactivity would make the counter inoperable.

As indicated in Figure 8, the hot lab and counting room are two different rooms. The only sample that would be in the counting room would be the one being counted. Since this sample would have to be diluted several times to be counted, it would not contribute significantly to the background.

Criterion 10-Response:

Gross activity gamma spectrum is determined using the normal procedure for reactor coolant samples. The range is as specified in the Criterion 9 response with an accuracy within a factor of 2 based on plant measurements.

Our consultant qualified a curcumin boron method for post-accident application on 1000:1 dilutions of post-accident matrix solutions. The analysis range is approximately 0.2 to 2.0 ppm (200 to 2000 ppm boron in the coolant) with relative standard deviation of approximately ± 13%. To measure values in excess of 2000 ppm, appropriate sample dilutions will be performed.

Chloride is determined by the mercuric thiocyanate method. The range is 5 ppb to 1 ppm chloride (5ppm to 1000 ppm in the coolant) This is the procedure now used for determining chloride in reactor coolant samples.

Based on our consultant's data, the gas chromatograph may be used to determine dissolved hydrogen concentration in the range of 3 to 2000 cc/Kg with an accuracy meeting the requirements of the post-implementation guidelines.

Based on our consultant's final verification testing, the dissolved oxygen analyzer can be used during post-accident conditions to determine dissolved oxygen concentrations with an accuracy of at least ±10 percent in the test range of 0.1 to 10 ppm. While the consultant did not test to 20 ppm, the linear response obtained indicates that measurements up to 20 ppm are achievable and the accuracy will be within ±10%.

Our consultant tested the pH probe by flowing PWR matrices through the pH probe and comparing monitor readings with grab sample measurements for the stream effluent. The data obtained indicates an absolute mean bias of ±0.3 pH units over a pH range of 5 to 8.

The equipment and analytical methods used for post-accident sampling will be calibrated and tested in accordance with plant procedures. Refresher training in post-accident sampling, analysis and transport are scheduled. The frequencies for equipment calibration and testing are also included in the specified schedules.

Criterion 11(a)-Response:

The PAS incorporates means for purging the gas and liquid sample lines as shown on Figures 1 and 7. To reduce plate out and minimize sample distortion, sample velocities will be maintained in the range of standard sampling practices. Sample lines and coolers are constructed of corrosion resistant material. To minimize sample loss, the sample lines up to the PAS Liquid and Gas Sample Panel are all of a welded construction. Dead legs and crud traps were minimized.

All sample lines inside the PAS Liquid and Gas Sample Panel can be flushed with demineralized water or nitrogen to minimize blockage. Filters are installed in the Panel's liquid sample inlet line and in the containment sump sample line.

Sample purge fluids are returned to the containment. Sample runs are kept to the minimum possible to limit the volume of fluid to be taken. The sample lines and sample purge return line have remotely operated containment isolation valves to shut off sample flow should a line rupture or excessive leakage occur.

To provide representative sampling, reactor coolant samples are taken from the normal reactor coolant 1 and 3 hot legs and pressurizer steam space. Samples are also taken from the containment sump sample pump recirculation line, and either RHR heat exchanger outlet in the main flow path downstream of the RHR pumps. The containment atmosphere sample line is taken from the lower containment volume.

Criterion 11(b)-Response:

A dedicated sample station filtration system which shall include HEPA and charcoal filters is being provided in the ventilation exhaust from the sampling station. This filtration system is scheduled to be completed on or before November 30, 1982.

TABLE - I

PREDICTED POST-ACCIDENT REACTOR COOLANT SAMPLE
ISOTOPES AND ACTIVITIES

Isotope	Activity (Ci)	Activity Conc.(ci/cc)
Kr-85	8.86×10^{5}	2.483×10^{-3}
Kr-85m	4.30×10^{7}	1.205×10^{-1}
Kr-87	7.79×10^{7}	2.183×10^{-1}
Kr-88	1.06×10^{8}	2.971×10^{-1}
Sr-89	8.06×10^{7}	2.259×10^{-3}
Sr-90	7.27×10^{6}	2.038×10^{-4}
Y-90	7.71×10^{6}	2.161×10^{-4}
Y-91	1.06×10^{8}	2.971×10^{-3}
Zr-95	1.59×10^{8}	4.456×10^{-3}
Zr-97	1.59×10^{8}	4.456×10^{-3}
Nb-95	1.59×10^{8}	4.456×10^{-3}
Nb-95m	1.95×10^{6}	5.465×10^{-5}
Nb-97	1.68×10^{8}	4.709×10^{-3}
Mo-99	1.77×10^{8}	4.961×10^{-3}
Tc-99m	1.59×10^{8}	4.456×10^{-3}
Ru-103	1.68×10^{8}	4.709×10^{-3}
Ru-106	5.85×10^{7}	1.64×10^{-3}

Table - I (cont'd)

Isotope	Activity (Ci)	Activity Conc. (Ci/cc)
Cs-134	2.30×10^{7}	6.446×10^{-4}
Cs-136	6.38×10^{6}	1.788×10^{-4}
Cs-137	1.06×10^{7}	2.971×10^{-4}
Ba-137m	9.75×10^{6}	2.733×10^{-4}
Ba-140	1.680×10^{8}	4.709×10^{-3}
La-140	1.77×10^{8}	4.961×10^{-3}
Ce-141	1.59×10^{8}	4.456×10^{-3}
Ce-143	1.42×10^{8}	3.98×10^{-3}
Ce-144	1.24×10^{8}	3.475×10^{-3}
Pr-143	1.42×10^{8}	3.98×10^{-3}
Pr-144	1.24×10^{8}	3.475×10^{-3}
Nd-147	6.29×10^{7}	1.763×10^{-3}
Pm-148	2.04×10^{7}	5.718×10^{-4}
Pm-148m	8.86×10^{6}	2.483×10^{-4}
Pm-149	6.29×10^{7}	1.763×10^{-3}
Sm-153	6.29×10^{7}	1.763×10^{-3}
Eu-156	3.28×10^{7}	9.143×10^{-4}
Rh-103m	1.68×10^{8}	4.709×10^{-3}
Rh-105	1.06×10^{8}	2.971×10^{-3}
Rh-106	6.82×10^{7}	1.911×10^{-3}

Table - I (cont'd)

Isotope	Activity (Ci)	Activity Conc.(Ci/cc)
Ag-110m	6.38×10^{5}	1.788×10^{-5}
Ag-111	5.67×10^{6}	1.589×10^{-4}
Sb-125	9.75×10^{5}	2.733×10^{-5}
Sb-127	1.06×10^{7}	2.971×10^{-4}
Te-127	1.06×10^{7}	2.971×10^{-4}
Te-129	3.28×10^{7}	9.913×10^{-4}
Te-129m	8.86×10^{6}	2.483×10^{-4}
Te-132	1.42×10^{8}	3.980×10^{-3}
I-131	9.57×10^{7}	1.366×10^{-1}
I-132	1.42×10^{8}	1.990×10^{-1}
I-133	1.95×10^{8}	2.733×10^{-1}
I-134	2.19×10^{8}	3.069×10^{-1}
I-135	1.70×10^{8}	2.382×10^{-1}
Xe-133	1.95×10^{8}	5.465×10^{-1}
Xe-133m	2.84×10^{7}	7.960×10^{-2}
Xe-135	5.31×10^{7}	1.488×10^{-1}
Xe-135m	5.22×10^{7}	1.463×10^{-1}

TABLE II

PREDICTED POST-ACCIDENT
CONTAINMENT ATMOSPHERE SAMPLE
ISOTOPES AND ACTIVITIES

Isotope	Activity (Ci)	Activity Conc. (Ci/cc)
Kr-85	8.86×10^{5}	8.491×10^{-5}
Kr-85m	4.30×10^{7}	4.121×10^{-3}
Kr-87	7.79×10^{7}	7.465×10^{-3}
Kr-88	1.06×10^{8}	1.016×10^{-2}
Sr-89	8.060×10^{7}	7.724×10^{-5}
Sr-90	7.27×10^{6}	6.967×10^{-6}
Y-90	7.710×10^{6}	7.389×10^{-6}
Y-91	1.060×10^{8}	1.016×10^{-4}
Zr-95	1.590×10^{8}	1.524×10^{-4}
Zr-97	1.590×10^{8}	1.524×10^{-4}
Nb-95	1.590×10^{8}	1.524×10^{-4}
Nb-95m	1.950×10^{6}	1.869×10^{-6}
Nb-97	1.680×10^{8}	1.610×10^{-4}
Mo-99	1.77×10^{8}	1.696×10^{-2}
Tc-99m	1.59×10^{8}	1.524×10^{-4}
Ru-103	1.68×10^{8}	1.61×10^{-4}
Ru-106	5.85×10^{7}	5.606×10^{-5}

Table - II (cont'd)

Isotope	Activity (Ci)	Activity Conc. (Ci/cc)
Rh-103m	1.68×10^{8}	1.61×10^{-4}
Rh-105	1.06×10^{8}	1.016×10^{-4}
Rh-106	6.82×10^{7}	6.536×10^{-5}
Ag-110m	6.38×10^{5}	6.114×10^{-7}
Ag-111	5.67×10^{6}	5.434×10^{-6}
Sb-125	9.75×10^{5}	9.344×10^{-7}
Sb-127	1.06×10^{7}	1.016×10^{-5}
Te-127	1.06×10^{7}	1.016×10^{-5}
Te-129	3.28×10^{7}	3.143×10^{-5}
Te-129m	8.86×10^{6}	8.491×10^{-6}
Te-132	1.42×10^{8}	1.361×10^{-4}
I-131	9.57×10^{7}	2.336×10^{-3}
I-132	1.42×10^{8}	3.402×10^{-3}
I-133	1.95×10^{8}	4.672×10^{-3}
I-134	2.19×10^{8}	5.247×10^{-3}
I-135	1.70×10^{8}	4.073×10^{-3}
Xe-133	1.95×10^{8}	1.869×10^{-2}
Xe-133m	2.84×10^{7}	2.722×10^{-3}
Xe-135	5.31×10^{7}	5.089×10^{-3}
Xe-135m	5.22×10^{7}	5.002×10^{-3}
Cs-134	2.30×10^{7}	2.204×10^{-5}
Cs-136	6.38×10^{6}	6.114×10^{-6}
Cs-137	1.06×10^{7}	1.016×10^{-5}

Table - II (cont'd)

Isotope	Activity (Ci)	Activity Conc. (Ci/cc)
Ba-137	9.75×10^{6}	9.344×10^{-6}
Ba-140	1.68×10^{8}	1.610×10^{-4}
La-140	1.77×10^{8}	1.696×10^{-4}
Ce-141	1.59×10^{8}	1.524×10^{-4}
Ce-143	1.42×10^{8}	1.361×10^{-4}
Ce-144	1.24×10^{8}	1.188×10^{-4}
Pr-143	1.42×10^{8}	1.361×10^{-4}
Pr-144	1.24×10^{8}	1.188×10^{-4}
Nd-147	6.29×10^{7}	6.028×10^{-5}
Pm-148	2.04×10^{7}	1.955×10^{-5}
Pm-148m	8.86×10^{6}	8.491×10^{-6}
Pm-149	6.29×10^{7}	6.028×10^{-5}
Sm-153	6.29×10^{7}	6.028×10^{-5}
Eu-156	2.28×10^{7}	3.143×10^{-5}

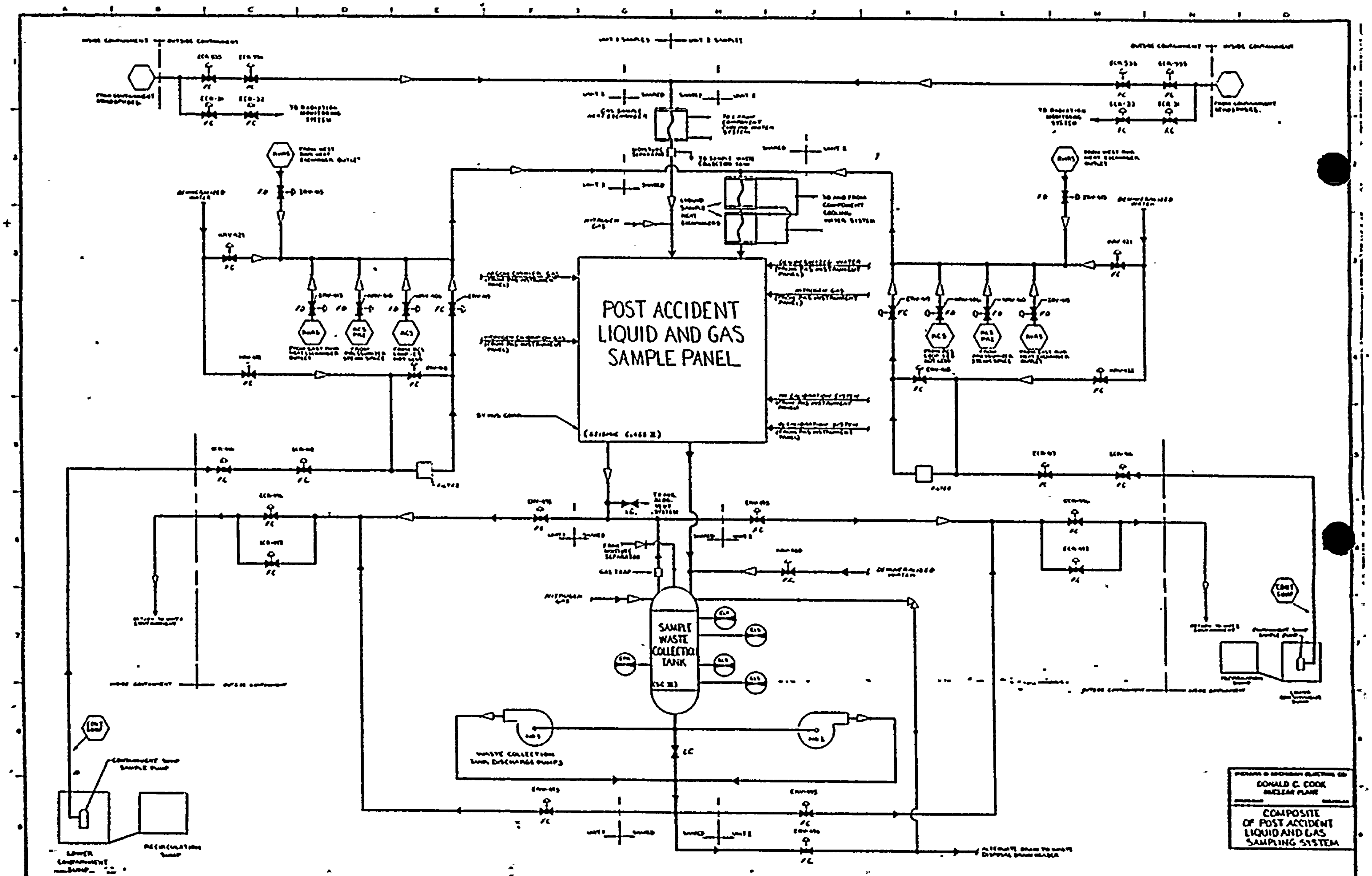


FIGURE 1

AEP:NRC:0716A

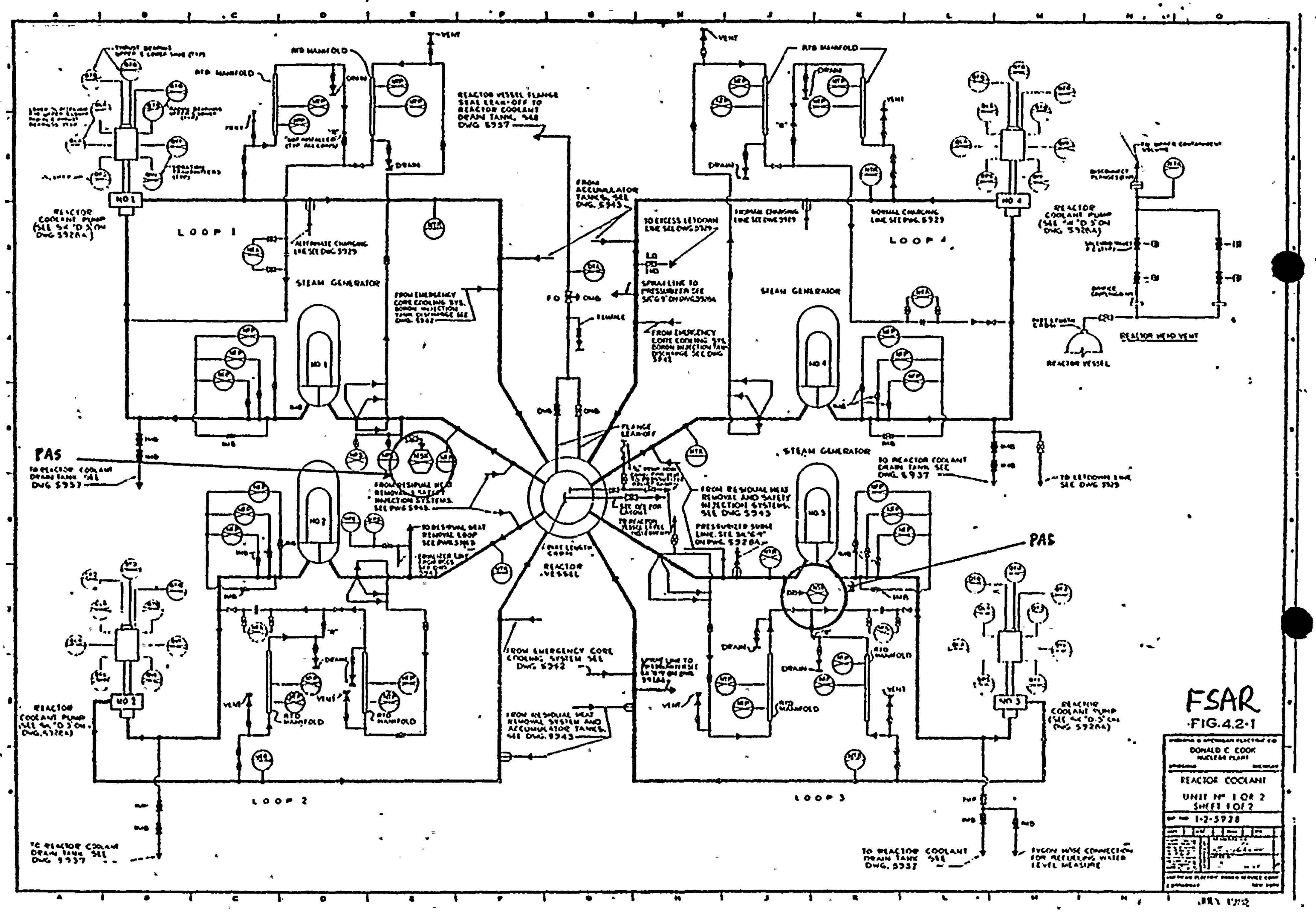


FIGURE-2

AEP:NRC:716A

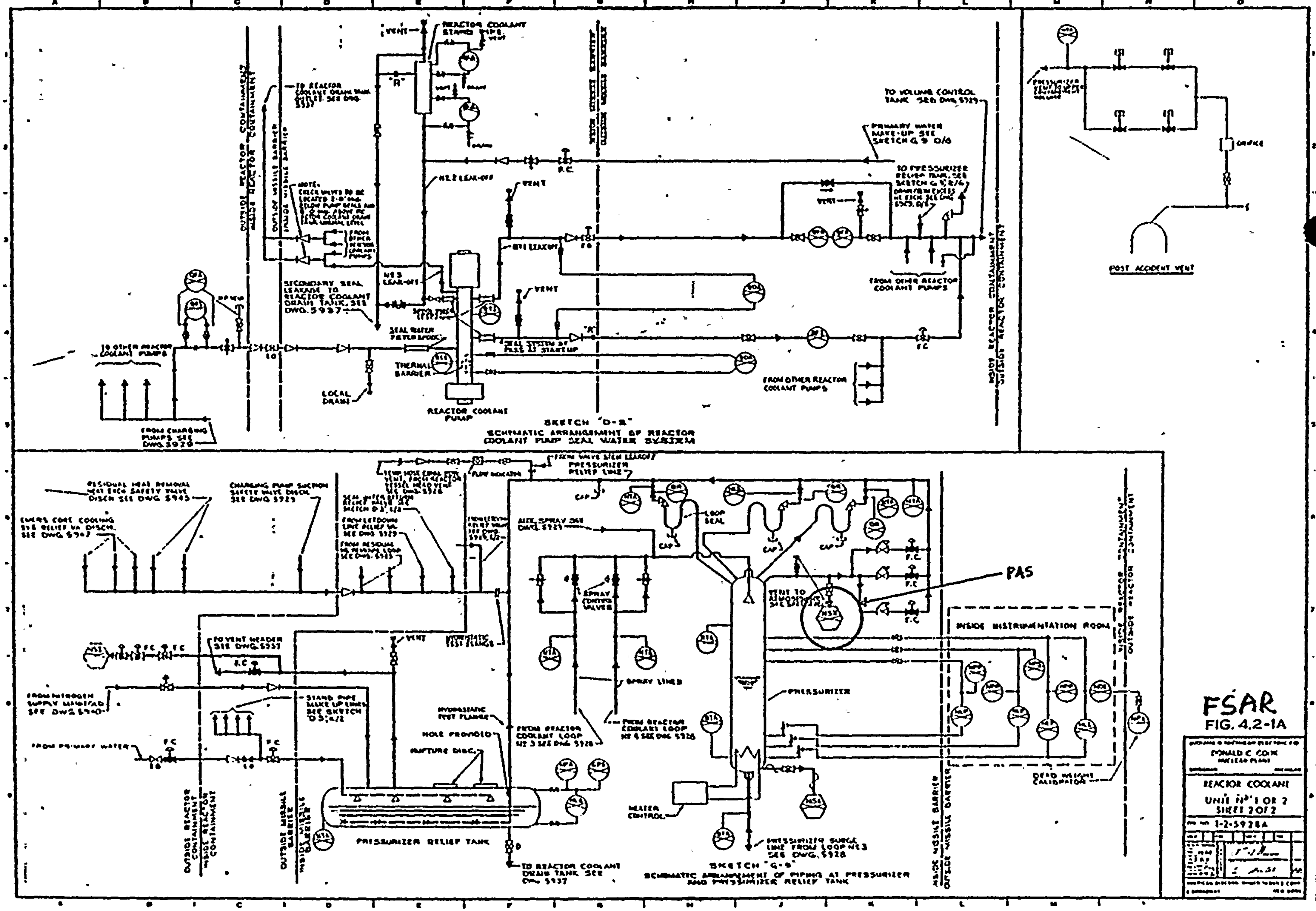


FIGURE-3

AEP: NRC-716A

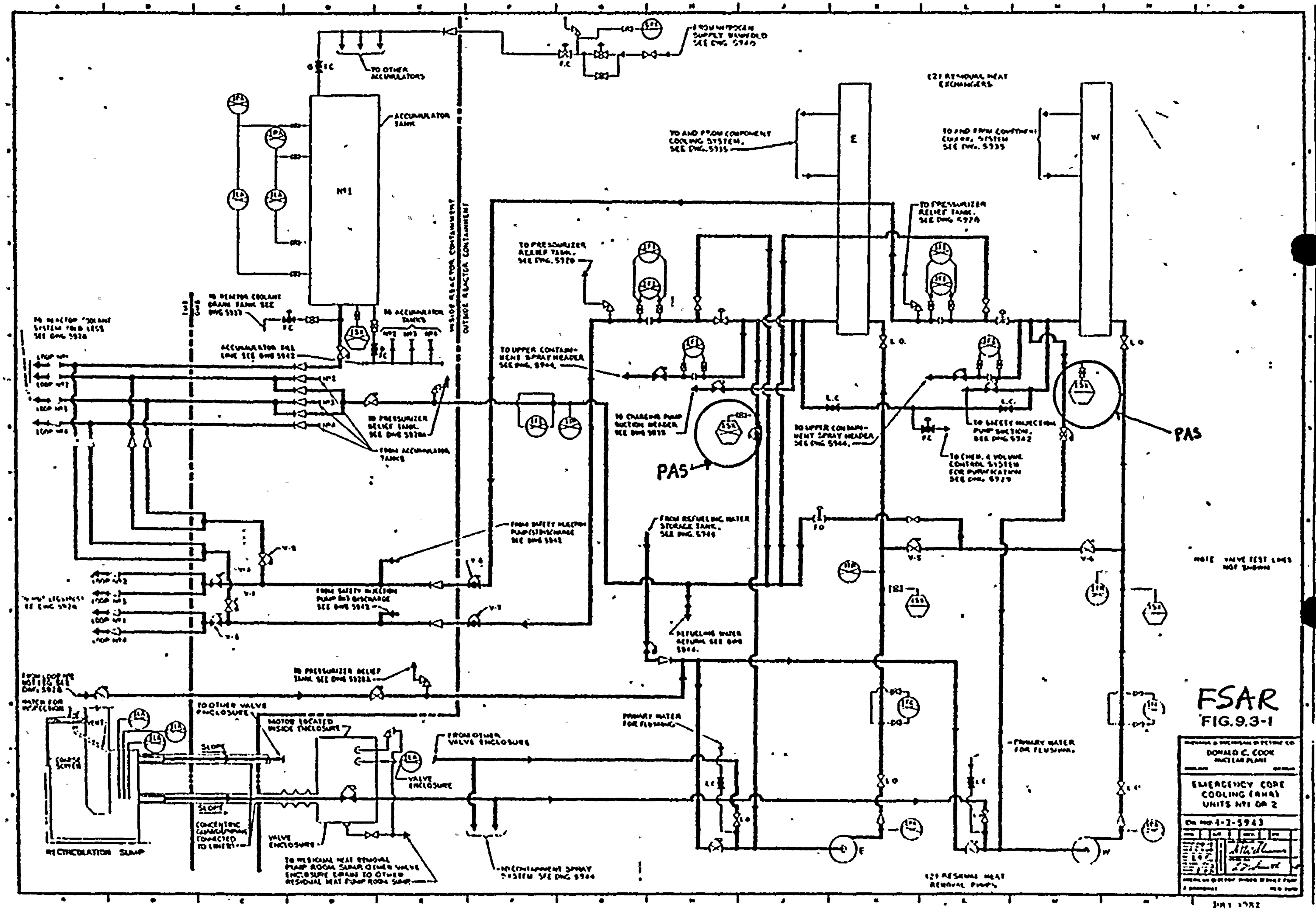


FIGURE-4

AEP:NRC:716A

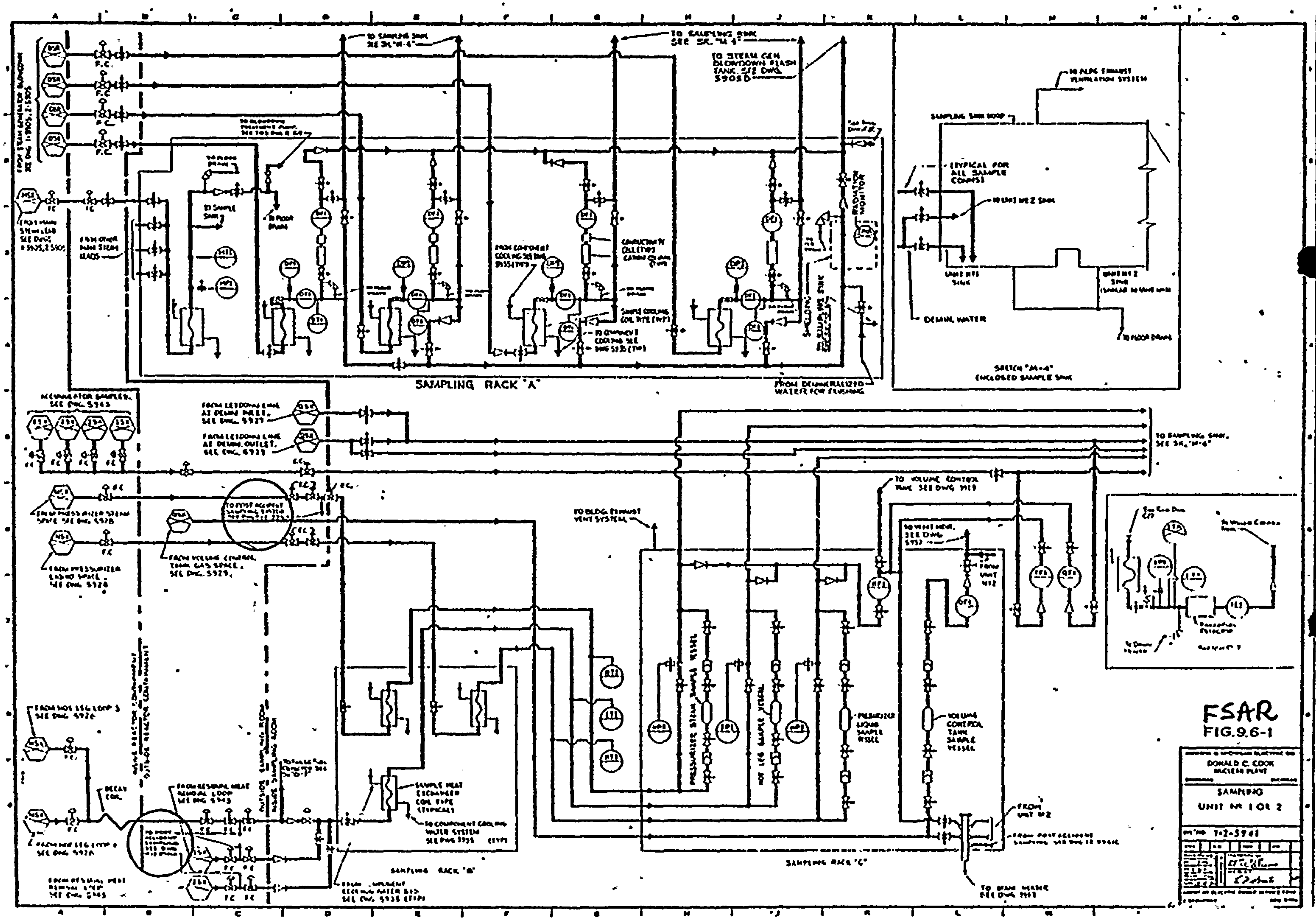
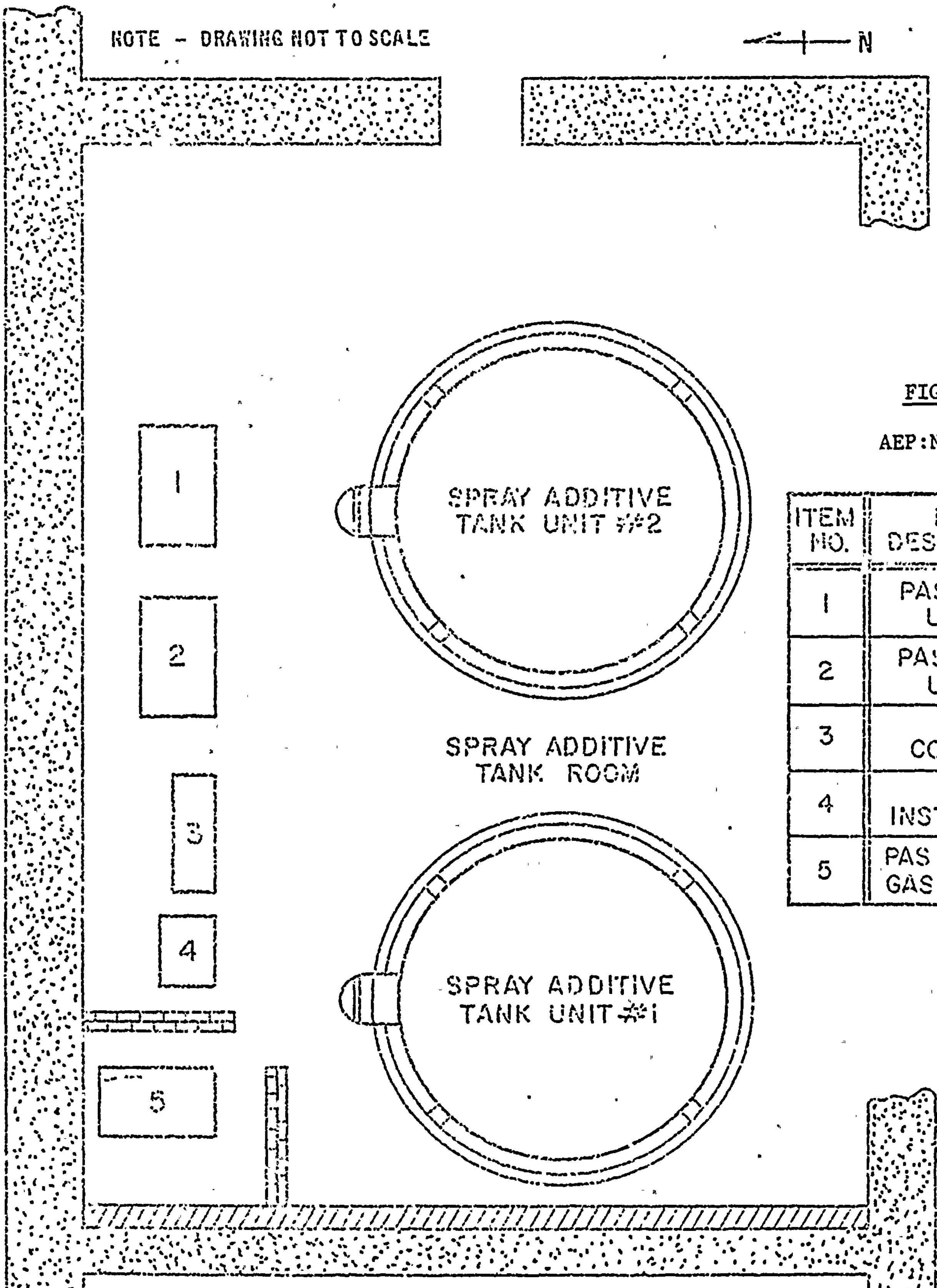


FIGURE-5

AEP:NRC:716A

FIGURE 6

AEP:NRC:0716A



ITEM NO.	PANEL DESCRIPTION
1	PAS VALVE UNIT-2
2	PAS VALVE UNIT-1
3	PAS CONTROL
4	PAS INSTRUMENT
5	PAS LIQUID & GAS SAMPLE

INDIANA & MICHIGAN ELECTRIC COMPANY

D C COOK NUCLEAR PLANT UNITS 1 & 2

ARRANGEMENT OF POST-ACCIDENT SAMPLING EQUIPMENT IN SPRAY ADDITIVE TANK ROOM

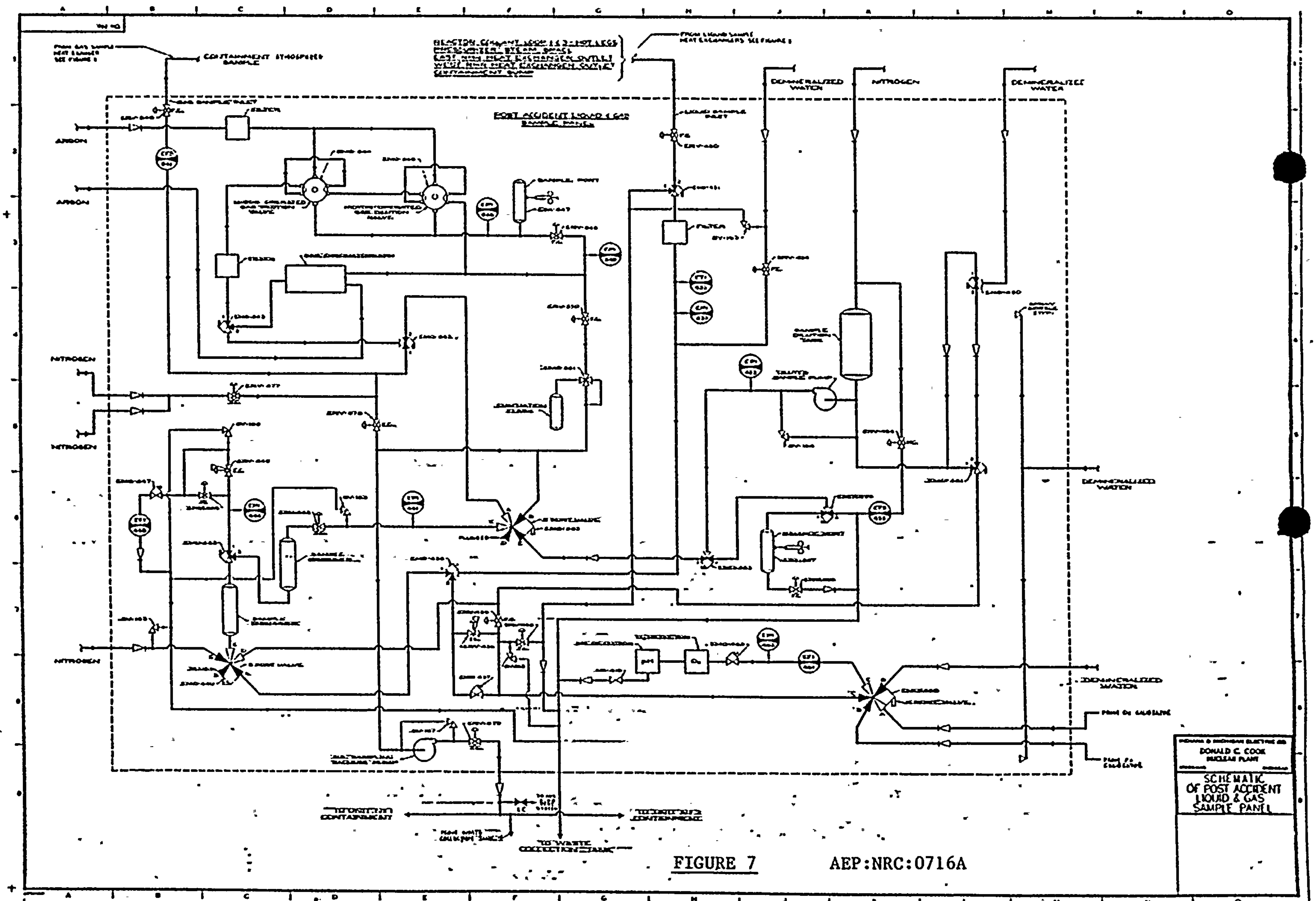


FIGURE 7

AEP:NRC:0716A

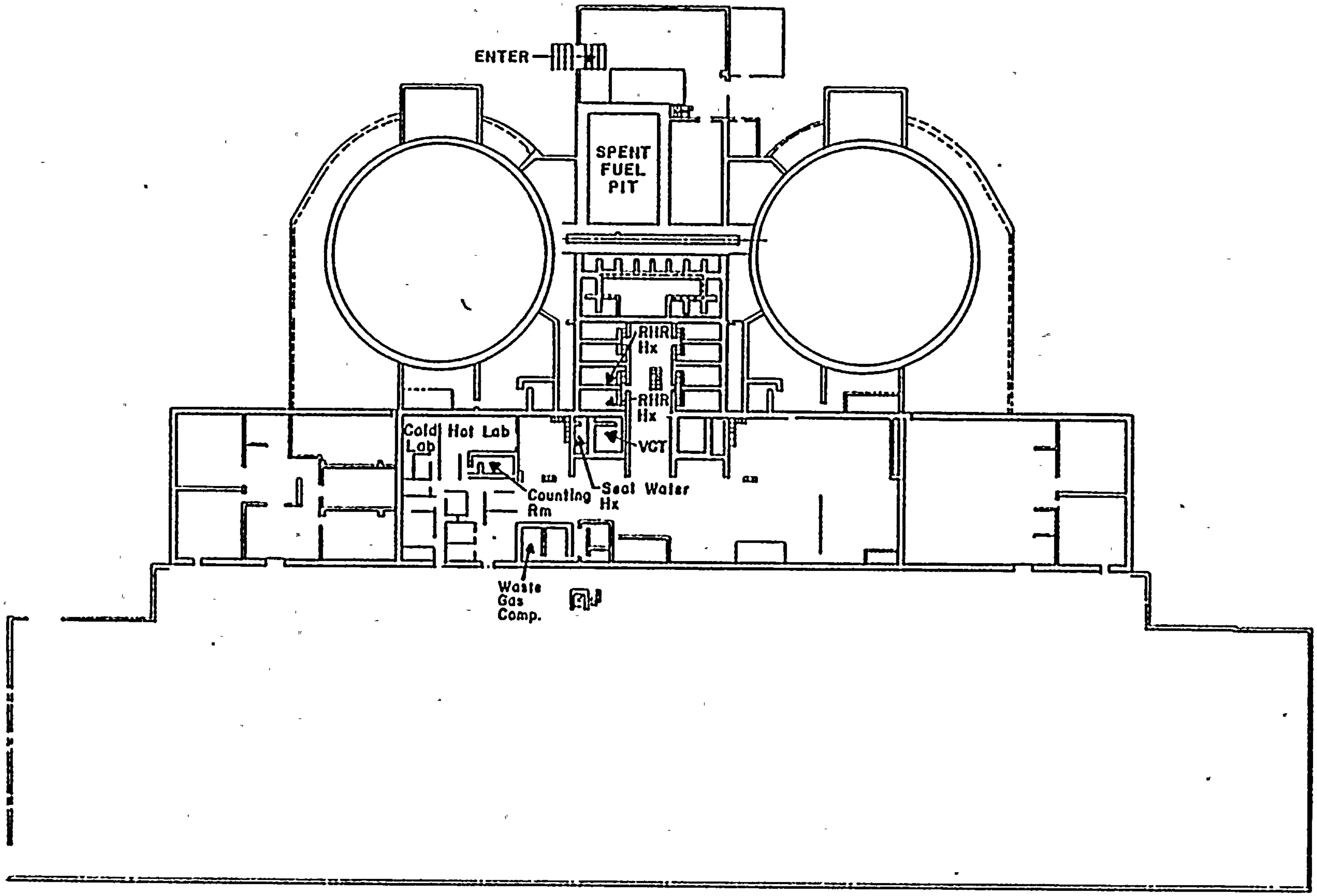


FIGURE - 8

POST-ACCIDENT SAMPLING LOCATION,
SHOWING ACCESS ROUTE FROM OUTSIDE THE
AUXILIARY BUILDING, ELEVATION 609

AEP:NRC:716A

D. C. COOK NUCLEAR PLANT

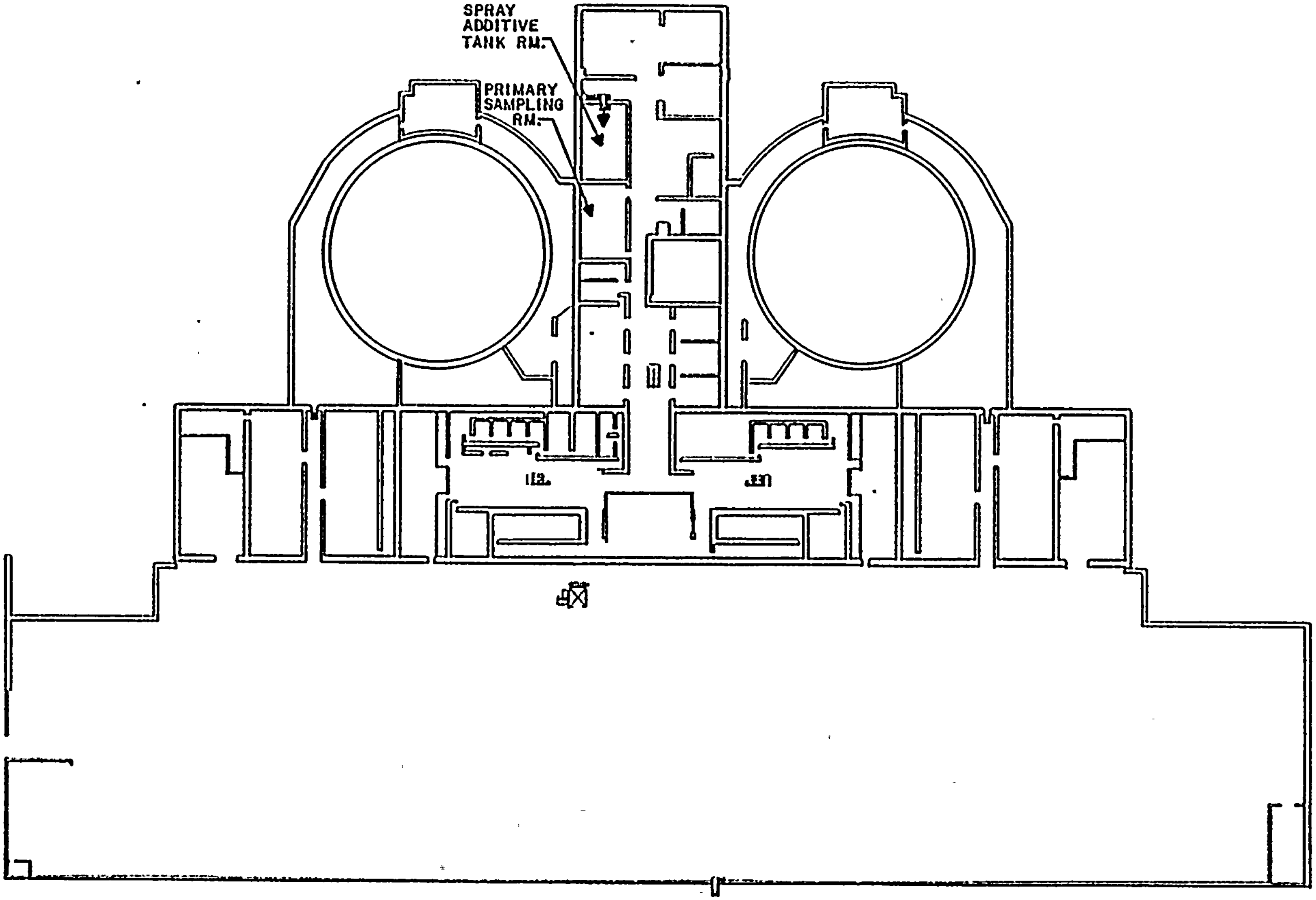


FIGURE - 9

POST-ACCIDENT SAMPLING LOCATION,
SPRAY ADDITIVE TANK ROOM, ELEVATION 587

D. C. COOK NUCLEAR PLANT

AEP:NRC:716 A

[illegible]