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August 31, 1982

Director of Nuclear Reactor Regulations
United States Nuclear Regulatory Commission
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
NUREG-0737, Item II.B.3; Post Accident Sampling System
Post Implementation Review

Gentlemen:

In accordance with our letter of July 30, 1982, and your letter of June 30, 1982, we are providing the information requested in order that the Staff may conduct a post-implementation review of our Post Accident Sampling System. We have responded to each item of Attachment No. 1 as contained in your letter thereby making this a complete response.

As indicated in our July 30 letter, we have identified several apparently new requirements contained in Attachment No. 1 of your letter which did not appear in the previously documented design requirements referenced by NUREG-0737 item II.B.3. Where our system has satisified these new requirements, we have provided the requested information. Where our design does not address these new requirements, we have provided a statement representing our position with regard to the new requirement.

Attachment A is provided for your review to complete the post implementation review requirements as documented in NUREG-0737. If you have any questions on this subject, please contact my office.

Very truly yours,

J. J. Carey
Vice President, Nuclear

Attachments

cc: Mr. W. M. Troskoski, Resident Inspector
U. S. Nuclear Regulatory Commission
Beaver Valley Power Station
Shippingport, PA 15077

U. S. Nuclear Regulatory Commission
c/o Document Management Branch
Washington, DC 20555

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DUQUESNE LIGHT COMPANY
Beaver Valley Power Station, Unit No. 1

Post Accident Sampling System
Response to NRC letter dated June 30, 1982

Attachment A

Note: The information contained in Attachment A is organized as follows:

- Criterion from NUREG-0737, item II.B.3
- Clarification of criterion provided by NRC (letter of June 30, 1982)
- Response

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

The Post Accident Sampling System (PASS) is located on the same elevation as the chemistry hot laboratory and approximately 200 ft. away. Refer to Figure 1. After a sample has been collected, it will be transported by the PASS operator to the counting room along the path indicated on Figure 1. It has been determined that the radiation levels of a diluted sample are sufficiently low to permit handling of the sample utilizing established techniques.

The PASS installation makes use of the existing reactor coolant sample piping. This piping is normally in continuous recirculation to the Volume Control Tank making representative samples available essentially instantaneously. Should a containment isolation signal stop the recirculation by closing the containment isolation valves,

approximately 15 minutes is required for re-establishment of recirculation from the time the sample leaves the reactor coolant system to the time it flows into the PASS. During accident conditions, the PASS will direct the recirculation flow from the Volume Control Tank to the containment sump. A recirculation path for containment atmosphere sampling can be set up at the PASS by taking suction from the containment dome at the inlet to the containment vacuum pump, pumping the containment atmosphere through the PASS containment sample piping, and returning the exhaust to the containment via the discharge of the containment atmosphere radiation monitor RM-215. A sample which is representative of the containment atmosphere can be obtained at the PASS within 15 minutes.

In-line instrumentation for determining the levels of boron, chlorides, dissolved hydrogen, dissolved oxygen and pH in the reactor coolant system or liquid in the containment sump are included in the PASS design. The PASS also has the capability to obtain diluted or undiluted samples of the containment atmosphere, containment sump or the reactor coolant system for isotopic analysis.

After the decision has been made to take samples, the estimated time for obtaining the samples and performing the necessary analysis is as follows:

1.	Pre-sample briefing, Anti-C dressing	30 minutes
2.	Travel to the PASS	1 minute
3.	Establishing liquid and containment atmosphere recirculation, reading of in-line instrumentation and obtaining a diluted sample	45 minutes
4.	Transport of the diluted samples to the hot laboratory	2 minutes
5.	Removal of Anti-C clothing and frisking	30 minutes
6.	Isotopic counting of samples	10 minutes
7.	Data reduction	20 minutes

Total Estimated Time: 2 hours 18 minutes

In the event the chemistry hot laboratory would be unavailable or inaccessible, heavily shielded backup laboratory and counting facilities are maintained in the Emergency Response Facility (ERF). The ERF is located approximately 1200 feet from the plant, on the same approximate elevation and accessible from the plant by way of a company owned roadway through the Unit No. 2 facility. To utilize this laboratory and counting room a diluted sample would be taken to the ERF via a company vehicle. The sample would then receive an isotopic analysis. The time for transporting the sample has been determined to be less than 15 minutes. When factored into the above time table, it can be demonstrated that the criteria of NUREG-0737 has been satisfied. Refer to Figure 2 for a plan view of the ERF and its relationship to the Beaver Valley Power Station, Unit No. 1.

With regard to having provisions for sampling during a loss of off-site power, this provision is not a design requirement and as such was not considered in the design of the PASS. Additionally, the station laboratory counting room originally was not designed to have an alternate backup power source. Sample collection and analysis will not be possible in the event of a loss of off-site power. However, the ERF will have a back-up power supply in the form of a diesel generator. As such the laboratory and counting room in the ERF would be operable in the event of a loss of off-site power.

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 in-line 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 Xe-133, I-131, Cs-137 Cs-134, Kr-85, Ba-140, and Kr-88 (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

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

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

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

RESPONSE

2(a) The PASS is capable of performing the following:

- obtaining a grab sample of the containment atmosphere using a gas sample capsule. Nitrogen gas may be metered in for dilution purposes as necessary for isotopic counting.

- obtaining a liquid grab sample from the reactor coolant system or the containment sump and performing the line monitoring as discussed below. Samples may be diluted at the PASS to a ratio of 10,000 to 1.

Isotopic analysis of grab samples is performed in the Chemistry hot laboratory utilizing a Germanium detector connected to a computer based multichannel analyzer. The Germanium detector is housed in a 4-inch lead cave lined with cadmium and copper and the cave sits in a Faraday shielded room. Procedures to relate specific nuclides to core damage have been prepared. Development of these procedures was accomplished using the Rogovin Report as a reference.

Isotopic analysis may also be performed on a Germanium detector connected to a computor based multichannel analyzer in the ERF. The design of the laboratory and counting room include concrete walls, twenty inches thick for shielding purposes. We are presently considering the purchase of a cave to further assure the counting accuracy of our detector. Additional dilutions for isotopic counting may also be performed in the ERF laboratory.

2(b) Hydrogen levels in the containment atmosphere is determined utilizing equipment installed as a result of NUREG-0737 item II.F.1.6. Redundant hydrogen monitors obtain samples from the containment atmosphere and the pressurizer cubicle. The sampling lines tap off of the existing lines of the containment vacuum pumps which extend to the high point inside containment. Readout is provided in the control room over the range of 0 - 10% hydrogen concentration.

2(c) Hydrogen in the liquid sample is separated in a phase separator located inside the PASS. Hydrogen and other gases pass through in-line hydrogen and oxygen detectors before being vented through the gas sample capsule and returned to the containment atmosphere. If the in-line detectors fail, the gas sample capsule can capture approximately 12 cc of gas. If radiation levels as read on the in-line gross gamma detectors warrant, the gas may be diluted with known amounts of nitrogen. After sufficient dilution to lower the radiation level of the capsule, the capsule can be removed to the hot laboratory. The gas can then be recirculated through a gas chromatograph or a thermal conductivity detector to analyze for the amount of hydrogen present.

The boron, chloride and pH levels will be analyzed in-line by microprocessor controlled electrodes. Should the electrode system fail, there is a backup in-line pH electrode installed. Additionally, a diluted liquid sample would be taken to the laboratory for a boron analysis using Carminic Acid and an agreement has been made with the Bettis Atomic Power Laboratory to perform a chloride analysis on an undiluted sample.

Regulatory Guide 1.97 Rev. 2, includes accident sample species which are beyond the design scope of our Post Accident Sampling System. Our system was designed to accomplish sampling and analysis as required by NUREG-0737. Evaluation of this Regulatory Guide will be performed as a separate issue, at a later time and with consideration of the recent Staff position with regard to SECY 82-111.

2(d) In-line monitoring capabilities have been utilized to the extent documented above. The boron and chloride microprocessor controlled electrodes were specifically designed for use in post-accident applications. This equipment has been developed for our facility and, as such, does not have a previous operational history. These components were manufactured by ORION Research and are identified by the following model numbers:

- boron analyzer; Model 1610
- chloride analyzer; Model 1617

The equipment utilized to determine dissolved hydrogen and oxygen were manufactured by Exosensors, Inc. We have been informed by Exosensor, Inc. that equipment similar to what has been installed in the PASS has successfully passed rigorous post-LOCA environment testing. They have been producing equipment of this type for two years and have not experienced reliability or maintenance problems. These components are identified by the following model number:

- dissolved hydrogen and oxygen analyzer; Model 791-115, Rev. A

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

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.

RESPONSE

The PASS is an independent system designed to be used in a post-accident environment. Operation is not dependent on the availability of any auxiliary system which may be isolated during accident conditions. The only auxiliary system which interfaces with the PASS is the Reactor Plant Component Cooling Water System. If for some reason this system is unavailable, backup is provided by the river water system. Both systems are capable of being powered from the station emergency diesel generators.

Remote-operated valves are environmentally qualified for the conditions in which they must operate. These are the containment isolation valves, containment sump suction and return valves and the reactor coolant loop sample valves.

Included with this submittal are 2 copies of the following system schematics:

- Operating Manual Figure No. 12-1
- Operating Manual Figure No. 14-1
- Operating Manual Figure No. 14-4

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 hydrogen gas in reactor coolant samples is considered adequate. Measuring the oxygen concentration is recommended, but is not mandatory.

CLARIFICATION

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

RESPONSE

The hydrogen gas in the reactor coolant can be measured as described in our response to criterion 2. The concentration of hydrogen gas may then be calculated by multiplying the percent hydrogen found by the total gas flowrate past the analyzer and dividing that figure by the reactor coolant flowrate through the phase separator. Hydrogen concentrations as low as 3cc/kg can be measured in this fashion. Dissolved oxygen is indicated directly in ppm as it is read by the in-line analyzer discussed in criterion 2. On the 0-1 ppm scale, oxygen concentration as low as 0.02 ppm can be measured. Additionally, hydrogen and oxygen concentrations can be measured in the laboratory using a gas chromatograph.

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

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 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

Chloride analysis will be performed utilizing in-line monitoring of undiluted samples in the PASS. As documented in our response to criterion 2, arrangements have been made with the Bettis Atomic Power Laboratory to perform chloride analysis on undiluted samples if it should become necessary. Transportation may be accomplished by either of two methods; helicopter or motor vehicle. Transportation times are estimated at 30 minutes and 90 minutes, respectively.

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

In accordance with NUREG-0737 item II.B.2, Design Review of Plant Shielding, a detailed report identifying plant radiation levels following an accident was prepared and forwarded to the NRC on June 30, 1981. Supplementing this report was additional in house review which resulted in the establishment of access routes to plant areas. These were considerations in the location of new sampling equipment.

All sides of the PASS which are not directly against a concrete shield wall are lined with three inches of lead. Additionally, there are viewing ports installed so the PASS operator can observe the sampling evolution. These viewing ports are constructed of five inch thick lead glass windows which is equivalent to three inches of solid lead. The design of the PASS box is to limit the external exposure rate to 500 mR per hr. from sources within the box.

The following is the predicted man rem exposures for sampling, transport and analysis. Exposure to an individual while obtaining the sample is estimated at 412 mR based on an exposure rate of 500 mR per hour from the PASS and 50 mR per hour from other sources in the plant. Transport to the chemistry hot laboratory or to the ERF will result in a minimal exposure of only a few mR. An unshielded, diluted sample is expected to have an exposure rate of less than 500 mR per hour on contact. While transporting the sample to the ERF counting room, maximizing the distance between the sample and personnel will significantly reduce the exposure rate. An exposure from the sample while positioning and counting is considered to involve only a brief exposure to the individual who is to position the sample, start the counting equipment, then remove the sample to a shielded box. This event should not generate a personnel exposure

greater than 10 mR. The location of the sample analysis, either the chemistry hot laboratory or the ERF counting room, will result in different background readings which will contribute to an individuals exposure. The exposure rate due to background sources at the hot laboratory at one hour following an accident is estimated to be 601 mR per hour. Sample positioning and counting may require 15 minutes and result in an exposure of 150 mR. If the counting is performed in the ERF laboratory, the background contribution will be negligible.

The final disposition of the sample is planned as its return to the PASS box for venting, storage or return to containment. This should not result in an exposure greater than 45 mR.

In summary, the processing of a single sample may contribute a personnel dose of approximately .64 man-rem.

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

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.

RESPONSE

The PASS will be capable of determining boron concentrations in either the reactor coolant system or the containment sump utilizing the in-line instrumentation as previously discussed.

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

The PASS has the capability to obtain both diluted and undiluted samples and provisions for obtaining a backup sample have been discussed. Additionally, provisions have been included in the design of the PASS to flush in-line monitors with demineralized water and/or decon solutions. As stated in our response to criterion 2, the Bettis Atomic Power Laboratory will perform chloride analysis of undiluted samples if the need arises. Bettis is located approximately 40 air miles away and transporting the samples would be accomplished using a Department of Defense vehicle.

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 uCi/g to 10 Ci/g.

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

CLARIFICATION

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 effect will be on a sample being counted to assure an accuracy within a factor of 2.

RESPONSE

a) As previously discussed, an isotopic analysis will be performed using a Germanium detector connected to a computor based multichannel analyzer. This will provide the required isotopic breakdown. The capability to dilute worst case reactor coolant or containment air samples to dose rates below 500 mR per hr. is within the design capability of the PASS.

The predicted activity of our core has been determined to be 16 Ci per gram. This is based on data contained in the Rasmussen Report and applied to Beaver Valley, Unit No. 1. The PASS can dilute liquid samples up to a ratio of 10,000 to 1 and additional

dilution may be performed in the laboratory. The sensitivity of our equipment to analyze a liquid sample is in agreement with your criteria contained in NUREG-0737. Due to potential high radiation fields, we would not employ normal sampling techniques when obtaining a post accident sample. Therefore, the amount of overlap between post accident and normal sampling capabilities is inconsequential for our application.

b) The shielding due to the construction of the ERF counting room and the shielding around the detector in the chemistry hot laboratory have been previously discussed. The contribution to the radiation level in either counting room from the sample has been estimated at 500 mR per hr. The background radiation level in the chemistry hot laboratory at one hour and at two hours has been estimated at 601 mR per hr. and 4.2 mR per hr, respectively. Based on an analysis of generic calculations performed for similar facilities at other sites, and on calculations performed at the site for other purposes, and normalized to the ERF purpose, the dose rate in the ERF counting room following an accident would be such that counting equipment will be operational with a sufficient accuracy, even in the first three hour period following the accident. With due consideration to the conservatisms regarding source term magnitude, radiation transport and release timing, the dose rate is likely to be significantly less.

Background readings will be taken in either counting area just prior to beginning analysis of the sample, then subtracting this reading from the counting results. This should result in an analysis with results within a factor of 2 as requested. Through consideration of the above, there is reasonable assurance that the required radiological and chemical sample analysis capability can be performed as required within the design envelope of NUREG-0737.

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

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

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

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

- 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.

- 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.

- Oxygen: monitored to assess 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.

- 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 Concentration (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	150	
NH3/4	5	
K+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Absorbed 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.

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 procedure has 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 receieve initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six 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.

RESPONSE

The following represents the range and accuracy of those instruments utilized for performing normal analyses of samples collected by the PASS.

Parameter	Range	Accuracy
Gamma Spectrum	up to 10 Ci/g (with dilution)	±50%
Boron	0-600 ppm and 0-6000 ppm	±10% of full scale
Chloride	0-100 ppm	±10% of full scale
Dissolved Hydrogen	0-10% and 0-100%	±6% of full scale
Dissolved Oxygen	0-200 ppm	±6% of full scale
pH	1-14 pH units	±0.1 pH unit

All instruments were purchased with certification that they would function in radiation fields exceeding 10 E4 RADS/gram of reactor coolant. Testing involving the standard chemical test matrix will be concluded within 4-6 weeks of receiving the necessary chemicals.

Each PASS operator will receive training on the PASS with annual retraining. The PASS will be utilized for sampling during normal operations, thereby maintaining a high degree of proficiency in its operation. Due to the difficulty of removing instruments for calibration, which would remove the PASS for extended periods of time, instrument calibration will be scheduled on a 24 month frequency.

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 absorbers and high-efficiency particulate air (HEPA) filters.

CLARIFICATION

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.

   BWR's 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 absorbers and HEPA filters.

RESPONSE

11(a) The location of the PASS was determined by considering the location of existing sample lines and pre-existing structures in the plant. New sample lines which were installed tap into existing lines and were routed to the PASS along the most desirable path. The Operating Manual Figures referenced in response to Criterion 3 identify the existing sample lines and the new lines utilized for the PASS.

Sample line purge is accomplished by the recirculation described in response to Criterion 1. All gas sample lines are heat traced where required to minimize plateout, all sample lines have minimum length to minimize sample loss and the consequences of a sample line rupture, and all sample waste is directed back to containment. The sample lines have been sized such that leakage from the reactor coolant system resulting from a failed sample line will be within the make-up capability of a single charging pump. The PASS was designed to produce a linear velocity of approximately 4-6 feet per second for water and 55-65 feet per second for air.

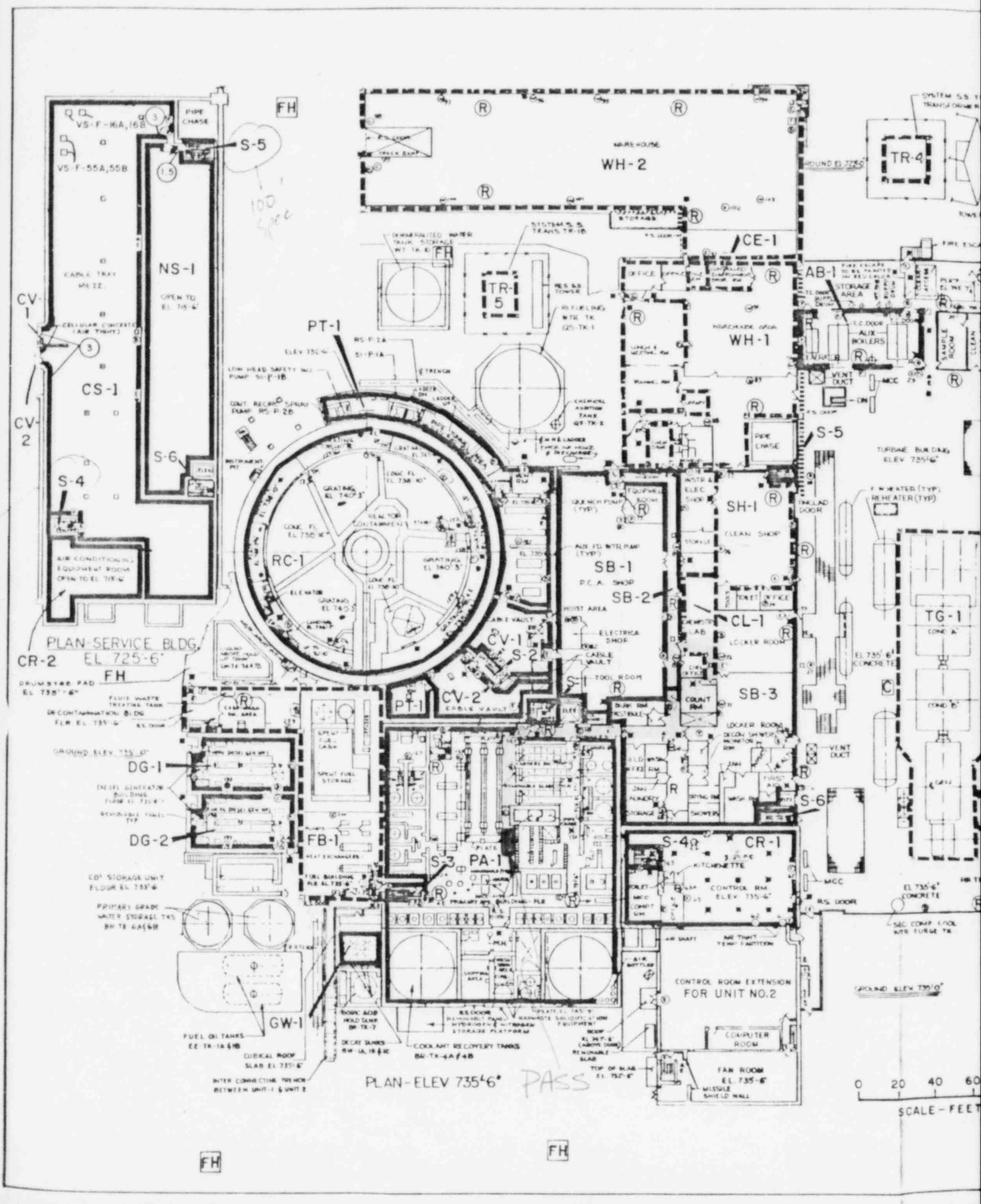
The normal reactor coolant system sample point is on the B loop hot leg, between the reactor vessel and the reactor coolant isolation valve. Due to being located on the hot leg of the reactor vessel, a sample which is representative of the reactor core will be obtained. Should a steam or gas pocket develop and render this sample point inoperative, procedures do exist which will permit the removal of such a pocket. The time this condition can exist is dependant upon the size of the void and the length of time required to vent or otherwise remove this void, and as such, it is not possible to predict the length of time this condition may exist. By procedure, the venting of the reactor coolant system requires close monitoring of coolant system and containment parameters which may preclude immediate, complete removal of the void in order to maintain the most desirable plant conditions. Sample points from the other loops and the cold legs may be valved into service if radiation fields at the normal sample panel permit.

A representative sample of the atmosphere in containment is taken from the dome area of containment. It is highly unlikely that these lines would become plugged or inoperable following a transient or accident. Samples collected from the containment sump are afforded protection from debris by screens which stop the movement of objects from getting into the containment sump.

The PASS has been designed making maximum use of existing sample lines equipped with new containment isolation valves. The containment isolation valves are environmentally qualified and will automatically close on a CIA signal.

11(b) A positive exhaust system exists for the PASS, exhausting through dedicated charcoal filters and HEPA filters before entering the supplementary leak collection and exhaust system. This system also exhausts through HEPA and charcoal filters.

Figure 1



PLAN-SERVICE BLDG EL 725'-6"

PLAN-ELEV 735'-6"

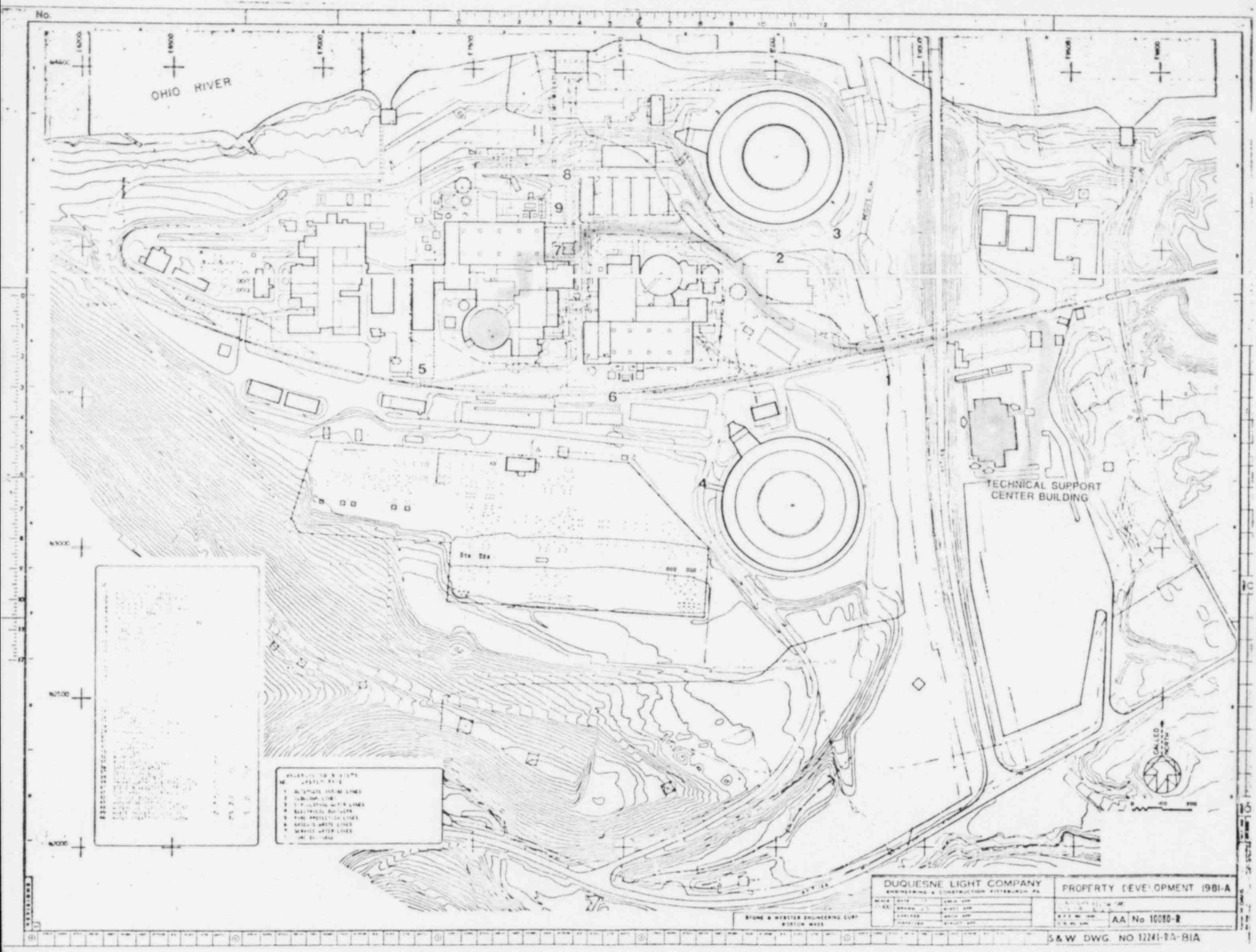

OHIO RIVER
TECHNICAL SUPPORT CENTER BUILDING
DUQUESNE LIGHT COMPANY
PROPERTY DEVELOPMENT 1981-A
AA No 10080-B
S&W DWG NO 12241-RA-81A

Figure 2