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MULLER, D. R. BWR Project Directorate 2

SUBJECT: Submits responses which address 10 out of 11 criteria of NUREG-0737, Item II.B.3 for redesigned postaccident sampling sys. Response to Criterion 6 currently being developed & will be submitted by 870331.

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NMSS/FCAF 1cy. NMSS/FCAF/PM 1cy. OIA 1cy. GROTENHUIS, M 1cy. 05000296

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DEC 19 1986

Director of Nuclear Reactor Regulation
Attention: Mr. D. R. Muller, Project Director
BWR Project Directorate No. 2
Division of Boiling Water Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Muller:

In the Matter of the	)	Docket	50-259
Tennessee Valley Authority	)		50-260
			50-296

BROWNS FERRY NUCLEAR PLANT (BFN) - NUREG-0737, ITEM II.B.3 - POSTACCIDENT SAMPLING SYSTEM

As indicated in my letter to you dated May 19, 1986, we have reevaluated our plans for implementing the Postaccident Sampling System (PASS) as required by NUREG-0737, Item II B.3 at BFN. This evaluation identified several practical problems with the original system plan and we concluded that regulatory requirements and a timely implementation schedule could best be met by design and procurement of a different system.

The new PASS design features a separate grab sample panel for each unit. Chemical analyses will be performed on diluted or small grab samples in the Postaccident Chemical Laboratory (PACL). The enclosed submittal, therefore, supercedes a number of aspects of the proposed PASS as outlined in previous submittals to NRC (Reference the following letters: L. M. Mills to H. R. Denton dated November 16, 1982; J. A. Domer to H. R. Denton dated August 13, 1984). Several aspects of the design have not significantly changed from those previously reviewed by the NRC. When this is the case, reference is made to the NRC documentation of this previous approval.

The enclosed submittal addresses 10 of the 11 criteria of NUREG-0737, Item II.B.3 (as clarified in NRC's letter from D. B. Vassallo to H. G. Parris dated July 13, 1982) for the redesigned PASS. Our response to criterion 6 is currently being developed and will be submitted by March 31, 1987.

If additional information is required, please get in touch with L. V. Tonty at (205) 729-2677.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. L. Gridley, Director
Nuclear Safety and Licensing

cc: See page 2

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Director of Nuclear Reactor Regulation DEC 19 1986

Enclosure
cc (Enclosure):
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ENCLOSURE

BROWNS FERRY POSITION IN REFERENCE TO NRC

"CLARIFICATION ON NUREG-0737, ITEM II.B.3 (POST ACCIDENT SAMPLING)"

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 three 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 offsite 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 individual Postaccident Sampling System (PASS) panels for units 1, 2, and 3, and the common Postaccident Chemical Laboratory (PACL) will be located outside the secondary containment on elevation 565 of the turbine building. Each PASS consists of a sample station and a control panel. Approximate distances from the PASS to the PACL are 300 feet for unit 1, 200 feet for unit 2, and 125 feet for unit 3. The PACL, serving all three units, contains the equipment for chemical and radiochemical analyses as required in NUREG-0737, Item II.B.3.

The following samples can be obtained from the General Electric (GE) supplied PASS:

a. Diluted (100:1) liquid reactor coolant grab samples (10 ml)

b. Undiluted degassed liquid reactor coolant samples (0.2 ml and 10 ml)

c. Undiluted dissolved gas samples from the reactor coolant (1 - 5 cc)

d. Undiluted containment atmosphere samples (14.1 cc)

The PASS has the capability to purge the sample lines before sampling to ensure representative samples are obtained. Sample lines from the PASS may also be flushed after the sampling operations are complete to reduce residual radioactivity. Samples are placed in shielded casks for transport to the PACL. The casks are mounted on wheeled carts and are separated from the analyst by approximately three feet during transport.

Listed below are the estimated sample acquisition and analysis times in minutes:

		Sampling*	Analysis**	Total
a.	Diluted (100:1) reactor coolant (RC)	15	45	60
b.	Undiluted degassed RC sample			
	0.2 ml	15	3	18
	10.0 ml	16	***	***
c.	Undiluted dissolved gas sample	25	20	45
d.	Undiluted containment atmosphere sample	10	45	55

Notes:

* Sampling time includes line flushing requirements and sample transport.

** Analysis times are estimated. Specific procedures for postaccident analyses are being developed.

*** Analysis for chloride required within 30 days (refer to clarification of criterion 5).

The redesigned PASS uses two separate sources of offsite power to supply all PASS requirements. This aspect of the design has not changed from the original plan and was previously accepted by NRC in a letter from D. R. Muller to S. A. White dated February 21, 1986.

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three-hour timeframe 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, ^{137}Cs, ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (see vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

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 online instrument is appropriate for this application. (See [8] and [10] below relative to backup grab sample capability and instrument range and accuracy).

RESPONSE

2 (a) Sampling and analysis times are given in the response to Criterion 1. Provisions for dilution of the samples and for quantifying radionuclide concentrations are referenced in the response to Criterion 9. The BFN core damage assessment procedure (Technical Instruction (TI) - 88) was previously reviewed and accepted by NRC in a letter from D. R. Muller to S. A. White dated February 21, 1986. This procedure will be used with other data to estimate the extent of any core damage.

2 (b) Separate hydrogen analyzers were previously installed to fulfill NUREG-0737, Item II.F.1.6, and will also meet the Item II.B.3 requirement to analyze for containment hydrogen. In addition, the BFN PASS has the capability to obtain undiluted gas samples from both the drywell and suppression pool atmospheres. These samples can then be analyzed in the PACL for hydrogen content .

2 (c) The majority of the chemical analyses are performed in the PACL. The capabilities are as stated below:

Analysis Performed	Range
Chloride (P)	0.05 - 20 ppm
Boron (P)	50 - 1000 ppm
Total Dissolved Gas (I)	25 - 400 cc/kg
Dissolved Oxygen (P)	4 - 20 ppm
pH (P)	1 - 13 units

(P) denotes PACL instrumentation
(I) denotes inline instrumentation

Isotopic analysis in the range of 1 uCi/ml - 10 Ci/ml will be performed on diluted grab samples in the PACL.

The PASS capability to perform the analysis listed here fulfill the requirements of Criterion 2 as well as those of Regulatory Guide 1.97, Rev. 2.

2 (d) The PASS uses an inline pressure indicator to determine total dissolved gases. The procedure, range, and accuracy for monitoring total dissolved gases by this method has been reviewed and accepted by NRC (reference letter from William V. Johnston (NRC) to Glenn G. Sherwood (GE) dated July 17, 1984).

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

When the reactor is pressurized, coolant samples are taken from the jet pump instrument piping. This operation does not require opening of any primary containment isolation valves. When the reactor is not pressurized, suppression pool samples can be obtained from Residual Heat Removal (RHR) heat exchanger 'C' when RHR is operating in suppression pool cooling mode. This operation requires opening of two primary containment isolation valves which will be environmentally qualified. This same sample point can be used to collect reactor coolant samples when the reactor is depressurized and the RHR system is operating in shutdown cooling mode. Placing shutdown cooling mode in operation will require opening of the shutdown cooling isolation valves which are not environmentally qualified and are inaccessible in a postaccident environment.

Containment atmosphere samples are taken from the drywell and suppression chamber through the existing isolation valves used for the H_2/O_2 monitoring system. These valves are environmentally qualified.

Criterion: (4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification: 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 $\geq$ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

RESPONSE

PASS uses inline instrumentation to determine the amount of total dissolved gasses present in the liquid reactor coolant samples. As a backup to inline instrumentation, samples of gas stripped from the reactor coolant can be obtained and analyzed for hydrogen and oxygen concentration by gas chromatography.

If the chloride concentration of the reactor coolant exceeds 0.15 ppm, the dissolved oxygen can be verified to be less than 0.1 ppm.

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 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

The two factors, (a) and (b), in Criterion 5 do not apply to BFN; therefore, the chloride analysis will be completed within four days. This analysis will be done in the PACL on a diluted (100:1) reactor coolant sample. An undiluted degassed reactor coolant sample will also be taken and retained for analysis within 30 days.

Criterion: (6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that itis 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 personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

RESPONSE

An analysis is being performed to predict personnel exposures during sampling and analytical operations. It is expected that dose criteria suggested above will be met for the sampling operations. Results of this analysis will be forwarded to NRC by March 31, 1987.

Criterion: (7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 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

Boron analysis can be performed in the PACL if boron is injected into the primary coolant system.

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, 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 uses inline pressure indicators for analysis of the total dissolved gas concentration of the reactor coolant. The monitor is located on the PASS panel, outside secondary containment, and should be accessible for repair. The monitor can be flushed to reduce residual radioactivity to facilitate access. As a backup, grab samples of gas stripped from the reactor coolant can be obtained and analyzed for hydrogen and oxygen concentration. Grab sample analyses will also fulfill the requirements of criterion 4.

Redundant H_2 monitors are available to measure containment atmosphere hydrogen. The PACL also has the capability to analyze containment atmosphere grab samples for hydrogen and oxygen by gas chromatography.

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

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

RESPONSE

9(a) and 9(b) Provisions for dilution and analysis of samples for radionuclide concentrations have not been changed from those presented in TVA's letter dated November 16, 1982 submittal from L. R. Mills to H. R. Denton. In addition, the expected background radioactivity sources and levels in the PACL will be lower than previously estimated, since the sample panels will not be located in the PACL as they would have been using the original design. The referenced provisions were accepted by NRC in a letter from G. C. Lainas to H. G. Parris dated May 1, 1984.

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_3^-	150	
NH_4^+	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4Rad/gm of Reactor Coolant	Absorbed Dose

NOTES:

1) Instrumentation and procedure 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 receive initial and refresher training in postaccident 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 range and accuracy of each analysis is as stated below:

Analysis	Range	Accuracy
Chloride Concentration	0.5-20 ppm	± 10%
	0.05-0.5 ppm	± 0.05 ppm
Boron	50-1000 ppm	± 50 ppm
Total Dissolved Gases	25-50 cc/kg	± 50%
	50-400 cc/kg	± 30%
Dissolved Oxygen	4-8 ppm	± 60%
	8-20 ppm	± 30%
pH Determination	pH 1 - 13	± 0.3 units
Radionuclides - liquid or containment atmosphere	1 uCi/ml - 10 Ci/ml	± 50%

Chloride and boron analysis will be done on an ion chromatograph. BFN will use procedures developed by NUS. These procedures have been verified in postaccident matrix solutions.

The total dissolved gases and dissolved oxygen methods have been reviewed by NRC and were found to be acceptable (reference Response 2d). A method to analyze for dissolved hydrogen is available as a backup to total dissolved gas measurements. The method used should be capable of measuring dissolved hydrogen down to 5 cc/kg.

Determination of pH will be done with a flat-surface combination electrode. This type of electrode performed satisfactorily in high radiation fields (approximately 1.3×10^6 R/hr) in tests performed by General Electric.

For radionuclide analysis, grab samples will be diluted to a total activity of about 1 uCi which is within the normal operating range of a multichannel analyzer (MCA).

The anticipated postaccident radiation levels are not expected to have any measurable effect on the accuracy of measurement and negligible effect on the operating lifetime of components exposed to radiation. These conclusions are based upon information provided by limited testing results, literature reviews, and contacts with experienced personnel engaged in similar analyses under high radiation conditions.

RESPONSES ON CALIBRATION, TESTING, AND TRAINING - PASS

Postmodification tests will be performed to ensure operability of the PASS before its initial service and specific calibration procedures will be established for the PASS instruments. This will ensure that this equipment is operable with a high degree of reliability. The frequency of calibration for these instruments will be established before making the facility operational.

Procedures for functionally testing the PASS and for training of the PASS radiochemical laboratory analysts will be developed and in place by startup of the PASS. A sufficient number of analysts and engineers to operate the PASS will be trained within one month of startup of the PASS. Every six months approximately one-half of the radiochemical laboratory analysts will both operate the PASS and take samples. At the same time, routine samples will be taken from the normal sample locations and compared with the corresponding PASS results. This will verify the PASS is functioning correctly. Please note that by using this timetable, the PASS operators will be retrained on a yearly basis, as a minimum, and the PASS equipment will be tested once every six months. TVA believes this program will provide adequate frequency for training and testing. This provision was previously found to be acceptable by NRC (reference letter from D. R. Muller to S. A. White dated February 21, 1986).

Any required changes to technical specifications will be submitted before operation of the PASS.

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

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

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

11(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) LIQUID SAMPLES

Collection of reactor water samples from the jet pump instrument system was accepted previously by NRC in the letter from D. R. Muller to S. A. White dated February 21, 1986. This sample location has not been changed in the system redesign.

Additionally, a sample line on RHR heat exchanger "C" provides a reactor coolant sample when the reactor is depressurized and RHR Loop 1 is operating in the shutdown cooling mode. Similarly, a suppression pool liquid sample can be obtained when the RHR loop has been lined up in the suppression pool cooling mode.

Strainers or filters are not used in the liquid sample lines. The location of the source of the primary liquid sample point is a jet pump instrument line tap inside the reactor pressure vessel in the annulus between the vessel wall and an internal shroud around the core. The instrument sensing line leaves the jet pump at right angles to the flow. This arrangement minimizes the likelihood of loose material entering the sensing line. Also, as noted previously, the sample lines may be flushed. Loose material, if it exists, would be swept out by the flush.

GAS SAMPLES

Containment atmosphere samples can be collected from either the drywell or the torus. The penetrations through which these samples are taken have not been changed from those described in TVA's submittal from L. M. Mills to H. R. Denton dated November 16, 1982.

BFN containment atmosphere sampling lines will not be heat traced since samples obtained from these lines will be analyzed for noble gases. Noble gases do not plateout on sample line surfaces, therefore, heat tracing is not needed. Iodine plateout is not a concern since gaseous iodine analysis is not required for the core damage estimate (reference BFN TI-88). This provision has been accepted by NRC for other plants (reference Safety Evaluation Reports from Vermont Yankee and Northeast Utilities Millstone Unit No.2).

Strainers or filters are not used in the gas sample lines. We do not expect that debris will exist in the drywell or torus airspaces in sufficient quantity to affect gas sampling operations.

GENERAL

Before collection of any sample the sample line will be purged. During this operation turbulent flow rates will be maintained. Flushing fluid will be routed to the torus. Turbulent flow will also be maintained during sample collection. In addition, after each sample is collected the sample lines will be flushed to the torus. For gas sample lines, nitrogen or air will be used to flush the lines. Portions of the liquid sample lines will be flushed with demineralized water.

FLOW RESTRICTIONS

BFN has redundant solenoid-operated primary containment isolation valves on the RHR and containment atmosphere sample lines. The isolation valves, which are closed except when sampling is in progress, can be secured closed by operator action in the main control room. The jet pump instrument line, from which the reactor coolant sample is obtained, relies on an existing excess flow check valve and flow restricting orifice to limit potential leakage from the sample lines.

11(b) The air exhaust from PASS is routed into the secondary containment by maintaining a slight negative pressure on the panel. The exhaust is returned to the secondary containment through a two inch exhaust duct. The secondary containment atmosphere can be exhausted through the Standby Gas Treatment System which is equipped with HEPA filters and charcoal absorbers.