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AUTH.NAME AUTHOR AFFILIATION
DOMER,J.A. Tennessee Valley Authority
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards response to 840501 ltr re Criteria 1,2,10 & 11 for permanent post-accident sampling facility.Info on interim sys submitted on 840606.

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TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

August 13, 1984

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

In response to G. C. Lainas' May 1, 1984 letter to H. G. Parris, we are providing the enclosed information regarding criteria 1, 2, 10, and 11, for the Browns Ferry permanent Post-Accident Sampling Facility (PASF). Your letter was the subject of a May 9, 1984 meeting between TVA and representatives of your staff and Region II during which we discussed the interim PASF and the NRC concern of possibly accelerating the completion of the permanent system. Our response to your concerns about the interim system was submitted to you June 6, 1984.

This completes our response to your May 1, 1984 letter.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

James A. Domer

James A. Domer
Nuclear Engineer

Subscribed and sworn to before
me this 13th day of Aug. 1984.

Bryant M. Lowery
Notary Public
My Commission Expires 4/8/86

Enclosure
cc (Enclosure):

U.S. Nuclear Regulatory Commission
Region II
ATTN: James P. O'Reilly, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

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

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ENCLOSURE
REPLY TO MAY 1, 1984 NRC LETTER REGARDING THE
PERMANENT POSTACCIDENT SAMPLING SYSTEM
BROWNS FERRY NUCLEAR PLANT
UNITS 1, 2, AND 3

Criterion 1

Provide information regarding provisions for sampling in the event of loss of offsite power during an accident which requires post-accident sampling.

Response

TVA maintains their position on the response concerning loss of offsite power given to NRC in the letter from L. M. Mills to H. R. Denton dated November 16, 1982, which read as follows:

> There are no provisions for sampling in the event of loss of offsite power.
>
> Based on the original requirement of item II.B.3 of NUREG-0737, no provisions for sampling during loss of offsite power were 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 gride 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.

Probalistic risk assessment analysis done by TVA determined that the log normal distribution for the loss-of-power initiating event frequency calculated for Browns Ferry is 5.9×10^{-2} failures per site year (mean) with a variance of 3.2×10^{-3}. This distribution was calculated using both generic and plant specific data. This frequency is better than that reported in EPRI report NSAC-80 which was 8.8×10^{-2} loss of offsite power (LOOP) failure per site-year for the average U.S. site.

Using data from EPRI NSAC-80 and taking into account Browns Ferry plant specific actions, the probability for recovery of offsite power is 0.886 in the first 6 hours and 0.923 in the first 24 hours.

The post-accident sampling equipment obtained from Sentry was not required to be designed to safety-related standards and is similar to that being used by numerous other utilities. The Browns Ferry PASF is located in the turbine building which is also not designed to safety-related standards; therefore, we believe the risk represented by the above probability data is consistent with the overall postaccident sampling design and regulatory requirements considering that the system was not required to be built and designed to safety-related standards. Furthermore, there have been no LOOP failures at Browns Ferry to date which also supports this analysis.

Criterion 2

Provide a core damage estimate procedure including radionuclide concentrations and other physical parameters as indicators of core damage.

Response

The Core Damage Assessment procedure and supporting documents are to be prepared as a Browns Ferry Technical Instruction. A task force consisting of plant and central office personnel and a consultant has been organized and is actively engaged in the preparation of the procedure. The procedure is based on the General Electric BWR generic procedure. Completion and implementation of the approved procedure for application to interim post-accident response is planned for September 30, 1984. The interim damage assessment procedure will be modified as needed to function with the permanent Browns Ferry post-accident equipment and instrumentation when this facility is operational.

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 post-accident sampling system (PASS) instrumentation in an accident environment.

Response

Procedures were developed by the sampling equipment contractor, Sentry Equipment Corporation, through a subcontract with NUS (Development of Procedures and Analysis Methods for Post-accident Reactor Coolant for Sentry Equipment Corporation, Oconomowoc, Wisconsin, April 1981), using a test matrix whose chemical composition was designed to be representative of the Browns Ferry matrix following an accident. The range and accuracy of the equipment reported earlier was determined using this matrix.

NUS, in their subcontract to Sentry, also considered the effect of radiation on the PASS and concluded that the anticipated radiation levels will have no measurable effect on the accuracy of measurement and negligible effect on operating lifetime of individual components exposed to radiation.

Procedures for functionally testing the PASS and for retraining of the PASS chemistry technicians will be developed and in place by startup of the PASS system. Sufficient technicians and engineers to operate the PASS will be trained within one month of startup of the PASS. Every six months, one-half of the chemistry technician operators will both operate the PASS and actually sample the fluids, etc., in pertinent systems. At the same time, routine samples will be taken using normal procedures from the normal sample locations and compared with the corresponding PASS results. This will verify that the PASS is functioning correctly. Please note that by using this timetable, the operators will be retrained on a yearly basis at a minimum, and the PASS will be tested every six months at a minimum. TVA believes this program will provide adequate frequency for training and testing.

Criterion 11

Provide information demonstrating that the reactor coolant sampling locations are representative of core conditions.

Response

Two reactor coolant samples - one each from the 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 Licensing Review Group-II (LRG-II) has determined that samples obtained from the jet pump flow instrument system will provide representative core coolant samples for boiling water reactors under accident conditions. This determination is generally applicable to Browns Ferry.

In order to ensure that this sample location will be representative, sufficient core flow is required to circulate water from the core through the 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. For essentially all power levels, natural circulation will induce a substantial core flow rate. Thus the entire reactor water inventory would be circulated through the jet pumps in a relatively short time, ensuring that representative samples of core coolant will be available at the jet pumps. At very low power levels, it may be necessary to raise the reactor water level in order to induce natural circulation core flow.

For large break post-LOCA conditions when the reactor is either at reduced pressure or depressurized, either the Core Spray System (CSS) or the Residual Heat Removal System (RHRS) will maintain a level inside the core shroud of at least two-thirds core height. If the CSS is providing makeup, the flow is reverse through the core and the jet pump instrument tap is in direct communication with water moving through the core. If the RHRS (LPCI mode) is providing makeup, the flow is forward through the core and there has been considerable inventory exchange between the vessel and the suppression pool. The jet pump instrument top sample point, in this situation, is essentially the same point as the RHR pump discharge sample point referenced in LRG II position paper 1-CHEB and thus, serves the same purpose.

For post-LOCA situations in which shutdown cooling is established, the jet pump instrument top sample point is again essentially the same as the RHR pump discharge sample point, referenced in LRG II position paper 1-CHEB; therefore, the above reasoning still applies.