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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION POST-ACCIDENT SAMPLING SYSTEM (NUREG-0737, ITEM II.B.3)

TENNESSEE VALLEY AUTHORITY BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2 AND 3 DOCKET NOS. 50-259, 50-260 AND 50-296

1.0 INTRODUCTION

By letter dated November 16, 1982, the Tennessee Valley Authority (TVA, the licensee) provided information on the Post-Accident Sampling System (PASS). Based on our draft evaluation, we concluded that seven of the eleven criteria were acceptable. The following criteria remained unresolved:

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 an onsite radiological and chemical analysis capability 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 a 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.

By letters dated June 6, and August 13, 1984, the licensee provided additional information on the unresolved criteria.

2.0 EVALUATION

Criterion (1)

Regulatory Guide 1.97, Rev. 3 requires only offsite power for Category 3 instrumentation. Because the Post-Accident Sampling System is classified as Category 3, only offsite power is required. We find that this meets Criterion (1), and is, therefore, acceptable.

Criterion (2)

In the letter of August 13, 1984, TVA informed us that a core damage assessment procedure and supporting documents i.e. a procedure for onsite radiological and chemical analysis capability, were being prepared, based on the General Electric (GE) BWR generic procedure. TVA also stated that completion and implementation of the approved procedure for application to interim post-accident response was planned for September 30, 1984. The GE procedure, NEDO-22215, dated August 1982, has been found acceptable by the staff as a basis for plant-specific procedures. TVA also stated that the interim damage assessment procedure will be modified as needed to function with the permanent PASS equipment and instrumentation when this facility is operational. In the transmittal letter for this Safety Evaluation, we are requesting TVA to provide the interim procedure so that we may evaluate whether the licensee's procedures for handling samples and for analysis of the specified radionuclides and other parameters (hydrogen, dissolved gasses, etc.) are acceptable, taking into consideration the probable radiation levels.

Our concerns about the interim PASS at Browns Ferry have been enumerated in meetings with TVA (e.g., May 9, 1984) and in various inspection reports (e.g., 50-259,260,296/83-40 dated January 10, 1984, 50-259,260,296/84-25 dated August 22, 1984 and 50-259,260,296/85-27 dated June 20, 1985). Our concerns relate primarily from two factors: 1) the length of time the interim PASS may have to substitute for the permanent PASS and 2) the potential radiation levels in the vicinity of the interim PASS in the event of an accident involving significant core damage. The permanent PASS is located in the turbine building. The basic structure was completed in 1981; however, based on the integrated schedule for plant modifications, the tie-ins may take a decade or more to complete. The interim PASS is located in the reactor building. In the event of a severe accident, radiation levels in the reactor building may restrict access to the interim PASS. In the guidance provided to licensees by the NRC staff on developing post-accident sampling procedures, the staff's position was that licensees should apply Regulatory Guide 1.3 ("Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2 dated June 1974) to evaluate the environment to which those personnel taking samples might be exposed, and design the shielding and sampling procedures accordingly. The procedures have to take into account that possible radiation levels will be not only from activity in the coolant but have to account (in the source term) for xenon and krypton that may leak from containment.

Until we (NRR and Region II) have evaluated TVA's interim procedures, we cannot determine whether they acceptably meet Criterion (2) of NUREG-0737, Item II.B.3.

Criterion (10)

The analytical methods and instrumentation were selected for their ability to operate in the post-accident sampling environment. The standard test matrix and radiation effects evaluation indicated no interference in the PASS analyses. Every chemistry technician operator will be trained on using the PASS system; at least once per year, each technician will both operate the PASS and sample the fluids. Results will be compared with samples obtained from normal sample locations to verify that the PASS is functioning correctly. We find that these provisions meet Criterion (10) and are, therefore, acceptable.

Criterion (11)

Reactor coolant samples are obtained from the jet pump instrument-sensing lines that will provide representative core coolant samples. To ensure sample representativeness sufficient core flow is required to circulate from the core jet pump intake. After a small break or a non-break accident, the reactor water level is maintained at or near normal by the operator using emergency procedures. At very low power levels, it may be necessary to raise the reactor water level in order to induce natural circulation core flow. We find these provisions meet Criterion (11) and are, therefore, acceptable.

3.0 CONCLUSION

On the basis of our evaluation, we now conclude that the Post-Accident Sampling System meets ten of the eleven criteria of Item II.B.3 in NUREG-0737. The licensee should provide a plant-specific procedure to estimate the extent of core damage based on the General Electric BWR Generic Procedure to evaluate whether the procedures meet Criterion (2) of Item II.B.3.

Principal Contributor: Frank Witt

Dated: February 21, 1986