TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

November 16, 1982

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

Dear Mr. Denton:

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

Enclosed is our response for Browns Ferry to the July 13, 1982 letter from D. B. Vassallo to H. G. Parris requesting documentation of how we meet each criterion of NUREG-0737, Item II.B.3, "Post Accident Sampling System."

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Licensing

Subscribed and sworn to before
me this 16th day of Nov. 1982.

Paulette H. White
Notary Public

My Commission Expires 9-5-84

A046

Enclosure
cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
ATTN: James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. R. J. Clark
Browns Ferry Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

BROWNS FERRY POSITION IN REFERENCE TO NRC "CLARIFICATION ON NUREG-0737, ITEM II.B.3 (POSTACCIDENT 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 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 postaccident sampling facility (PASF) is located on floor elevation 565 of the turbine building between columns M, J, T10, and T14 and contains all equipment necessary for sample acquisition and chemical and radiochemical analyses for all three units (except for containment atmosphere H_2 analysis) as required in NUREG-0737, item II.B.3. Also, see responses 11.A.4 and 11.A.7.

For sample acquisition and portions of the chemical analysis, the Sentry Equipment Corporation (SEC) "Model A" High Radiation Sampling System (HRSS) is used. This system is composed of the liquid sample panel (LSP), chemical analysis panel (CAP), containment atmosphere sample panel (CASP), and their associated control panels. During accident conditions, the following samples can be obtained from the LSP:

(a) Undiluted and diluted (1000:1) liquid grab samples of the reactor coolant.

(b) An inline sample of reactor coolant which is depressurized and degassed in place, and the stripped gas and depressurized coolant is sent to the CAP.

(c) Diluted (15000:1) stripped gas grab samples from the reactor coolant pressurized liquid sample.

The LSP and CASP have the capability to purge lines before sampling to ensure representative samples can be obtained. Sample lines from the LSP, CASP, and CAP can be flushed after the sampling operations are complete to reduce residual radioactivity.

The LSP uses shielded cart/casks for the removal of the reactor coolant. The cask is mounted on a cart which allows the samples obtained to be mobile. A shielded syringe is used to transfer an aliquot of (1000:1) diluted reactor coolant to the fume hood for offline analysis. The fume hood is located in the PASF near the panels.

Samples from the CASP can be collected in shielded cart/casks. These cart/casks are similar to those described for the CSP. TVA will use the Radiological and Chemical Technology (RCT) containment atmosphere separations device. This device separates the containment air sample into particulates, iodine, and noble gases. Particulates and iodine are removed by filter and the noble gases are then obtained in a vial. This system provides samples that can easily be transported.

SEC provided the following sample acquisition and analysis times:

a.	Reactor coolant (RC) diluted sample	30 minutes
b.	RC inline chemical analysis (pH, conductivity, dissolved oxygen, chloride)	45 minutes
c.	RC offgas and dissolved H_2	35 minutes
d.	Containment atmosphere sample	15 minutes

In addition to the above stated sampling-analysis times, a maximum of 30 additional minutes may be needed to purge the sample lines.

After appropriate additional dilution and/or sample preparation in the fume hood, isotopic analysis will be performed with counting equipment located in the PASF. The manual boron analysis will be performed in the fume hood. The time required for the offline analyses of samples has not been determined; however, one sampling-analysis process will be completed within 3 hours after requested.

Based on the original requirement of item II.B.3 of NUREG 0737, no provisions for sampling during loss of offsite power was specified in the design of the Browns Ferry PASF.

As described in chapter 8 of the FSAR, Browns Ferry is connected into an existing network of large load centers. The three generating units are tied into TVA's 500-kV transmission system by way of seven 500-kV transmission lines. The 161-kV switchyard is supplied by two 161-kV transmission lines.

These sources have sufficient capacity to supply the total required power to the plant's electrical auxiliary power system under normal, shutdown, and loss of coolant accident (LOCA) conditions for any single transmission contingency. Separation of the lines, the protection system, and a strong transmission grid minimize the probability of simultaneous failures of offsite power sources. Steady-state studies show these offsite sources to be capable of supplying the onsite power system when all nuclear units are simultaneously removed from service.

The probability of a total loss of offsite power is sufficiently small to preclude the need for providing additional backup power for the PASF.

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within a 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 capabilties 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, ^{133}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).

2. Revisions 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) A discussion of counting equipment capabilities will be given in responses 9.A and 10. Provisions for reducing background radiation will be given in response 9.B.

As stated in NUREG-0737, part II.B.3, the presence outside the core of abnormally high levels of certain radionuclides is indicative of certain types of core damage, e.g., noble gases indicate cladding failure, iodines and cesiums indicate high fuel temperatures, and nonvolatile isotopes indicate fuel melting. For those accidents in which significant release of primary coolant to the containment occurs, the analysis of containment atmosphere samples will give a good estimate of the fraction of the core noble gas fission product inventory which has been released from the core. This noble gas release fraction correlates directly with the amount of cladding failure. For core damage beyond cladding failure, analyses of liquid reactor coolant and sump water are needed to estimate damage.

Specific counting procedures will be developed for use with the sampling equipment provided for the Browns Ferry PASF to estimate the total quantity of indicator radionuclides released during the accident. These evaluations combined with available knowledge of the core isotopic inventories permit estimation of the fractions of core isotopic inventories that have been released to the coolant. These release fractions provide qualitative information on the extent of core damage. A specific procedure to relate radionuclide concentrations to core damage will be developed once acceptable guidelines become available.

2 (b) NUREG-0737 did not specify the need for a grab sample to analyze for hydrogen levels in containment atmosphere. We planned early to use an inline monitor to meet this requirement. The existing Hays-Republic model No. SH-A00643-D hydrogen analyzer presently used to fulfill NUREG-0737, item II.F.1, attachment 6 will also meet the requirement here to analyze for containment hydrogen.

2 (c) The majority of the chemical analyses on the reactor coolant is performed by the SEC CAP. Its capabilities are as stated below:

Analysis Performed	Range
Chloride	100-1000 ppb 1-20 ppm
Dissolved Hydrogen	10-2000 cc/kg
Dissolved Oxygen	0-20 ppb 0-200 ppb 0-20 ppm
Dissolved Oxygen	0.1-5 ppm 1-10 ppm 1-20 ppm
pH	1-13

The range of the Hays-Republic hydrogen analyzer is 0-20% hydrogen. TVA has not yet decided upon a boron analysis method. Investigations are being performed to determine the most suitable method.

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

2 (d) The SEC CAP uses the following inline instrumentation:

a. Baseline gas chromatograph

b. Beckman pH monitor

c. Dionex ion chromatograph

d. REXNORD and YSI dissolved oxygen analyses

This equipment was tested by NUS; and TVA entered into a contract with ORNL to determine the technical adequacy of the SEC equipment. Both of these evaluations were favorable.

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

Reactor coolant and containment atmosphere sampling during postaccident conditions do not require use of an isolated auxiliary system. Certain solenoid valves that are inaccessbile after an accident will need to be opened by overriding primary containment isolation systems (PCIS) signals. These remotely operated valves meet applicable IEEE Class 1E environmental requirements. Some of these valves are in the containment inerting system and are used for taking and returning sample streams for H_2 analysis of drywell and torus containment atmospheres. Other such solenoid valves are in the sampling system itself.

Additional information as to sample sources is provided in response 11.A.1.

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

The LSP has the capability to obtain pressurized liquid samples. These samples are then depressurized and degassed and sent to the CAP. In the CAP, dissolved H_2 and O_2 can be determined.

If the chloride concentration exceeds 0.15 ppm, the dissolved oxygen measurements 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 anlaysis 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 TVA; therefore, our chloride analysis will be completed in four days. This analysis will be done by the CAP on undiluted samples.

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 personnel exposures based on person-motion for sampling, transport, and analysis of all required parameters.

Response

The major source of personnel dose while obtaining the sample is attributable to radiation brought into the Browns Ferry PASF in connection with sample acquisition. This is primarily due to the location selected for the PASF and to low airborne concentration of radioactivity.

The radiological basis for the postaccident sampling equipment purchased for Browns Ferry is given in the vendor's (SEC) design specification for the system. This basis is a source strength which uses radionuclide release fractions specified in NUREG-0578. Two accident cases have data provided for them--a major reactor coolant line break and a case with severe core damage but no line break. The data in the SEC specification are given for both containment atmosphere and reactor coolant samples in terms of a volumetric gamma ray source strength as a function of time and accident and are attached as tables 1, 2, and 3. TVA has determined that these sample activities are applicable to Browns Ferry (due to the reactor thermal power/dilution volume ratio used in deriving these tables being the same for Browns Ferry and the reference plant).

Since airborne concentrations are low, personnel doses during transit to and from the PASF are small. The whole body (gamma) doses incurred in obtaining the sample and transporting it to the fume hood are less than half the GDC 19 criterion of 5 rem. Additionally, the corresponding skin and thyroid doses are negligible.

The predicted man-rem exposure from the offline analysis for boron and isotopic analysis will be significantly less than half the GDC criterion of 5 rem.

TABLE 1*
AIRBORNE SOURCES FOR LINE BREAK ACCIDENT
100% NOBLES & 25% HALOGENS RELEASED
DILUTION VOLUME = 4.49E+09 CC

SOURCE GAMMA SPECTRA

EBAR (MEV)	VOLUMETRIC SOURCE STRENGTH(PHOTONS/SEC/CC) 0-HR	1-HR	4-HR	5-HR	8-HR	12-HR	24-HR
1.500-02	5.336+08	8.755+07	5.510+07	4.797+07	3.307+07	2.274+07	1.336+07
2.500-02	1.829+08	4.889+07	2.423+07	1.972+07	1.147+07	6.675+06	3.225+06
3.500-02	8.020+08	6.942+08	6.831+08	6.797+03	6.697+08	6.566+08	6.185+08
4.500-02	8.679+07	4.871+06	2.866+06	2.650+06	2.321+06	2.072+06	1.489+06
5.500-02	4.684+07	3.713+06	2.178+06	2.010+06	1.753+06	1.562+06	1.121+06
6.500-02	3.762+07	2.920+06	1.620+06	1.478+06	1.266+06	1.121+06	8.120+05
7.500-02	1.257+08	5.272+06	4.151+06	4.029+06	3.840+06	3.702+06	3.379+06
8.500-02	4.949+08	4.548+08	4.521+08	4.514+08	4.490+08	4.450+08	4.303+08
9.500-02	8.036+07	5.339+06	2.646+06	2.367+06	1.976+06	1.736+06	1.261+06
1.500-01	1.521+09	2.448+08	1.218+08	9.993+07	5.795+07	3.118+07	1.042+07
2.500-01	2.203+09	3.214+08	3.973+08	4.189+08	4.494+06	4.374+08	2.846+08
3.500-01	4.234+08	1.603+08	1.482+08	1.454+08	1.393+08	1.343+08	1.263+08
4.750-01	2.150+09	6.748+08	4.666+08	4.305+08	3.546+08	2.862+00	1.711+08
6.550-01	1.364+09	5.111+08	4.258+08	4.151+08	3.911+08	3.844+08	3.414+08
8.250-01	1.591+09	7.421+08	3.336+08	2.866+08	2.330+08	2.113+08	1.788+08
1.000+00	3.428+08	1.502+08	8.343+07	7.479+07	6.272+07	5.555+07	4.450+07
1.225+00	1.082+09	2.735+08	1.972+08	1.750+08	1.354+08	1.004+08	5.047+07
1.475+00	6.479+08	1.099+08	7.773+07	7.116+07	5.777+07	4.759+07	3.472+07
1.700+00	6.855+08	9.290+07	4.102+07	3.579+07	2.532+07	1.673+07	5.076+06
1.900+00	8.722+07	5.060+07	2.978+07	2.648+07	2.003+07	1.486+07	7.590+06
2.100+00	5.269+08	9.437+07	3.454+07	2.706+07	1.332+07	5.572+06	1.168+06
2.300+00	1.828+08	1.167+08	5.555+07	4.336+07	2.064+07	7.670+06	3.935+05
2.500+00	9.445+07	2.512+07	4.982+06	2.972+06	7.033+05	2.091+05	9.679+04
2.700+00	6.227+07	1.587+05	9.784+04	8.575+04	6.285+04	4.898+04	3.782+04
3.000+00	1.361+08	2.606+06	2.457+05	1.329+05	2.430+04	2.706+03	3.804+00
6.143+00	3.062+07	1.357+01	1.578-16	3.575-22	0.000	0.000	0.000
7.112+00	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Totals	1.552+10	4.878+09	3.642+09	3.465.09	3.144+09	2.875+09	2.330+09
**	1.751-01	5.016-02	3.209-02	2.949-02	2.492-02	2.156-02	1.624-02

*Taken from the NUS document, "Evaluation of the Dose Rate and Shielding Requirements for the HRSS Equipment."

**Dose rate rad/hr at 1 meter from unshielded 1 cc sample

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TABLE 2*
LIQUID SOURCES FOR LINE BREAK ACCIDENT
50% HALOGENS & 1% SOLIDS RELEASED
DILUTION VOLUME = 3.45E+09 CC

SOURCE GAMMA SPECTRA

EBAR (MEV)	VOLUMETRIC SOURCE STRENGTH(PHOTONS/SEC/CC) 0-HR	1-HR	4-HR	5-HR	8-HR	12-HR	24-HR
1.500-02	2.015+08	5.716+07	3.641+07	3.365+07	2.922+07	2.590+07	2.003+07
2.500-02	7.168+07	2.899+07	2.094+07	1.994+07	1.820+07	1.675+07	1.404+07
3.500-02	8.400+07	5.267+07	4.841+07	4.768+07	4.617+07	4.461+07	4.099+07
4.500-02	2.988+07	1.157+07	8.826+06	8.441+06	7.759+06	7.194+06	6.087+06
5.500-02	2.368+07	0.397+06	7.174+06	6.846+06	6.241+06	5.707+06	4.614+06
6.500-02	1.724+07	4.803+06	3.041+06	2.805+06	2.417+06	2.124+06	1.602+06
7.500-02	1.830+07	5.443+06	2.496+06	2.247+06	1.903+06	1.675+06	1.274+06
8.500-02	2.811+07	1.464+07	1.313+07	1.282+07	1.255+07	1.222+07	1.147+07
9.500-02	3.559+07	9.762+06	6.168+06	5.728+06	5.067+06	4.620+06	3.843+06
1.500-01	1.768+08	9.675+07	5.374+07	4.928+07	4.429+07	4.214+07	3.794+07
2.500-01	2.148+08	8.075+07	6.174+07	5.962+07	5.652+07	5.426+07	4.907+07
3.500-01	4.832+08	3.684+08	3.568+08	3.554+08	3.516+08	3.467+08	3.328+08
4.750-01	1.076+09	9.357+08	7.458+08	7.119+08	6.406+08	5.687+08	4.113+08
6.550-01	1.523+09	1.377+09	1.133+09	1.101+09	1.050+09	1.007+09	8.954+08
8.250-01	2.337+09	1.839+09	8.438+08	7.339+08	6.159+08	5.707+08	4.919+08
1.000+00	4.991+08	3.858+08	2.145+08	1.935+08	1.648+08	1.474+08	1.186+08
1.225+00	1.090+09	6.806+08	4.884+08	4.455+08	3.487+08	2.606+08	1.321+08
1.475+00	4.519+08	2.721+08	1.789+08	1.699+08	1.502+08	1.322+08	1.031+08
1.700+00	2.211+08	1.748+08	1.053+08	9.223+07	6.574+07	4.359+07	1.332+07
1.900+00	1.971+08	1.332+08	7.829+07	6.939+07	5.249+07	3.896+07	1.996+07
2.100+00	1.260+07	5.209+06	3.414+06	3.278+06	3.070+06	2.933+06	2.636+06
2.300+00	4.939+07	1.551+06	1.873+05	1.609+05	1.415+05	1.348+05	1.286+05
2.500+00	8.461+07	2.156+07	4.220+06	2.543+06	9.910+05	7.629+05	7.101+05
2.700+00	7.072+07	1.221+06	2.545+05	2.055+05	1.441+05	1.186+05	9.878+04
3.000+00	1.238+08	5.325+06	3.875+05	2.190+05	6.130+04	2.367+04	1.538+04
6.143+00	6.690+07	1.287+04	6.197+03	4.838+03	2.303+03	8.559+02	4.391+01
7.112+00	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Totals	9.189+09	6.528+09	4.415+09	4.128+09	3.675+09	3.337+09	2.713+09
**	1.295-01	8.586-02	5.597-02	5.183-02	4.513-02	4.012-02	3.138-02

*Taken from the NUS document, "Evaluation of the Dose Rate and Shielding Requirements for the HRSS Equipment."

**Dose rate/rad/hr at 1 meter from unshielded 1 cc sample

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TABLE 3*
LIQUID SOURCES FOR LINE BREAK ACCIDENT
100% NOBLES, 50% HALOGENS, & 1% SOLIDS RELEASED
DILUTION VOLUME = 2.69E+08 CC

SOURCE GAMMA SPECTRA

EBAR (MEV)	VOLUMETRIC SOURCE STRENGTH(PHOTONS/SEC/CC) 0-HR	1-HR	4-HR	5-HR	8-HR	12-HR	24-HR
1.500-02	1.047+10	1.893+09	1.201+09	1.063+09	7.810+08	5.836+08	3.833+08
2.500-02	3.693+09	1.109+09	6.242+08	5.401+08	3.862+08	2.920+08	2.077+08
3.500-02	1.409+10	1.202+10	1.180+10	1.174+10	1.156+10	1.132+10	1.065+10
4.500-02	1.701+09	1.904+08	1.363+08	1.297+08	1.187+08	1.097+08	9.007+07
5.500-02	9.814+08	1.510+08	1.085+08	1.030+08	9.361+07	8.550+07	6.766+07
6.500-02	7.639+08	8.450+07	5.000+07	4.591+07	3.948+07	3.484+07	2.580+07
7.500-02	2.262+09	1.369+08	8.877+07	8.457+07	7.866+07	7.461+07	6.617+07
8.500-02	8.491+09	7.691+09	7.636+09	7.621+09	7.578+09	7.509+09	7.257+09
9.500-02	1.620+09	1.689+08	9.807+07	9.019+07	7.862+07	7.112+07	5.717+07
1.500-01	2.700+10	4.922+09	2.543+09	2.144+09	1.405+09	9.383+08	5.524+08
2.500-01	3.855+10	6.043+09	7.152+09	7.494+09	7.978+09	7.755+09	5.152+09
3.500-01	1.064+10	5.145+09	4.814+09	4.755+09	4.625+09	4.509+09	4.283+09
4.750-01	4.346+10	1.751+10	1.275+10	1.193+10	1.019+10	8.584+09	5.647+09
6.550-01	3.336+10	1.784+10	1.474+10	1.434+10	1.371+10	1.314+10	1.163+10
8.250-01	4.193+10	2.441+10	1.117+10	9.677+09	8.019+09	7.364+09	6.312+09
1.000+00	9.223+09	5.074+09	2.810+09	2.528+09	2.135+09	1.898+09	1.521+09
1.225+00	2.525+10	8.955+09	6.361+09	5.788+09	4.506+09	3.355+09	1.695+09
1.475+00	1.410+10	3.471+09	2.557+09	2.384+09	2.023+09	1.732+09	1.325+09
1.700+00	1.291+10	2.681+09	1.362+09	1.191+09	8.457+08	5.599+08	1.708+08
1.900+00	2.766+09	1.714+09	1.007+09	8.918+08	6.743+08	5.000+08	2.560+08
2.100+00	8.926+09	1.620+09	6.019+08	4.755+08	2.438+08	1.132+08	3.770+07
2.300+00	3.400+09	1.968+09	9.294+08	2.260+08	3.464+08	1.297+08	8.219+06
2.500+00	2.164+09	5.762+08	1.165+08	7.019+07	2.160+07	1.149+07	9.160+06
2.700+00	1.511+09	1.755+07	4.156+06	3.332+06	2.179+06	1.643+06	1.273+06
3.000+00	3.101+09	8.684+07	8.565+06	4.888+06	1.189+06	3.487+05	1.973+05
6.143+00	9.580+08	1.654+05	7.948+04	6.204+04	2.954+04	1.098+04	5.632+02
7.112+00	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Totals	3.233+11	1.255+11	9.067+10	8.581+10	7.743+10	7.068+10	5.741+10
**	3.810+00	1.411+00	9.100-01	8.390-01	7.186-01	6.294-01	4.836-01

*Taken from the NUS document, "Evaluation of the Dose Rate and Shielding Requirements for the HRSS Equipment."

**Dose rate/rad/hr at 1 meter from unshielded 1 cc sample

B1301A.WA

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

Because Browns Ferry is a BWR, TVA will perform a boron analysis at Browns Ferry if boron is injected into the 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 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 SEC panels provide inline analysis as well as backup grab samples. The LSP has the capability to provide both diluted and undiluted liquid grab samples, and the CASP can provide both diluted and undiluted samples. Both sampling panels have provisions to be flushed upon completion of the sampling operations.

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 1u Ci/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) The operational procedures are being written to handle diluted grab samples from reactor-coolant and containment atmosphere with activity levels given in tables 1, 2, and 3. The grab liquid samples will be further diluted in the fume hood to a total activity of about 1 uCi which is within the normal operating range of a multichannel analyzer (MCA). The computerized MCA will have a library of nuclides expected under accident situations. With this nuclide library, the identified gamma rays with their associated intensities will be used to quantify the radionuclides in the sample.

(9)(b) The major source of background radiation in the counting room (which adjoins the PASF) is attributable to radiation that is brought into the Browns Ferry PASF in connection with taking the sample. By flushing the sample lines after use, shielding the grab samples in a transport cask, and providing ventilation and shielding between the counting room and sampling equipment, the background radiation level is kept low.

The measurement of nuclide concentrations will be performed in a nitrogen-purged thick lead shield. Consequently, the low background radiation in the PASF will not affect the accuracy of the measurement. The accuracy of the measurement will be within a factor of 2.

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
Ra^{+2}	10	Barium Nitrate
La^{+3}	5	Lanthanum Chloride
Ce^{+4}	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li^+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^+	5	
K^+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and 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 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 ranges and accuracies of the accident sample analyses are discussed below:

A,B Gamma Spectrum and Boron

The procedures for offline analysis are being written and will be tested.

C,D,E,F Chloride, Dissolved Oxygen, Dissolved Hydrogen, and pH

The range and accuracy of the equipment is as stated below:

Analysis	Range	Accuracy
Chloride Concentration	100-1000 ppb	± 15%
	1-20 ppm	± 20%
Dissolved Hydrogen	10-2000 cc/kg	± 15%
Dissolved Oxygen	0-20 ppb	± 10%
	0-200 ppb	
	0-20 ppm	
Dissolved Oxygen	0.1-5 ppm	± 10%
	1-10 ppm	
	1-20 ppm	
pH Determination	pH 1 - 13	± 0.5%

Instrumentation in the CAP that performs in-line analysis was tested by NUS using several test matrices. This literature is available for review upon request. The following is a matrix that closely resembles the matrix requested:

Constituent	Concentration (ppm)	Chemical Prepared from
I^-	39	Potassium Iodide
Cs^+	246	Cesium Nitrate
Ba^{+2}	8.1	Barium Nitrate
La^{+3}	2.3	Lanthanum Chloride
Ce^{+4}	5.3	Ammonium Cerium Nitrate
Cl^-	1.8	
B	2000	Boric Acid
Li^+	1.98	Lithium Hydroxide
NO^-	136	
NH^+	1.4	
K^+	12	

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 the equipment supplier, limited testing results, literature reviews, and contacts with experienced personnel engaged in similar analyses under high radiation levels.

Response on Calibration, Testing, and Training - PASF

The Division of Nuclear Power Operational Quality Assurance Manual guidelines will be used to establish specific calibration procedures for the PASF instruments (Ref: N-OQAM, Part III, Section 2.4 - Control of Installed Process Instrumentation and Part III, Section 3.1 - Control of Measuring & Test Equipment). 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. The initial training and retraining of the radiochemical laboratory analyst for postaccident sampling will be implemented as required by present plant procedures (Ref: N-OQAM, Part III, Section 6.1 - Selection & Training of Personnel for Nuclear Power Plants). The retraining will be performed at least annually and more frequently when there are significant changes in subject matter or where other indicators reflect the need. Portions of the PASF will be used for counting during normal operation, and this should give the technicians "hands on" experience in PASF operation.

Criterion: (11) In the design of the post accident sampling and analysis capabilty, 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 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 capabiltities 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.1.a Provisions for purging sample lines.

The LSP has capability for regulating a pre- and post-sample purge flow to 1900 milliliters each minute as well as an actual sample flow to 200 milliliters each minute.

In addition, there is a provision in the PASF to purge at higher flow rates. The sample stream, after passing through the sample cooler, can bypass the LSP (and its internal flow restrictions) and the waste vessel with final routing to the torus of the sampled unit or to the clean radwaste (CRW).

The containment atmosphere sample flowrate can be adjusted at the CASP by adjusting the nitrogen supply pressure to the eductor, a type of jet pump. The eductor's vacuum draws the containment atmosphere sample from the accident unit's drywell or torus.

11.A.1.b Provisions for reducing plateout in sample lines.

To reduce plateout in liquid sampling lines, the purge flow shall be turbulent (Reynolds number $\geq$ 4000). With 1/2-inch schedule 40 pipe, turbulent flow will not occur at minimum temperature with a purge flow through the liquid sampling panel (LSP) of 1900 milliliters each minute or approximately 0.5 gallons each minute. See response 11.A.1.a above for a discussion on bypassing the LSP and the waste vessel as a means of increasing sample velocity and Reynolds number to minimize plateout during pre-sample purging.

11.A.2 To reduce plateout in containment atmosphere sampling lines, the lines are heat traced to maintain a surface temperature of 280°F and are thermally insulated. The temperature setpoint was chosen since it is the maximum design temperature specified in the FSAR for the drywell during the worst postaccident conditions. This heat tracing shall maintain the sample stream's temperature en route to the PASF, thus reducing iodine plateout and steam condensation and retaining the integrity of the sample.

11.A.3 Provisions for minimizing sample loss or distortion.

To minimize sample losses, a tight sampling system must be maintained. On the long runs for both water and containment atmosphere sampling lines, socket-weld connections for the piping are used for minimum leakage. These lines will be pressure tested after being installed. The various solenoid valves are sealed internally against stem leakage. Hand valves and check valves have been chosen for their low leakage construction.

For minimum sample distortion, the samples should arrive at the sample station (PASF) with the same constituents in their same percentages as in the reactor systems from which they came. To do this, the sample stream should not lose any of its entrained particles and should not pick up any additional particles.

As a description of the effort to retain entrained particles, see the discussion in responses 11.A.1 and 11.A.2. They discuss Reynolds number and its importance in reducing plateout in the liquid sampling lines and the use of heat tracing on the containment atmosphere sampling lines to reduce plateout.

Following each sampling process, the liquid lines are backflushed with demineralized water, and the atmosphere sampling lines are backflushed with nitrogen. These operations clean out the previous sample fluids and reduce residual radioactivity in the lines between sampling operations.

11.A.4 Provisions for preventing blockage of sample lines by loose material in the RCS or containment.

No strainers or filters are used to prevent line blockage of containment atmosphere sample lines by loose material in the containment. Our position is that of the total amount of material entrained in the containment atmospheres, very little of it could be drawn into the sample lines because of the very small volume of the sample taken versus the very large volume of either the torus or dryweil atmosphere.

No strainers or filters are used to prevent blockage of water lines by loose material in the reactor coolant system. The location of the source of a liquid sample is at a jet pump which is inside the reactor pressure vessel in an annulus between the vessel's wall and an internal shroud around the core. The instrument sensing line leaves the jet pump at right angles to the flow. Very little loose material in the flowing water stream through the jet pump would likely enter the sensing line due to its momentum. Our conjecture is that the sample stream will contain only finely entrained particles and materials in solution.

Pipe scale and crud from the long sampling lines should not be a problem due to stainless steel being chosen as the piping and tubing materials.

Some benefits will likely be obtained when the lines are backflushed. Loose material, if it exists, and some plateout will be swept out by the flushing fluids which empty into the torus of the sampled unit. However, not all of the length of some lines can be backflushed due to the presence of check valves in certain locations.

It should be pointed out that any device, such as a filter or strainer, used to catch material carried along by the sample flow would interfere with the inherent goal of the sampling system. This goal is to obtain representative samples from the reactor systems being sampled.

11.A.5 Provisions for appropriate disposal of the samples.

All reactor coolant samples can be returned to the torus of the sampled unit or to the CRW system. The used sample liquid drains by gravity from the liquid sample panel into a waste vessel. A waste vessel transfer pump pumps the waste vessel liquid, which is a mixture of used sample and a continuous flow of demineralized water used for dilution, to the torus or the CRW.

All containment atmosphere samples can be routed to the sampled unit's torus through a penetration used also as the return line for the torus H_2-O_2 analyzers. A sample is drawn from drywell or torus to the CASP by the vacuum created by a nitrogen-powered eductor. A second eductor is used to pump the used sample to the torus of the sampled unit.

11.A.6 Flow restrictions to limit reactor coolant loss from a rupture of the sample line.

The PASF does not use any flow restrictors to limit reactor coolant loss from a rupture of a sample line. Instead, dependence is placed on redundant IEEE class 1E solenoid-operated isolation valves which trip automatically to closed positions on primary containment isolation system (PCIS) signals or by operator action.

11.A.7 The postaccident 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 two reactor coolant samples--one each from reactor recirculation loops A and B--are taken through instrument-sensing lines normally used in measuring jet pump flow and reactor vessel water level. These sensing lines are connected to recirculation jet pumps located inside the reactor pressure vessel between its outer wall and its inner shroud.

The containment atmosphere sample from the dome of the drywell is taken through a penetration near the 580 elevation by way of solenoid isolation valves which also supply a sample to the H_2-O_2 analyzer. The containment atmosphere sample from the top of the torus is obtained through a penetration at approximately the 541 elevation in the same manner as the drywell sample.

11.A.8 The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment.

The Browns Ferry PASF is located in the turbine building for unit 3. To reduce shielding requirements, the sampling lines are routed through existing pipe tunnels. A pipe tunnel runs north from each reactor building toward the turbine buildings where these three tunnels intersect a common east-west pipe tunnel. The PASF is located over the eastern end of that portion of the pipe tunnel which is in the three turbine buildings. The sampling lines extend through the roof of the pipe tunnel, through a layer of earth, and then through the floor of the PASF. Individual sampling line lengths vary from 200 to 600 feet from the sample tapoff point just outside primary containment to the first valve rack inside the PASF.

For 1/2-inch schedule 40 pipe, the line volume is 365 cubic inches each 100 feet. This is 1.58 gallons of liquid sample or 0.211 cubic foot of air sample. The line volume for the longest line of approximately 600 feet is 9.48 gallons or 1.27 cubic feet.

Used sample fluids (those discharged from the PASF), both liquid and air, are returned to the torus of the sampled unit. Liquids can also be returned to the CRW. Afterward, the liquid sample lines and sample-return lines are flushed to the torus with demineralized water; and the contianment atmosphere sample lines and sample-return lines are flushed to the torus with nitrogen. The clean flushing fluids are allowed to remain in the sample lines until the next sampling operation.

11.A.9 The residues of sample collection should be returned to containment or to a closed system.

Used sample liquid is piped to the CRW or back to the unit from which it came, and it enters the torus through a 10-inch core spray pump test line. Used atmosphere sample is piped back and enters the torus through an H_2-O_2 sample return line.

11.B The exhaust from the PASF is routed during the postaccident operational mode through an air cleanup unit (ACU). This is also true during periodic training operations. The ACU consists of a charcoal adsorber and HEPA filters.

During normal operating conditions when no sampling is being done, the exhaust is not routed through the ACU. In this mode, there will be no significant radioactive material in the exhaust.