ATTACHMENT A

RESPONSES TO 6/30/82 NRC INFORMATION REQUEST CONCERNING POST ACCIDENT SAMPLING

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
OCTOBER 1982

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ATTACHMENT A

Criterion: (1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

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

Response to Clarification 1

The primary coolant sample cask was designed with a nominal 3½ inches of lead shielding within a steel housing. It is transported via a 3-wheeled dolly designed to easily withstand the weight of the sample cask. The primary coolant sample is removed through a specially shielded sample port located on the 55' elevation of the primary auxiliary building (Attachment 1A and 1B). The sample cask can be enclosed in the sample port shielding while samples are being removed. The sample cask is then wheeled via the dolly approximately 112' to a chain fall located over a floor grate. The grate is lifted by hand, and the cask is lowered to the 41' level. It is then transported approximately 74' to the analysis cave (Attachment 1C). The total distance from point of sample to the analysis cave is approximately 200' and requires approximately one-half hour to complete. Sample laboratories are located within 10' of the sample and analysis caves.

The containment atmospheric sampling cave is located on the 41' level of thh primary auxiliary building (Attachment 1D). The sample is transported via a lead cask weighing approximately 18 pounds to the chemistry counting room located on the 61' level (4th floor) of the Administrative Services Building; a distance of approximately 800'.

The primary coolant sample is taken from the hot leg loops of the reactor coolant system through the normal sampling lines to which the post-accident sampling system is attached (see Attachment 1A). The unused sample is then returned to the volume control tank.

The containment air sample is recirculated to containment via a pump. Particulates and iodines are vented to the exhaust vent duct.

Two full-scale drills have been performed to determine whether sampling and analytic times would meet the three-hour time requirements. The results of these drills have been documented and are available, on site, for inspection. On May 11, 1982, both plant vent and containment samples were obtained and analyzed in 2 hours and 25 minutes. On May 12, 1982, a primary coolant sample was obtained and analyzed in 2 hours 40 minutes. The question of radiation exposure will be addressed in the Response to Clarification 6.

All equipment vital to post-accident sampling and analysis are connected to MCC37. During loss of offsite power, this equipment is backed up by a site emergency diesel generator.

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

(a) 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 non-volatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Clarification: 2 (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non-volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{188}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

Response to Clarification 2(a)1

Radiochemical analyses are performed using a Nuclear Data (ND) 6620 counting system with Ge(Li) detector. A brief description of the system's hardware and software capabilities is contained in Attachment 2(a)1-1. In accordance with the Indian Point 3 standard health physics practices, sampling equipment is not handled after sampling without measurement of dose rate and evaluation. If handling of a highly radioactive source is required, remote carrying tools or shielded containers are used. The time that these sources are handled is kept to a minimum.

The Ge(Li) detector is shielded with a standard lead brick cave with copper and cadmium liners to suppress background radiation. In addition, the sample storage vault is approximately 90' from the cave with a minimum of 52" of concrete interrupting a straight line path from the sample vault to the detector. The counting room walls are constructed of 18" of concrete to further aid background suppression. The ND 6620 system has the capability of automatic subtraction of background spectra as a standard feature.

The Authority currently has the capability to monitor 68 different nuclides associated with the nuclear fuel cycle, including those specified in Clarification 2(a)1 and the Rogovin Report, Vol. II, Part 2, pp. 524-527 (Attachment 2(a)1-2), with the exception of Te, Br and Pd. Currently an in-house procedure is being developed which takes into account the Authority's analytic capabilities and expertise to relate radionuclide concentration to core damage.

Clarification: 2(b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

Response to Clarification 2(b)

Attachment 2(b)-1 shows a diagram of the vapor containment sampling system with a gas bulb located between valves CAS 2 and 4. A hydrogen sample of approximately 10 μl is removed from this bulb and analyzed via a gas chromatograph per procedure with results reported in percent H_2.

Clarification: 2(c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

Response to Clarification 2(c)

The Authority currently has the ability to sample and analyze reactor primary coolant for gross activity, liquid isotope, boron, chloride, dissolved hydrogen, total gas, dissolved O_2 and pH. Gross activity and liquid isotope samples are removed via the primary coolant post-accident sampling system (55' - PAB) and analyzed on the ND 6620 system previously described. Chloride and pH are measured on separate samples (due to sample preparation necessary for chloride analysis) with specialized electrodes contained within the analysis cave on the 41' level of the PAB. Dissolved hydrogen and total gas are analyzed using a gas chromatograph located in the sample analysis lab (41' level - PAB). Boron concentrations are determined by plasma emission spectrometry.

Clarification: 2(d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line isntrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Response to Clarification 2(d)

A grab sampling system is the method by which the Authority would obtain post-accident samples at Indian Point 3. The only in-line instrumentation currently installed and operating at IP3 is a Rex-Nord oxygen analyzer. The Rex-Nord oxygen analyzer is not being used to meet any of the criteria as specified in NUREG 0737, Section II.B.3 or any clarifications thereof.

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 hours 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: BWR's 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 hours. All other plants have 96 hours to perform a chloride analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at < 0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Response to Clarification 5

Indian Point 3 uses brackish water from the Hudson River as the heat sink for the fan cooler units. Under certain postulated accident conditions, this could be construed as a single barrier protection between the reactor coolant and the river water. Chloride analysis can be performed on an essentially undiluted sample in-house within a 24-hour period and the results are reported as ppm Cl. Since the maximum dilution involved is a factor of two, the lower limit of detectability of the instrument (18 ppb) assures an accurate

concentration value. The question of verification of dissolved oxygen levels will be addressed in the 180 day response which includes clarification 4.

Criterion: (6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H.R. Denton to all licensees).

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

Response to Clarification 6

Exposure to individuals operating the post-accident sampling system are based on a 2.75 Ci/g source term as per Reg. Guide 1.4 with no dilution other than the reactor coolant system. This results in 5.4 R/HR at one foot from the sample cask. The 30 minute handling period would give a maximum exposure to an individual of 2.7 rem. Assuming a 10 Ci/g source term, per NUREG 0737, which is significantly greater than the design basis accident postulated for IP3, 20 R/HR at one foot from the cask would result. Thus, over 30 minutes, an exposure of 10 rem is possible. In this case, alternating of personnel performing the sampling and analysis would be required.

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: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Response to Clarification 7

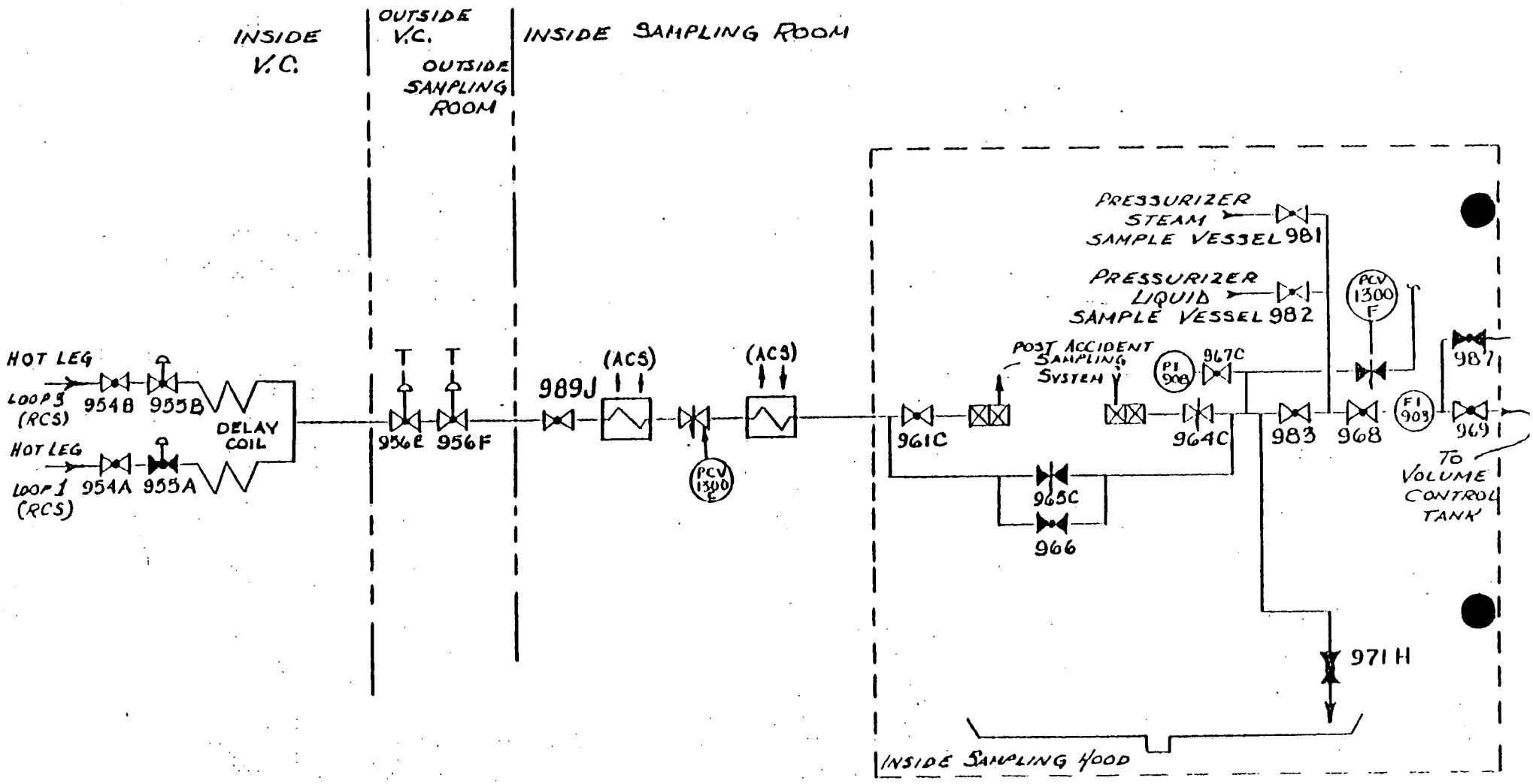
As previously stated in the response to clarification 2(c), boron analysis is performed by plasma emission spectrometry.

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.

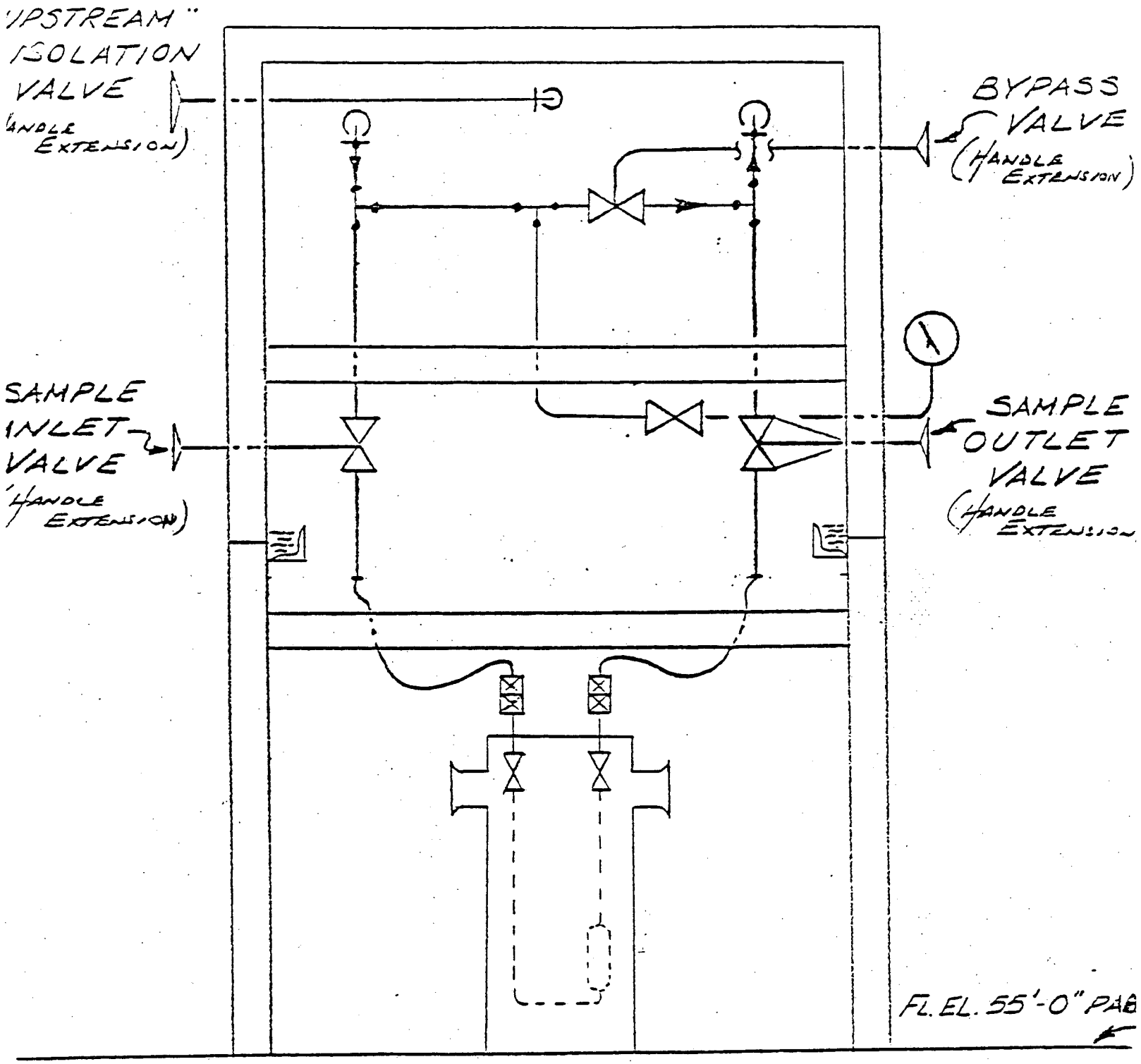
Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis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

Response to Clarification 8

The post-accident sampling system at IP3 provides for withdrawal of an undiluted sample, and facilities are available to perform dilutions as necessary. Currently, no in-line monitors are used. An in-line O_2 monitor is available. The grab sampling system can be partially flushed for maintenance. Currently, there are three sample casks designed such that they can be easily shipped if analysis by an outside laboratory becomes necessary. In addition, five waste casks are available to flush contents of the sample casks and the analysis cave.

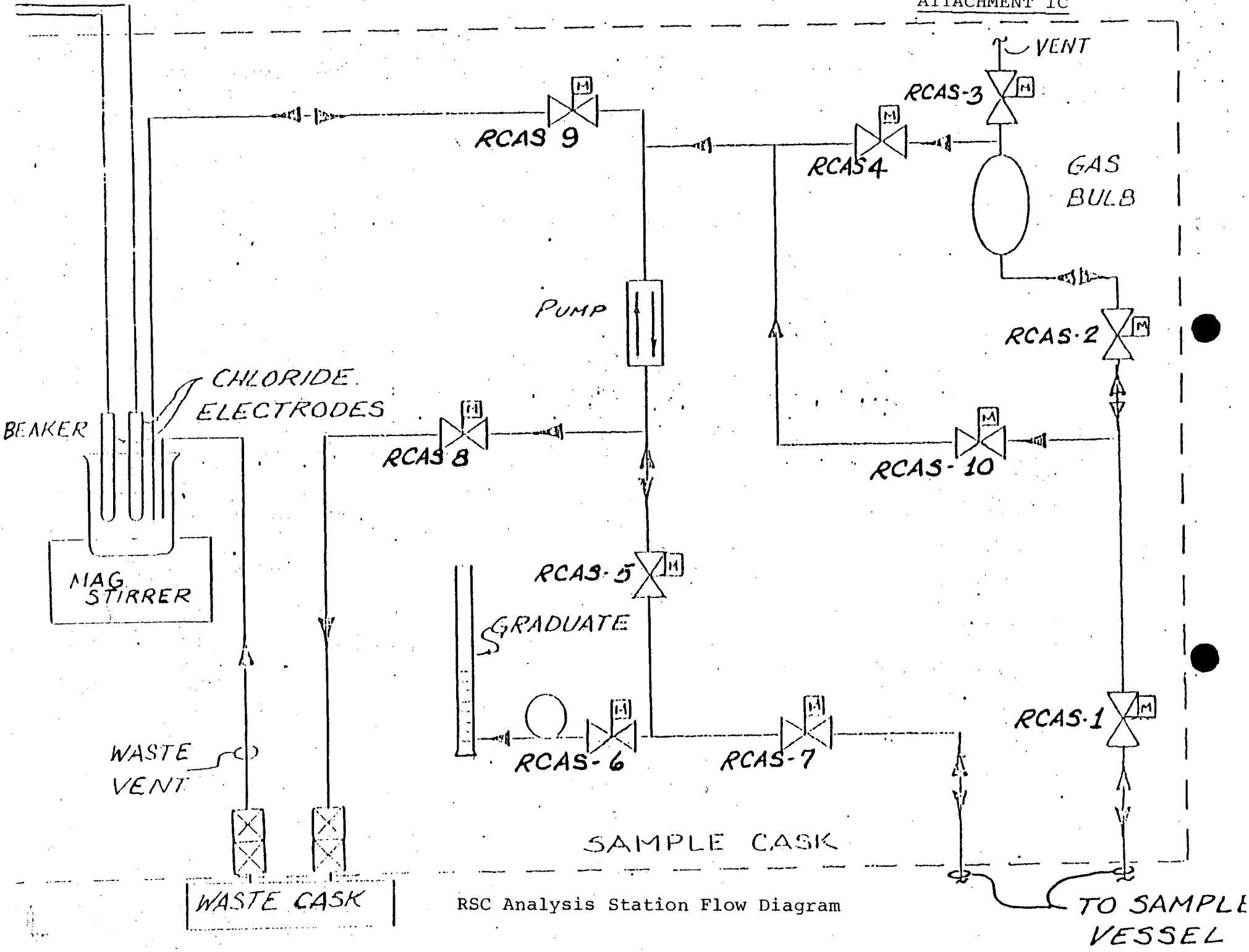


Reactor Coolant - Post-Accident Sampling System Flow Diagram

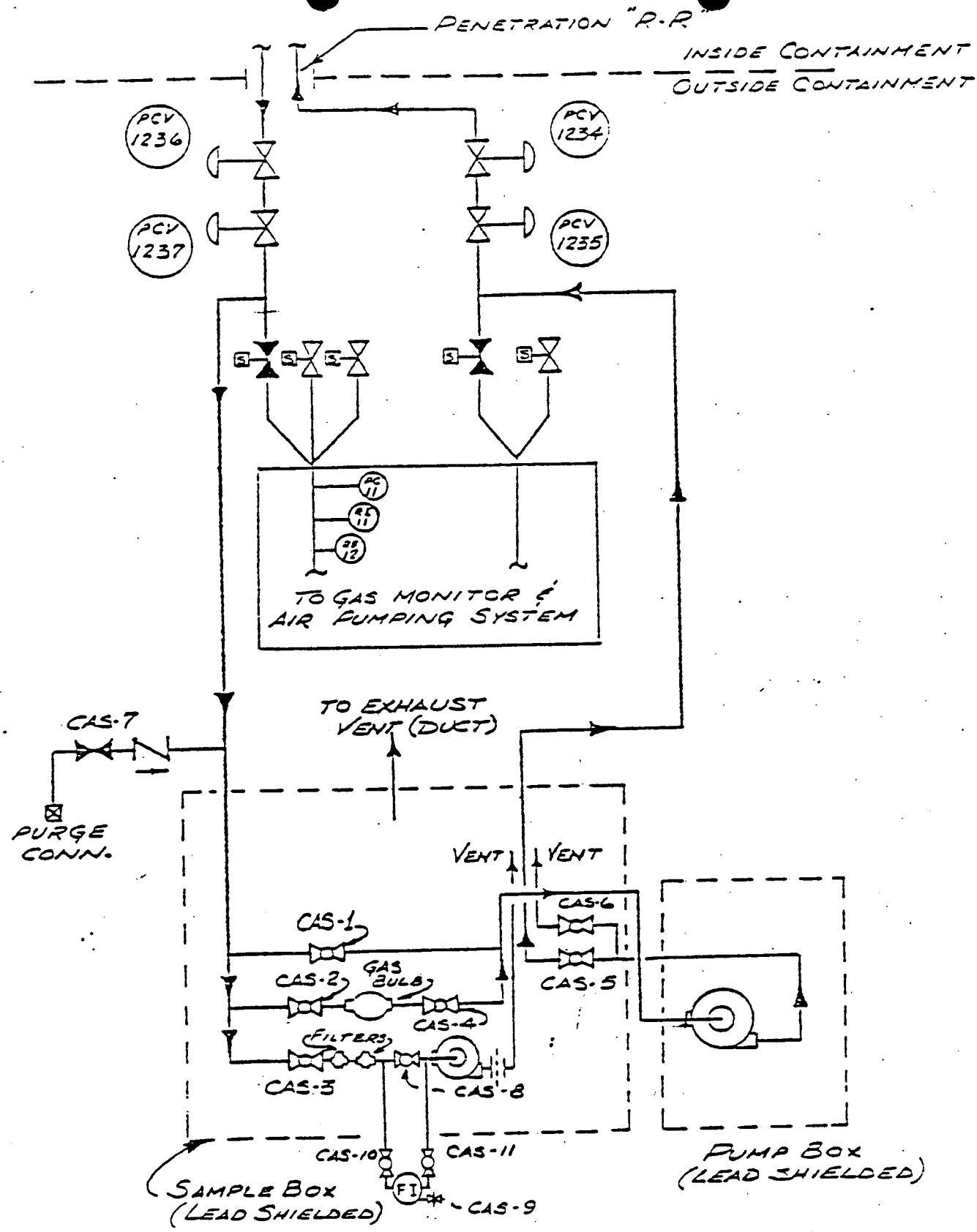


RCS Sample Collection Station

ATTACHMENT 1C



RSC Analysis Station Flow Diagram



Post-Accident Flow Diagram

1.1 References

1.1.1 ND-6620 Installation and Maintenance Manual; Nuclear Data, Inc: July 1976.

1.2 Hardware

1.2.1 The system Central Processing Unit is a Digital Equipment Corporation LSI-11 microcomputer, attached to the system Memory Management Circuit Board.

1.2.2 The system memory is solid state and expandable in 64 kilobyte increments. It currently contains 85 K (word) memory capacity.

1.2.3 The DAS is a programmable subsystem of the 6620 including and LSI-11 microcomputer identical to and interchangeable with the system CPU. The DAS performs all real-time acquisition and display functions independently of the system CPU. (One DAS is associated with one display terminal.) Each DAS may control up to 8 ADC's, although only two are currently connected. Alternatively, Fortran programs may be read into and executed from the DAS microcomputer, independently of the system CPU. The system contains two Display and Acquisition Systems, one Decwriter and a Centronics Line Printer.

1.2.4 The COMBUS is a non-dedicated, high-speed, decentralized, sychronous bus which permits direct, bidirectional data communication between system components. This permits, for instance, any keyboard or teletype connected to the system to control any DAS or the CPU. MIDAS, the system software monitor, allows memory and device assignments according to priority levels designated by the system control operator.

1.2.5 The system uses two Pertec removable dual hard disk drives. Each disk contains 9743 sectors, sufficient for about one hundred forty 4096-channel spectra. (All spectra are stored with 30-bit precision.) The disks also contain MIDAS, FORTRAN, and JOBSTREAM programs. The disk drives are identified by front thumbwheel selectors. Setting the selector to "1" identifies that drive as DK0, DK1. Setting the selector to "3" identifies it as DK2, DK3.

1.3 Software

1.3.1 One system master disk must be on line to operate the 6620. This disk contains the MIDAS program (Multiple

1.3 Software (cont'd)

Independent Display and Acquisition System) which is the system software monitor. MIDAS manages memory, interrupts, performs device assignments, etc.

1.3.2 Nuclear Data also supplies several Fortran applications software programs such as APS. PEAK, APS. NID or APS. ENERGY. These programs provide the basic data manipulation required to reduce spectral data acquired in memory through the DAS.

1.3.3 MIDAS allows ND-supplied and user-generated programs to be automatically executed in sequence, either immediately or at a predetermined future time. These program sequences are called "Jobstreams" and can be written and stored on disk via the APS. EDITOR program.

1.3.4 Fortran programs can be written and stored on the disk. The source programs written by the user must be converted into machine-language by the Fortran Compiler before they can be executed. However, debugged machine-language programs can be stored on the disk for immediate execution similarly to the ND-supplied applications software.

1.4 Systematics

1.4.1 The ND-6620 has been designed with a great deal of flexibility in operation, so each user can tailor the system to his own ends. Unfortunately, this makes the 6620 very complicated to operate under strictly manual control. Therefore, jobstreams have been written to support this application.

1.4.2 MIDAS allows user #Ø some special functions, such as file-element creation or modification, not allowed to user #1-255. User #Ø is known as the "control" operator.

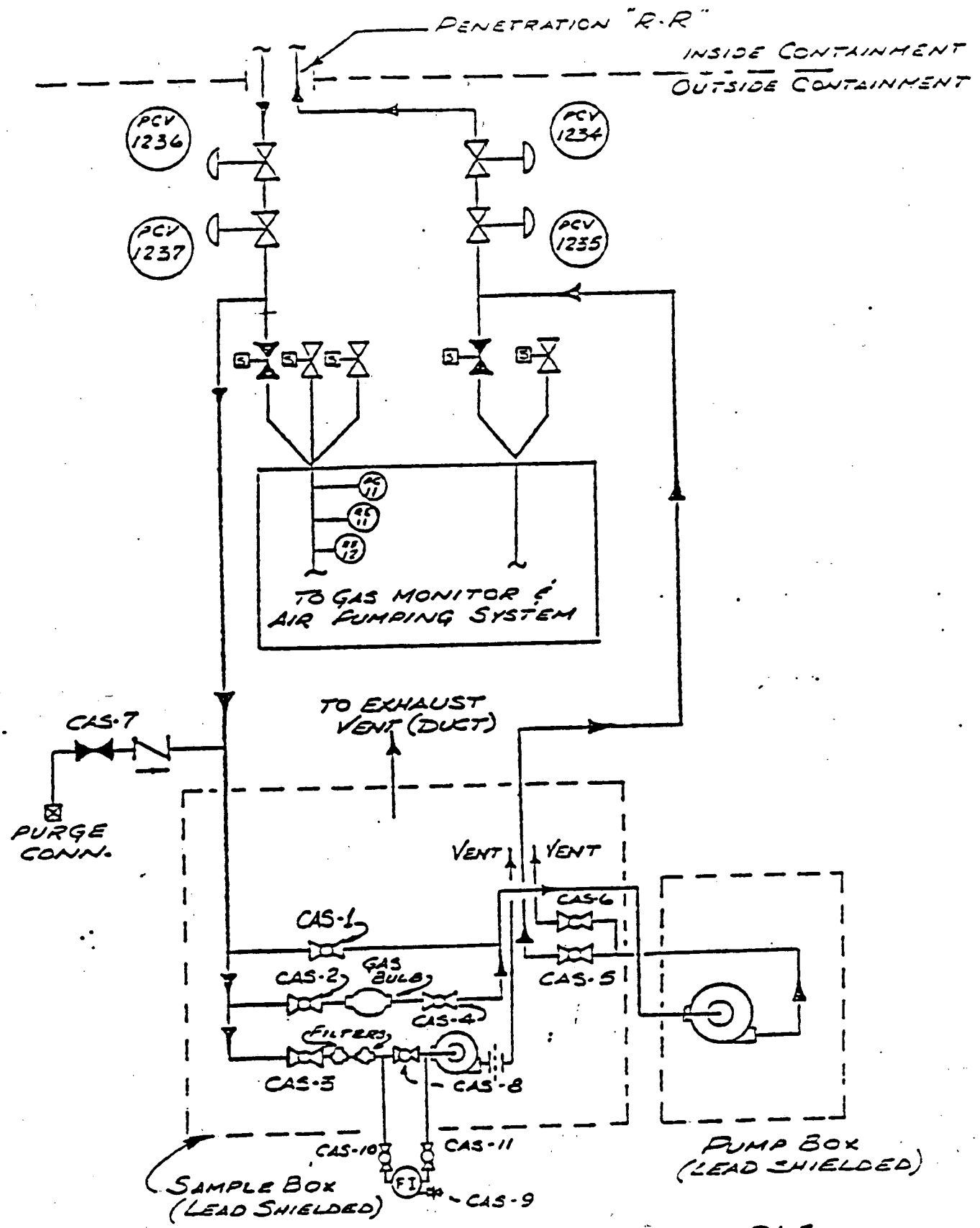
1.4.3 DKØ normally contains the system (program) disk. The disk in DK1 contains files for storing spectral data.

	NUCLIDE	SBHR	HALF	ENERGY	%ABN	BETA	GAMMA
1.	AR-41	AG	1.83H	1293.64	99.2*	4.025E-01	1.283E 00
2.	XE-131M	FG	11.90D	163.93	2.0*	1.420E-01	1.980E-02
3.	XE-133	FG	5.25D	81.00	37.1*	1.500E-01	4.600E-02
4.	XE-133M	FG	2.19D	233.18	10.3*	1.880E-01	4.150E-02
5.	XE-135	FG	9.09H	249.79	90.1*	3.130E-01	2.460E-01
				608.18	2.9		
6.	XE-135M	FG	15.29M	526.57	81.2*	8.100E-02	4.310E-01
7.	XE-138	FG	14.17M	153.75	5.9	5.794E-01	1.096E 00
				242.56	3.5		
				258.31	31.5*		
				396.43	6.3		
				401.36	2.2		
				434.49	20.3		
				1768.26	16.7		
8.	KR-85M	FG	4.48H	151.18	75.5*	2.420E-01	1.570E-01
				304.87	14.0		
9.	KR-85	FG	10.72Y	513.99	0.4*	2.280E-01	2.230E-03
10.	KR-87	FG	76.30M	402.58	49.5*	1.040E 00	7.830E-01
				1740.52	2.0		
11.	KR-88	FG	2.84H	196.32	26.0*	3.310E-01	1.940E 00
				834.83	13.0		
				1529.77	10.9		
12.	NA-24	AP	15.00H	1368.53	100.0*	4.633E-01	4.119E 00
13.	NP-239	FP	2.36D	103.76	22.1	2.608E-01	1.686E-01
				106.13	22.7*		
14.	CR-51	AP	27.70D	320.08	9.8*	3.610E-03	3.249E-02
15.	W-187	AP	23.90H	479.50	26.0*	0.000E-01	0.000E-01
				685.70	32.0		
16.	MN-54	AP	312.50D	834.83	99.9*	3.816E-03	8.360E-01
17.	MN-56	AP	2.57H	846.75	98.9*	6.686E-01	1.689E 00
				1810.72	27.2		
18.	MO-99	AP	66.00H	140.51	5.7	3.600E-01	1.486E-01
				181.07	6.5		
				739.48	13.0*		
19.	TC-99M	FP	6.02H	140.51	89.0*	1.569E-02	1.264E-01
20.	FE-59	AP	44.60D	1099.22	56.1*	1.270E-01	1.180E 00
				1291.56	43.6		
21.	CO-57	FP	270.90D	122.06	85.6*	1.053E-02	1.250E-01
				136.47	10.6		

22.	CO-58	AP	70.80D	810.75	99.4*	2.729E-02	8.230E-01
				863.94	0.7		
				1674.73	0.5		
23.	CO-60	AP	5.27Y	1173.21	99.9*	1.060E-01	2.505E 00
				1332.46	100.0		
24.	NI-65	AP	2.56H	1115.00	16.0	0.000E-01	0.000E-01
				1481.00	25.0*		
25.	CU-64	AP	12.70H	1345.90	0.5*	4.111E-02	7.842E-03
26.	ZN-65	AP	244.10D	1115.52	50.7*	6.150E-03	5.698E-01
27.	AG-110M	AP	250.80D	657.75	94.6*	7.291E-02	2.750E 00
				884.66	73.0		
				937.48	34.4		
				1384.27	24.7		
				1505.00	13.3		
28.	NB-94	AP	20300.00Y	702.63	100.0	1.570E-01	1.574E 00
				871.10	100.0*		
29.	ZR-95	AP	65.50D	724.00	49.0	1.320E-01	7.250E-01
				756.00	49.0*		
30.	NB-95	FP	35.50D	765.00	100.0*	0.000E-01	0.000E-01
31.	ZR-97	FP	16.90H	747.40	92.0*	6.070E-01	1.783E-01
32.	NB-97	FP	72.10M	657.92	98.3*	4.234E-01	6.516E-01
33.	RB-88	FP	17.80M	898.03	14.4*	1.535E 00	6.552E-01
				1836.00	21.9		
34.	RB-89	FP	15.20M	657.71	10.1	8.166E-01	2.079E 00
				1031.88	59.0*		
				1248.14	43.0		
35.	I-131	FP	8.04D	284.30	6.1	2.014E-01	3.806E-01
				364.48	81.2*		
				502.99	0.3		
				636.97	7.3		
				722.89	1.8		
36.	I-132	FP	2.30H	522.65	16.1	4.400E-01	3.189E 00
				630.22	13.7		
				667.69	98.7*		
				772.60	76.2		
				954.55	18.1		
37.	I-133	FP	20.80H	529.89	87.3*	3.800E-01	6.079E-01
				856.36	1.2		
				875.37	4.4		
				1236.56	1.4		
				1298.33	2.3		
38.	I-134	FP	52.60M	595.36	11.4	5.500E-01	2.601E 00
				621.79	10.6		
				847.02	95.4		
				884.00	65.3*		
				1072.55	15.3		

39.	I-135	FP	6.61H	288.45	3.1	5.221E-01	1.558E 00
				417.63	3.5		
				546.55	7.1		
				1038.80	7.9		
				1131.51	22.5		
				1260.41	28.6*		
				1678.03	9.5		
40.	SR-85	FP	64.85D	513.99	98.0*	4.940E-03	5.117E-01
41.	SR-91	FP	9.67H	555.57	61.5	5.330E-01	7.410E-01
				645.00	15.0		
				748.00	27.0		
				1025.00	30.0*		
42.	SR-92	FP	2.71H	1370.00	90.0*	2.150E-01	1.260E 00
43.	Y-88	FP	107.00D	898.02	93.4	5.640E-03	2.703E 00
				1836.04	99.4*		
44.	Y-91M	FP	58.80M	551.00	95.0*	2.700E-02	3.100E-01
45.	Y-92	FP	3.53H	934.00	14.0*	0.000E-01	2.950E-01
				1400.00	4.7		
46.	RH-106	FP	367.00D	621.80	9.8*	0.000E-01	0.000E-01
				1050.10	1.4		
47.	RU-103	FP	39.35D	497.08	86.4*	8.420E-02	4.680E-01
48.	CS-134	FP	2.06Y	604.79	97.6	1.704E-01	1.555E 00
				795.84	85.4*		
				1365.15	3.0		
49.	CS-136	FP	13.10D	86.29	6.3	1.683E-01	2.164E 00
				340.57	46.8		
				818.50	100.0		
				1048.07	79.7*		
				1235.34	19.8		
50.	CS-137	FP	30.00Y	661.64	89.9*	6.218E-01	5.964E-01
51.	CS-138	FP	32.20M	462.79	30.7	1.880E 00	2.334E 00
				1009.78	29.8		
				1435.86	76.3*		
52.	BA-133	AP	10.50Y	302.85	18.7	0.000E-01	5.035E-01
				356.00	62.4*		
53.	BA-140	FP	12.80D	305.00	6.0	2.830E-01	2.410E-01
				537.00	34.0*		
54.	LA-140	FP	40.22H	328.80	20.0	4.360E-01	2.120E 00
				487.00	40.0		
				815.00	19.0		
				1596.00	96.0*		
55.	CE-144	FP	284.00D	134.00	11.0*	1.020E-01	2.070E-02
56.	CD-109	FP	453.00D	88.04	3.7*	0.000E-01	1.074E-01
57.	CE-139	FP	137.65D	165.85	80.1*	0.000E-01	1.940E-01

58.	HG-203	AP	46.59D	279.19	81.5*	0.000E-01	0.000E-01
59.	SN-113	FP	114.90D	255.11	2.1	0.000E-01	2.370E-02
				391.40	100.0*		
60.	K-40	NP	1000.00Y	1460.85	11.0*	0.000E-01	0.000E-01
61.	SB-122	AP	2.70D	564.00	71.0*	4.963E-01	4.374E-01
62.	SB-124	FP	60.20D	722.78	11.3	3.530E-01	1.825E 00
				1691.02	48.8*		
63.	SB-125	FP	2.71Y	427.00	31.0*	0.000E-01	0.000E-01
				634.00	11.0		
64.	EU-152	AP	12.70Y	344.00	27.0*	0.000E-01	0.000E-01
				1408.00	22.0		
65.	RA-226	NP	1602.00Y	295.20	20.1	0.000E-01	0.000E-01
				351.96	39.3		
				609.19	48.3*		
				1120.30	16.0		
				1764.50	16.6		
66.	U-235	NP	2000.00Y	143.00	11.0*	0.000E-01	0.000E-01
				185.00	54.0		
				204.31	5.0		
67.	CE-141	FP	32.50D	145.44	48.0*	3.640E-01	7.621E-02
68.	XE-127	FG	36.41D	57.60	1.3	2.990E-02	2.806E-01
				145.22	4.1		
				172.10	25.4		
				202.84	68.0		
				374.96	17.7*		



CONTAINMENT AIR SAMPLE

Post-Accident Flow Diagram