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

Docket No. 50-259
50-260
50-296

SEP 2 1982

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

Dear Mr. Parris:

SUBJECT: MULTIPLANT ACTION ITEM B-65, SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS IN BWR SCRAM SYSTEMS (NUREG-0803).

Re: Browns Ferry Nuclear Plant, Units 1, 2, and 3

Your letter of January 20, 1982 responded to our generic letter of August 31, 1981 on the above subject. Your submittal included a comprehensive evaluation for Browns Ferry of a scram discharge system pipe break scenario from initiation through mitigation activities. Your response was consistent with the guidance given in NUREG-0803 and was fully responsive to the request in our generic letter. In addition, you provided a plant-specific probabilistic risk assessment (PRA) of the postulated event from which you concluded that the sequence of events assumed in NUREG-0803 would not be a significant contributor to core damage and overall plant risk. Our evaluations support this conclusion.

Your letter requested that further regulatory actions on the subject issue be deferred pending completion of our severe accident sequence analysis (SASA) program which uses Brown Ferry as the model BWR. We fully appreciate TVA's participation in the SASA Program, the recently completed Interim Reliability Evaluation Program (IREP), which also used Browns Ferry Unit 1 as the model BWR, and other NRC research programs. We would not be able to conduct these programs without the extensive plant data and engineering support which you have provided. The event trees analyzed in the IREP assumed a range of pipe breaks; the results published in NUREG/CR-2802 tended to support the conclusion of your PRA. While the overall SASA program extends into 1986, the part pertinent to this issue is the detailed systems and building response analysis performed by ORNL based on a break in the scram discharge piping. This has been completed (NUREG/CR-2672) and also tends to support your PRA.

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To complete our review of the plant-specific analyses submitted by your letter of January 20, 1982, we need the additional information identified in the enclosure to this letter. We would appreciate a response within 60 days of receipt of this letter. If you have any comments or questions, please contact Dick Clark (301-492-7162).

This request for additional information is specific to one licensee. The reporting and/or recordkeeping requirements contained in this letter effect fewer than ten respondents; therefore OMB clearance is not required under P.L. 96-511.

Sincerely,

ORIGINAL SIGNED BY

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

cc: See next page

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Mr. Hugh G. Parris

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REQUEST FOR ADDITIONAL INFORMATION
MULTIPLANT ACTION ITEM B-65,
SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS
IN BWR SCRAM SYSTEMS (NUREG-0803)
BROWNS FERRY UNITS 1, 2 AND 3

Your response in the January 20, 1982 letter concerning the recommendations of NUREG-0803 is not complete. Provide the following additional information.

ASB 1 You have only partially addressed HCU equipment maintenance procedures with respect to possible loss of SDV integrity. Verify that all HCU and SDV system maintenance, surveillance, inspection and modification procedures provide guidance to plant personnel, as necessary, to ensure that SDV integrity is available at all times when it is required. (Refer to NUREG-0803 Section 3.2.1.8).

ASB 2 Verify that the temperature trip monitors for the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pump turbines are located sufficiently remote from the scram system and SDV to prevent initiation of turbine trip signals because of high ambient temperature resulting from the postulated scram system pipe break. Your analysis should account for the potential leakage path from the pipe break and air flow within the reactor building with normal ventilation systems in operation in order to determine if the temperature at the location of these monitors increases to the point where trip is initiated. (Refer to NUREG-0803 Section 4.3.1.3).

ASB 3 Your response implies that the availability of the other units' RHR systems is assumed in the event of HPCI and RCIC turbine trips caused by high ambient temperatures. Verify that depressurization to RHR system pressure can be accomplished under such a scenario, and that Technical Specification modifications are proposed to ensure the availability of another units' RHR system (for crosstie) under all operating modes of the nonaffected unit(s).

MEB 4 In your response to NUREG-0803 (letter from L. Mills to H. Denton dated January 20, 1982) it is not clear to what extent the SDV piping and supports will be seismically analyzed. Verify that the SDV piping header, the SDV small piping (less than 2 1/2 inches nominal pipe size), and the piping supports have been designed for seismic loadings.

Additionally, verify that the actual piping and support installation have been checked to assure the validity of the seismic analysis.

AEB 5 The radiological consequences of a scram discharge volume failure are analyzed generically in NUREG-0803 with respect to on-site occupational exposure to workers entering the scram discharge volume area, as well as offsite doses, and were found to be within the relevant guidelines for plants with General Electric Standard Technical Specifications (GE STS) for reactor coolant iodine concentration; while worker exposure and offsite consequences were found to exceed the guidelines for coolant iodine technical specifications similar to Browns Ferry.

The staff notes that the licensee has neither proposed to adopt the General Electric Standard Technical Specifications (GE STS) for reactor coolant iodine activity nor calculated occupational or offsite dose consequences for the scram discharge volume break, using the licensee's technical specifications in the analysis. Also, the staff finds that the licensee has not provided clear evidence to prove that the probability of the reactor coolant iodine concentration exceeding the GE STS is 0.001 per reactor year or less. As noted on p. 5-5 of NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," 1981, a scram discharge volume break which causes a rupture of the blow-out panels may result in excessive offsite doses in addition to causing an exposure problem for workers (for instances, those workers who might enter the scram discharge volume vicinity to manually close valves). Therefore, the licensee should either: 1) propose GE STS for reactor coolant iodine activity, or 2) provide the staff an evaluation of radiological dose consequences, using calculative methods describes in NUREG-0803, and demonstrate that the doses from this fission product release do not exceed occupational or offsite dose guidelines. The assumptions do not exceed occupational or offsite dose guidelines. The assumptions used should include the proposed or existing technical specifications on reactor coolant iodine concentration and an iodine spike caused by the accident.

MTEB 6 By letter dated January 20, 1982, the Tennessee Valley Authority made the following statement concerning the periodic inservice inspection and surveillance of the Scram Discharge Volume (SDV) System:

> "We are incorporating the SDV piping into the Browns Ferry inservice inspection (ISI)program in accordance with the applicable requirements of ASME Section XI Class 2. The inservice examinations of the SDV piping will begin during the second cycle of the first inspection interval."

To evaluate the adequacy of the inservice inspection and surveillance program for the SDV system, the additional information listed below is required.

a. What Code Edition and Addenda of Section XI are being used to perform the required examination and tests on the SDV System?

b. What are the pipe schedule numbers and diameters, and from what materials are the discharge header and instrument volume fabricated?

c. Has any portion of the discharge volume header and associated piping been exempted from examination by any of the criteria given in IWC-1220 of Section XI of the ASME Code? If so, please state which portion and the criteria used to establish the exemption.

d. Will any relief from Code requirements be requested in the inservice inspection program for the SDV System? If so, please state the relief and the basis for requesting it.

EQB 7 Identify all systems and equipment that would be used to detect a break and/or leak in the SDV system and state that this equipment is, or provide a commitment that it will be a) included in the environmental qualification program established in response to IE Bulletin (IEB) 79-01B, and b) qualified for service either in a 212°F and 100% humidity environment, or in a plant specific SDV break environment.

EQB 8 Identify all systems and equipment needed for the prompt depressurization function and all emergency systems and equipment, i.e., systems and equipment needed for mitigation of an SDV system pipe break, safe shutdown of the plant, and long-term core cooling.

State that this equipment is, or provide a commitment that it will be a) included in the environmental qualification program established in response to IEB 79-01B, and b) qualified for service either in a 212°F and 100% humidity environment, or a plant specific SDV break environment.

EQB 9 Identify any emergency systems and equipment that could be sprayed with water from dripping or splattering of overflow leakage down open stairwells following a break in the SDV system, and state that this equipment is, or provide a commitment that it will be a) included in the environmental qualification program established in response to IEB 79-01B, and b) designed to, or qualified to, operated with water impingement.

EQB 10 Identify all systems and equipment needed for mitigation of an SDV system pipe break that could be wet down from leakage through equipment hatches following the break, and state that this equipment is, or provide a commitment that it will be a) included in the environmental qualification program established in response to IEB 79-01B, and b) qualified for wet down by 212°F water.

EQB 11 If any equipment needed a) to detect a break and/or leak in the SDV system, b) for mitigation of an SDV system pipe break, c) for safe shutdown of the plant, and d) for long-term core cooling is not qualified for service in an environment that could exist following a break in the SDV system, provide justification for interim operation pending qualification of the equipment or replacement with qualified equipment.