May 1, 1984

Docket Nos. 50-259/260/296

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Mr. Hugh G. Parris
Manager of Power
Tennessee Valley Authority
500A Chestnut Street, Tower II
Chattanooga, Tennessee 37401

Dear Mr. Parris:

SUBJECT: NUREG-0737, ITEM II.B.3, POST-ACCIDENT SAMPLING SYSTEM

Re: Browns Ferry Units 1, 2 and 3

The purpose of this letter is threefold: 1) to present our evaluation of the design of the permanent post-accident sampling system at Browns Ferry; 2) to discuss what we have determined to be an unacceptable situation with respect to the interim actions on post-accident sampling (pending completion of the permanent modifications); and 3) to express our concern over the relative priority you have apparently accorded this subject.

Your letter of November 16, 1982 provided design information on the permanent post-accident sampling system and discussed how you intend to meet the criterion in NUREG-0737, Item II.B.3. Our evaluation of your proposed design is discussed in Enclosure 1. We have concluded that the system as described in your letter will satisfactorily meet seven of the eleven criteria - namely, Criteria 3 through 9. With respect to Criteria 1, 2, 10 and 11, we require that you provide, within 90 days of receipt of this letter, the additional information described in our evaluation to complete our review on the design of the permanent system.

NUREG-0578 sets forth the interim actions that should be taken until any necessary modifications to meet the requirements of NUREG-0737 are completed. Your letter of October 17, 1979 stated that:

> "A design and operational review of the reactor coolant sampling systems and analysis facilities is being performed and will be complete by January 1, 1980. TVA expects to complete required modifications by January 1, 1981, provided that equipment procurement/installation conflicts are not encountered. These modifications will make provisions for sampling water from the reactor coolant system for the degraded accident condition. TVA will also identify the type and nature of onsite analysis required."

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Your letter of January 17, 1980 further stated that "until the (permanent) design modifications are complete, procedures have been devised to evaluate the primary coolant system activity depending on the accessibility of the sampling stations for particular degraded conditions." These commitments were required to be completed by the Confirmatory Order issued January 2, 1980 for Browns Ferry Unit 1. During a staff visit to the Browns Ferry site on February 19-20, 1980, staff representatives were told that you planned shielding modifications for the present sampling system until the permanent sampling facility becomes operational.

During the week of October 17-21, 1983, a Region II inspector conducted a routine emergency preparedness inspection of the Browns Ferry Nuclear Plant. In the course of that inspection, the provisions for the interim sampling of primary coolant and drywell atmosphere under post-accident conditions were reviewed. The initial evaluation of the post-accident sampling capability was described in paragraph 12 of Inspection Report 50-259, 260 and 296/83-40, which was transmitted to you by Richard C. Lewis' letter of January 10, 1984. A more detailed discussion of the inspection findings is provided in Enclosure 2. Based on these inspection findings, it appears that:

1. Nothing has been done to provide supplemental shielding of either the sampling hood sinks or the exposed primary coolant sampling lines which traverse the walls of the sampling areas in close proximity to personnel working at the sampling station.

2. There was no substantiating documentation at the Browns Ferry site that a design and operational review of the reactor coolant sampling systems and analysis facilities had been performed which demonstrated that the present sampling system would meet all the criteria in NUREG-0578 and NUREG-0737 without need for any modifications. Our intent in issuing a Confirmatory Order was that existing sampling systems would be modified, as necessary, to meet the interim criteria until such time as the permanent sampling system is operational.

3. You committed to have procedures in place by January 31, 1980 to meet the interim criteria in NUREG-0578. The plant Technical Instructions (TI) include interim procedures for utilizing existing equipment to meet the interim criteria in NUREG-0578. These procedures appear to have significant technical deficiencies and procedural errors. Documentation was not available to demonstrate that the procedures had been tested and were workable.

Within 30 days of receipt of this letter you are requested to provide:

1. The results of the design and operational review which you conducted of the reactor coolant sampling systems and analysis facilities with respect to the capability to meet the criteria for interim sampling systems in NUREG-0578 and NUREG-0737.

2. The bases for your determination that the interim procedures (e.g., BF TI 38, pgs. 799-810) for post-accident sampling are workable and meet the interim criteria in NUREG-0578.

The third purpose of this letter is to express our concern over the relative priority you apparently have given to post-accident sampling capability. Most operating reactors have completed the permanent sampling facilities; except for Browns Ferry, all operating plants will complete the permanent facilities by July 1984. The projected completion date for the permanent facilities at Browns Ferry is late 1987. This schedule is unacceptable if the interim sampling facilities and procedures fall significantly short of meeting the criteria for interim facilities in NUREG-0578 and NUREG-0737. You are requested to take appropriate remedial action to either provide an adequate interim post-accident sampling facility or to accelerate installation of the permanent sampling system fully meeting the requirements of NUREG-0737, Item II.B.3. As a first step, a meeting with the Browns Ferry project manager is requested to review a) the work remaining to complete the permanent facility; b) the crafts involved in the approximately 9000 man-hours you estimate for completion; c) the delivery schedules on equipment; d) the interrelationship with other plant modifications; and e) the impact if you were required to complete the permanent post-accident sampling facility prior to startup in Cycle 6 for all three units (i.e., completion in 1984 and spring 1985).

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

Original signed by/

Gus C. Lainas, Assistant Director
for Operating Reactors
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

*Please see previous concurrence page.

DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:AD-OR
SNorris:ajs*	RClark*	DVassallo	GLainas
04/10/84	04/10/84	04/19/84	04/ /84

Mr. Hugh G. Parris
Tennessee Valley Authority
Browns Ferry Nuclear Plant, Units 1, 2 and 3

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ENCLOSURE 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Safety Evaluation by
the Office of Nuclear Reactor Regulation
Related to Operation of
Browns Ferry Unit Nos. 1, 2, and 3
Nuclear Generating Plants
Tennessee Valley Authority
Docket Nos. 50-259, 50-260, 50-296

Post-Accident Sampling System (NUREG-0737, II.B.3)

Introduction

Subsequent to the TMI-2 incident, the need was recognized for an improved post-accident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g. noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

To comply with NUREG-0737, Item II.B.3, the licensee should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary and (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the criteria are met.

Evaluation

By letter dated November 16, 1982, the licensee provided information on the PASS.

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.

The PASS has sampling and analysis capability to promptly obtain and analyze reactor coolant samples and containment samples within three hours from the time a decision is made to take a sample. We determined that the PASS partially meets Criterion (1). Consistent with our clarification of NUREG-0737, Item II.B.3, Post-Accident Sampling Capability, transmitted to the licensee on July 13, 1982, the licensee should provide an alternate power source, not necessarily a Class 1E system, that can be energized in sufficient time, during loss of offsite power, to meet the three-hour sampling and analysis time limit.

Criterion (2):

The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the three-hour time frame established above, quanitification 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.

The PASS provides the capability to collect diluted or undiluted liquid and gaseous grab samples that can be transported to the radiochemical laboratory for hydrogen, oxygen, total dissolved gas, boron, chloride, and radionuclide analyses.

We find that the licensee partially meets Criterion (2) by establishing an on-site radiological and chemical analysis capability. However, the licensee should provide a procedure, consistent with our clarification of NUREG-0737, Item II.B.3, Post-accident Sampling Capability, transmitted to the licensee on July 13, 1982, 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. Guidance for the procedure to estimate core damage is attached. We have reviewed the BWR Owners Group procedure and have found it acceptable (Letter from T.J. Dente, Chairman, BWR Owners' Group, to Darrell G. Eisenhut, NRC, for Estimation of Core Damage Using Post-Accident Sampling System). To fully meet Criterion (2) the licensee should provide a plant specific procedure to estimate the extent of core damage.

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.

Sampling of the reactor coolant and containment atmosphere during post-accident conditions does not require an isolated auxiliary system to be placed in operatior in order to perform the sampling function. The PASS provides the ability to obtain samples from the jet pump instrument sensing line, the residual heat removal system, and the containment atmosphere, without using an isolated auxiliary system. The PASS valves which are not accessible after an accident are environmentally qualified for the conditions in which they need to operate. We find that these provisions meet Criterion (3) and are, therefore, acceptable.

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.

Equipment for stripping dissolved gases from a fixed-volume pressurized liquid sample is provided for dissolved H_2 and O_2 analyses. If chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is possible. We determined that these provisions meet Criterion (4) and are, therefore, acceptable.

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.

The chloride analysis is performed on the coolant by PASS inline instrumentation within the 96 hour time limit, with a measurement range of 0.1 to 20 ppm and accuracy of $\pm$ 20%. We determined that these provisions meet Criterion (5) and are, therefore, acceptable.

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).)

The licensee has performed a shielding analysis to ensure that operator exposure while obtaining and analyzing a PASS sample is within the acceptable limits. This operator exposure includes entering and exiting the sample panel area, operating sample panel manual valves, positioning the grab sample into the shielded transfer carts, and performing manual sample dilutions, if required, for isotopic analysis. PASS personnel radiation exposures from reactor coolant and containment atmosphere sample and analysis are within 5 rem whole body and 75 rem extremities which meet the requirements of GDC 19 and Criterion (6) and are, therefore, acceptable.

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.)

The PASS has the capability to analyze coolant boron concentrations in the range of 1000 to 6000 ppm with an accuracy of $\pm$ 5%. At concentrations below 1000 ppm the tolerance is $\pm$ 50 ppm. We find that this provision meets Criterion (7) and is, therefore, acceptable.

Criterion (8):

If in-line 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.

The PASS provides inline analysis as well as backup grab samples. The grab samples can either be diluted or undiluted. The inline analysis sample can also be either diluted or undiluted. Backup chemical and radiochemical analyses will be performed in the post-accident sampling facility in the turbine building. We find that these provisions meet Criterion (8) and are, therefore, acceptable.

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 term given in Regulatory Guides 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 provide. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1m 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.

The radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source term given in Regulatory Guides 1.3 and 1.7. Radiation background levels will be restricted by shielding between the counting room and sampling equipment and ventilation in the counting room such that analytical results can be obtained within an acceptably small error (approximately a factor of 2). We find that these provisions meet Criterion (9) and are, therefore, acceptable.

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.

The PASS has the analytical ranges and accuracies that are consistent with the recommendation of Regulatory Guide 1.97, Rev. 2 and the clarification of NUREG-0737, Item II.B.3, Post-Accident Sampling Capability, transmitted to the licensee on July 13, 1982. However, information on the applicability of these procedures under accident conditions was incomplete.

We find that the licensee partially meets Criterion (10). The licensee should provide additional information consistent with the guidelines in our letter dated July 13, 1982, on performance of the PASS instrumentation in an accident environment. Additionally, all equipment and procedures which are used for post-accident sampling and analysis 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.

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 line, 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 adsorbers and high-efficiency particulate air (HEPA) filters.

The licensee has addressed provisions for purging to ensure samples are representative, size of sample line to limit reactor coolant loss from a rupture of the sample line, and ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters. To limit iodine plateout, the containment air sample line is heat traced. We determined that these

provisions partially meet Criterion (11) of Item II.B.3 of NUREG-0737. The licensee should demonstrate that the jet pump instrument-sensing lines which are used for PASS sampling are representative of core conditions. The attached LRG II position on reactor coolant sampling should be considered by the licensee in showing that sampling is representative.

Conclusion

We conclude that the post-accident sampling system partially meets the criteria of Item II.B.3 of NUREG-0737. The licensee's proposed methods to meet seven of the eleven criteria are acceptable. The four criteria which have not been fully resolved are:

Criterion (1)	Provide information regarding provisions for sampling in the event of loss of offsite power during an accident which requires post-accident sampling.
Criterion (2)	Provide a core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage.
Criterion (10)	Provide information demonstrating applicability of procedures and instrumentation in the post-accident water chemistry and radiation environment, and retraining of operators on semi-annual basis. Provide performance test data on the PASS instrumentation in an accident environment.
Criterion (11)	Provide information demonstrating that the reactor coolant sampling locations are representative of core conditions.

ATTACHMENT 1

ILLINOIS POWER COMPANY

U-0454
L30-82(03-30)-6

500 SOUTH 27TH STREET, DECATUR, ILLINOIS 62525

March 30, 1982

Mr. James R. Miller, Chief
Standardization & Special Projects Branch
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Miller:

Clinton Power Station Unit 1
Docket No. 50-461

By the letter dated March 12, 1982, Mr. Dale Holtzscher, Chairman of the LRG-II Working Group, provided you with position papers on twelve of the fifty-six presently identified LRG-II issues. As you know, Illinois Power Company is a member of the LRG-II. We would like to advise you, in accordance with the practices developed by NRR and the LRG-II, that Illinois Power Company hereby incorporates those twelve position papers provided in Mr. Holtzscher's March 12, 1982 letter into the Clinton Power Station OL application.

For specificity, the March 12, 1982 letter provides information on the LRG-II issues identified in the following manner:

3-CPB	4-ICSB	1-MEB
11-CPB	1-PSB	1-SEB
1-CSB	1-CHEB	2-SEB
3-ASB	2-CHEB	4-CPB, Revision 1

When subsequent LRG-II position papers are developed and provided to NRR, we intend to incorporate them in similar fashion.

Sincerely,

G. E. Wuller

G. E. Wuller
Supervisor-Licensing
Nuclear Station Engineering

bcc: CPS/DRC (0950)
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A. J. Budnick
E. W. Kant - Engr. Supr.
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GEW:mr

cc: J. H. Williams, NRC Clinton Project Manager
H. H. Livermore, NRC Resident Inspector
H. F. Faulkner, NRC LRG-II Project Manager
Illinois Dept. of Nuclear Safety

1-CHEB
REACTOR COOLANT SAMPLING

ISSUE

In response to the requirements of NUREG-0737, Item II.B.3, "Post Accident Sampling Capability", LRG-II plants are required to demonstrate that the reactor coolant system sampling locations will provide coolant samples that are representative of core conditions. Of specific concern is the potential for significant dilution of the sample by makeup water which can result in the samples being analyzed at lower concentrations of soluble species (chlorine, boron, iodine, etc) than are actually present in the core.

LRG-II RESPONSE

The LRG-II position is that reactor coolant samples obtained from a tap off the jet pump pressure instrument system will provide representative core coolant samples for accident conditions and that samples be taken from this location.

In order to assure that this sample location provides a representative sample, sufficient core flow is needed to circulate water from the core to the jet pump intake. After a small break or non-break accident, the reactor water level is maintained at or near normal water level by the operator using emergency procedures. For decay power above 1% of rated power the core flow is estimated to be greater than 10% rated flow due to natural circulation. The entire reactor water inventory would be circulated through the jet pumps in about 3 to 4 minutes, thus assuring that representative samples of core coolant will be available at the jet pumps.

MJA:hjr:rf/113A29

At power levels of less than 1% rated, a sample that is representative of core conditions would be obtained by increasing the reactor water level by 18 in. This will fully flood the moisture separators and will provide a thermally induced recirculation flow path for mixing.

Makeup water does not significantly dilute the sample. Makeup water flow amounts to approximately 2% of the core flow for small steam line breaks or non-break accidents. For small liquid line breaks, the makeup water flow rate is estimated to be less than 18% of the core flow. Thus, no significant dilution occurs and the water circulating through the jet pump is representative of reactor coolant inventory for small break or non-break accidents.

Further, sample lines in the RHR system provide for a reactor coolant sample when the reactor is depressurized and at least one of the RHR loops is operating in the shutdown cooling mode.

Finally, for larger line breaks where reactor water level cannot be maintained, reverse flow through the core to the suppression pool is provided. Suppression pool samples are obtained from the RHR pump discharge as discussed in the LRG-II position paper 2-CHEB "Suppression Pool Sampling".

MJA:hjr:rf/113A30
3/12/82

2-CHEB
SUPPRESSION POOL SAMPLING

ISSUE

In response to the requirements of NUREG-0737, Item II.B.3, "Post Accident Sampling Capability," LRG-II plants are required to demonstrate that the suppression pool sample locations will provide samples that are representative of pool inventory.

LRG-II RESPONSE

The LRG-II position is that suppression pool samples, obtained from the Residual Heat Removal pump discharge with the RHR loop lined up in the suppression pool cooling mode, will be representative of the pool inventory and that samples will be taken from this location.

The sample lines will be installed on the discharge side of the RHR pumps downstream of the pump check valve. Representative samples will be assured by operating the selected RHR loop for approximately 30 minutes prior to taking a sample. Since no SRV's discharge directly into the RHR suction and the SRV discharge locations in the pool facilitate mixing, the suppression pool sample location will provide adequately mixed samples that will be representative of pool inventory.

ILLINOIS POWER COMPANY

U-0551
L30-82(09-23)6

500 SOUTH 27TH STREET, DECATUR, ILLINOIS 62525

September 23, 1982

Mr. Cecil O. Thomas, Chief
Standardization & Special Projects Branch
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Thomas:

Clinton Power Station Unit 1
Docket No. 50-461

By the letter dated September 3, 1982, Mr. Dale Holtzscher, Chairman of the LRG-II Working Group, provided you with position papers on six of the fifty-eight presently identified LRG-II issues. As you know, Illinois Power Company is a member of the LRG-II. We would like to advise you, in accordance with the practices developed by NRR and the LRG-II, that Illinois Power Company hereby incorporates five of the six position papers provided in Mr. Holtzscher's September 3, 1982 letter into the Clinton Power Station OL application.

For specificity, the September 3, 1982 letter provides information on the LRG-II issues identified in the following manner:

2-CPB, Rev. 1	3-CHEB, Rev. 1
2-HFS, Rev. 1	5-ASB
3-HFS, Rev. 1	4-MEB

Illinois Power is not incorporating the LRG-II position paper on issue 2-HFS into our application at this time since we are attempting to develop a plant-unique program to address this issue.

LRG-II Position Paper

September 3, 1982

Revision 1

3-CHEB

ESTIMATION OF FUEL DAMAGE FROM POST-ACCIDENT SAMPLES

ISSUE:

A procedure for relating post-accident radionuclide concentrations in reactor coolant and suppression pool samples should be developed.

LRG-II POSITION:

It is the LRG-II position to develop plant specific programs to estimate fuel damage based on the Enrico Fermi-2 Project procedure transmitted by a letter dated April 29, 1982 from Harry Tauber (Detroit Edison) to L. L. Kinter (NRC).

The estimation of core damage will be calculated by comparing the measured concentrations of major fission products in either gas or liquid samples, after appropriate normalization with reference plant data from a BWR-6/238 with a Mark III containment. (Reference: General Electric; Procedures for the Determination of the Extent of Core Damage Under Accident Conditions; RPE 8/CCL01 dated November 1981.)

The procedures will provide locations for obtaining the most representative samples (see 1-CHEB and 2-CHEB Position Papers) depending on accident severity and system conditions. Water samples (reactor coolant, suppression pool and RHR) and gas samples (containment and drywell) are analyzed by gamma spectroscopy for determination of I-131, Cs-137, Xe-133 and Kr-85 concentrations. The measured fission products are corrected for decay and the concentrations are normalized to the reference plant data appropriately for comparison to graphs to indicate percent cladding failure or percent fuel meltdown. Isotopic ratios for noble gasses and iodine are calculated for comparison with the ratios that are normally expected to be found in the core inventory and in the fuel gap.

In addition, LRG-II plant-unique programs will address post-accident sampling system testing and operator training programs as required by Section 6.8.4.c of the Standard Technical Specifications. A third core damage category that is in between cladding failure and core melt, i.e. fuel overheating (metal-water reaction) will be included. Other plant indicators (e.g. reactor water level, hydrogen generation from zirconium-water reaction, containment monitors, etc.) will be factored into the program to aid in the interpretation of the extent of the core damage and cross check whether sampling is representative or sample analysis is reasonable.

INSPECTION FINDINGS
BROWNS FERRY POST-ACCIDENT
SAMPLING SYSTEMS

1. Background

A Region II inspector conducted a routine unannounced emergency preparedness inspection of the Browns Ferry Nuclear Plant, Units 1, 2 and 3, during the week of October 17-20, 1983. In the course of that inspection, the licensee's post-accident sampling provisions (NUREG-0737, Item II.B.3) were reviewed.

2. Findings and Comments

The system which is to be installed is a Sentry/NUS post-accident sampling system (PASS). In discussions with plant personnel, it was indicated that the current estimated date for operation is sometime in 1986. One possible reason for the extended date is the relocation of the PASS from a building which was specially constructed for that purpose to a section of the common turbine building which is being partitioned off and designated for PASS use. It was also reported that a number of design changes have been proposed for the Sentry PASS design.

Until such time as the Sentry/NUS PASS is placed in operation, all primary coolant sampling, including normal operational sampling and post-accident sampling, is being (or will be) performed using the original equipment primary coolant sampling hood sinks. So far as the inspector could determine, the sampling hood sinks have not been modified in any significant manner which would facilitate interim post-accident sampling or mitigate the potential for high radiation exposures expected to be encountered in post-accident situations. This finding is applicable to all three operating reactors at Browns Ferry.

In a letter dated February 29, 1980, from Thomas Ippolito, Chief, Operating Reactors Branch 3, Division of Operating Reactors (NRC), to Hugh G. Parris, Manager of Power, TVA, the NRR staff transmitted an evaluation of the licensee's compliance with "Category A" items of NRC recommendations. On page 7 of that letter, it was stated that the licensee "contemplated" shielding modifications of the installed reactor coolant sampling systems as an interim measure. The licensee's commitment was used as a basis for determining that the licensee had satisfied the intent of the NUREG-0737 requirements for post-accident sampling, had implemented interim post-accident sampling procedures and, therefore, had complied with the requirements for post-accident sampling.

In the inspection, it was determined that nothing had been done to provide supplemental shielding of either the sampling hood sinks or of the exposed primary coolant sampling lines which traverse the walls of the sampling areas in close proximity to personnel working at the sampling station.

In discussions with licensee personnel, it was learned that a study had been done of the feasibility of providing shielding to mitigate the potential dose problem. Reportedly, the study concluded that it was not practicable to provide enough shielding to reduce doses to ALARA levels in the event of an accident and, therefore, it was recommended that no action be taken to install additional shielding. While the above is unsubstantiated by documents, it does support the inspector's findings that no action had been taken to mitigate the potential radiation dose consequences of post-accident sampling.

Operating procedures for post-accident sampling are detailed in a "technical instruction", BF TI-38, dated 7/7/81. This constitutes the only provision for interim post-accident sampling being implemented at Browns Ferry. TI-38 suggests that in the event of an accident requiring post-accident sampling, lead blankets, lead sheet, and lead bricks could be used to reduce radiation levels. This recommendation is in direct contradiction to the licensee's study conclusions relative to the practicability of shielding.

The principal thrust of the NUREG-0578 and NUREG-0737 recommendations for post-accident sampling was that licensee should prepare in advance to counteract or mitigate potential problems known or anticipated to accompany the sampling of primary coolant fluids under accident conditions. This has not been done to date at Browns Ferry.

3. Primary Coolant Sampling System and Procedures for Interim Post-accident Sampling

Plant Technical Instructions (TI's) establish interim procedures for utilizing existing equipment (circa March 1979) to meet the interim criteria which followed publication of NUREG-0578. In reviewing the procedures, equipment, and diagrams in the TI's, the inspector identified several technical deficiencies and procedural errors.

The liquid reactor coolant sampling stations at Browns Ferry are remarkably similar to the TMI-2 primary coolant sampling station and would be subject to many of the same problems and deficiencies in an accident situation. The sampling station is a typical chemistry laboratory fume hood with inverted "J" faucets at the rear of the hood. Incoming sample lines are totally exposed to the room environment over a distance of about 25 feet and no attempt has been made to shield these sample lines. The sampling stations are located in an area of the reactor building which may well be radiologically untenable in a maximum accident situation. Chemistry personnel interviewed were of the unanimous opinion that any accident conditions would prevent the entry of anyone into the sampling station area. While this is probably correct for the maximum accident situation, the maximum accident is the least probable to occur and pre-planning for lesser accidents should not be ignored.

The sampling operation involves manually turning on a faucet from a valve panel on the front of the sampling hood, letting the sample drain to a radwaste tank for five minutes, and then collecting a 50 ml sample into an open plastic bottle containing 950 ml of distilled water. The procedure then requires the operator to physically reach into the hood and manually turn a petcock in the base of the bottle to drain 50 ml of the diluted solution into a second bottle also containing 950 ml of distilled water. 15 ml of the diluted solution in the second bottle is then to be manually pipetted into a 15 ml vial, which is then transported in a lead shield to the radio-chem lab where it is again diluted up to 1 liter, for a total dilution of 1:26,800. What is not mentioned or provided for is that in some accident sequences, little or no dilution may be needed; the TI makes no provisions for that circumstance.

On page 802 of TI-38, item "N" states that temporary shielding may be installed to reduce radiation exposure at the sampling station. After-the-fact action of this nature is of little value. If shielding cannot be in-place prior to the time it is needed, there is not sufficient time to do the job after-the-fact and still be able to perform the essential function of obtaining a valid sample in the desired time frame. NUREG-0578 and NUREG-0737 specify the design capability to get a sample and analyze it in three hours. The stated procedure is basically incapable of meeting this goal under accident conditions without advance shielding of the primary coolant sampling system. The Browns Ferry Interim Sampling System and implementing procedures make no such provision.

4. Drywell Atmosphere Sampling System and Procedure for Interim Post-Accident Sampling

Section 1303 of BF TI-38 describes the equipment and procedures for meeting interim post-accident sampling requirements for taking samples from the drywell at Drywell Monitor RE-90-256 when radiation dose rates are > 500 mR/hr and < 100 R/hr.

The inspector's review concluded that the proposed interim sampling method appears to be incapable of procuring the desired sample due to basic design flaws and procedural omissions.

The sampling system relies on vacuum to draw gas from the drywell and also to draw dilution air from a source provided from the service air supply. The procedure calls for dilution flowrate to be established by opening the service air valve and for controlling the flowrate with the service air valve. The instruction is in error since this valve has no effect whatever on the system flowrate. The procedure then directs the dilution flow to be continued for 5 minutes, after which the system valves and vacuum pump are to be turned off. The procedure is missing any instruction to open the sample line valve to allow the drywell gas sample to enter the dilution system. Even if the sample line valve were to be opened, there is no procedural step calling for observation and recording of flowrates in the appropriate legs of the system; such information is needed to determine the dilution ratio.

It was the inspector's conclusion that even if the missing procedural steps had been provided, the sampling system is fundamentally incapable of functioning in the manner described for it. Gas flow in a series-connected string under vacuum conditions is limited by the sum of the pressure drops across the various components but is the same in each series component. If a side stream of air at nominally atmospheric pressure is introduced into a vacuum system at some intermediate point in the system, the system flow conditions will change, with essentially all flow passing through the new side stream, almost no flow going through that portion of the system upstream of the point at which the side stream enters, and a somewhat increased flow will be present in the downstream leg of the system due to a lower overall pressure drop.

The procedure does not recognize that flow in the upstream system leg, which is the source of dilution air, will stop when the sample line valve to the drywell is opened. It is possible that if some form of throttling valve were to be introduced into the system in the incoming drywell gas sample line, a measure or degree of control would be achieved; no such provision is made or described.

The dilution ratio is supposed to be determined by comparison of the difference in flows in the upstream and downstream portions of the series sampling string. Assuming that the throttling conditions described above are achieved, it may be possible to make such a measurement and to determine the dilution ratio based on the difference in flows; however, the inspector noted that no such provision is described in the procedure and no instructions are provided for the calculation of the dilution ratio using the differential flow measurements. It was also noted that no specified range of dilution was mentioned in the procedure. The inspector estimated that a maximum dilution ratio of between 20:1 to 30:1 could be determined from differential flow measurements.