TENNESSEE VALLEY AUTHORITY
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APR 18 1989

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) UNITS 1, 2, AND 3 - SUPPLEMENTAL RESPONSE TO NRC INSPECTION REPORT NOS. 50-259/88-07, 50-260/88-07, AND 50-296/88-07

This letter transmits TVA's revised responses for inspection report 88-07 items 4, 7, 22, 26, and H. These revisions are a result of discussions with the NRC staff during the Design Baseline and Verification Program inspection conducted at TVA's Knoxville office on February 27 through March 9, 1989.

Enclosure 1 contains the revised responses. A summary list of commitments is provided in enclosure 2.

If you have any questions, please telephone Patrick P. Carier, BFN, Site Licensing, at (205) 729-3570.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

C H Fox

C. H. Fox, Jr., Vice President and
Nuclear Technical Director

Enclosure
cc: See page 2

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Concern

4. Two pipe stress calculations (NI-367-7T and NI-270-1R) deviate from the requirement stated in the applicable design criteria and FSAR commitments. TVA should revise those calculations to meet the applicable design criteria and FSAR commitments.

Additional detailed related to this concern from Section 4.1.2.2 of report

- Pipe stress problem NI-367-7T.

The NRC team found that two separate values for the soil shear wave velocity are used in the calculation.

In the calculation for the soil strain, a value of the maximum ground velocity of 48 inches per second is used, which is not referenced, and which may not accord with licensing commitments.

- Pipe stress problem NI-270-1R

TVA, therefore, should access the seismic qualification documents for valves 2-FCV-70-313 and -47 to confirm that the valves are rigid, as modeled in the piping analysis, or revise the calculation in accordance with the requirement of Design Criteria Document BFN-50-C-7103 if the valves show a fundamental frequency of less than 20 H_z.

TVA Revised Response

Pipe stress problem NI-367-7T.

The calculation for the EECW piping in the RHR-EECW tunnels was revised to incorporate the appropriate soil properties and site conditions. A soil shear wave velocity (Vs) of 1000 ft/sec was used in the analysis which is consistent with Section C.2.1 of Appendix C to the FSAR. This velocity was considered more appropriate for the firm clay soil conditions around the RHR-EECW tunnels. The minimum Vs (250 ft/sec) represents an anomaly for the site. This anomaly is very soft soil that was only encountered in the area of the intake channel and subsequently excavated. Therefore, this minimum Vs was not considered in establishing a reasonable average Vs for analysis of the piping. A normalized site specific value of 17 in/sec (OBE) was used for the peak ground velocity.

Pipe stress problem NI-270-1R.

The evaluation of the impact of flexible valves on seismic qualification of piping (including piping analyzed in NI-270-1R) is being tracked by Condition Adverse to Quality (CAQR) BFP880121 R1. Based on similar evaluations for Sequoyah (SQN), it is expected that there will be few flexible valves identified on Browns Ferry and, where flexible valves are identified, the impact on piping seismic qualification will be minimal. The identification of nonrigid valves and the consideration of the impact of these valves on piping seismic qualification will be completed after unit 2 restart.

Concern

7. Calculations 48N1004-MS2-75-R5 and -12 indicate that certain steel members are overstressed. The overstressing was alleviated by a reanalysis of the STRUDL computer model in which the anchor at Node 23 was deleted. TVA should review this practice to ascertain the adequacy of the affected steel members.

TVA Revised Response

Calculation 48N1004-MS2-75-R12 (new ID CD-Q2303-882053) identifies the overstressing of certain components of the miscellaneous steel support framing for core spray supports R-12, H-27, and H-28. A modification was developed in the calculation to resolve the overstressed conditions and is analyzed in the calculation by incorporating changes to the STRUDL model. The modification is shown on drawing 48W1004-2 Revision 0, and field implementation is complete.

Calculation 48N1004-MS2-75-R5 (new ID CD-Q2303-882765) identifies the overstressing of certain components of the miscellaneous steel support framing for core spray supports R-5, H-7, and H-8. The overstressed condition is the result of the loading from a 10" diameter pipe anchor for the containment inerting system which is attached to the miscellaneous steel frame.

The overstressed condition was originally resolved by assuming the pipe anchor would be removed and was documented in the calculation as an unverified assumption. TVA has since decided to leave the anchor in place and therefore, has revised the calculation to include a modification to the miscellaneous steel support frame to alleviate the overstress condition. The effects of this over stress condition have been evaluated and determined to not affect system operability for the pre-restart plant conditions.

However, to ensure long term post-restart operability the modification is required and will be completed before restart of unit 2.

Concern

H. Questionable Methodology of Containment Electrical Penetration Calculations Identified by TVA Engineering Assurance Audit

The calculations use the approach given in ICEA Standard P32-382-1969 for generally predicting the short circuit thermal capacity of insulated cables, instead of the penetration manufacturer's short circuit test data. The calculations do not sufficiently demonstrate that the general ICEA formula gives conservative results compared with actual test data, as TVA and Bechtel maintained when the team called the issue to their attention.

The criterion for acceptable penetration protection adopted in the calculations is that overcurrents shall be interrupted, before thermal damage to the penetration, by the first-line circuit breaker or fuse, rather than the backup protective device as recommended by NRC Regulatory Guide (RG) 1.63. In its initial oral response, TVA pointed out that Browns Ferry has never been committed to RG 1.63, which was issued after the plant was built. However, containment penetration integrity is critical to the "defense-in-depth" principle, and experience at other plants (e.g., Sequoyah) suggests that adequate protection by backup devices can be obtained (without hardware modifications) throughout most, if not all, of the range of possible overload and short circuit currents by appropriate protective relay and circuit breaker trip settings. Therefore, TVA should ensure that penetration protection is a high-priority objective of the protective device coordination analysis now in progress.

TVA Revised Response

Although BFN is not committed to RG 1.63, TVA has ensured that containment penetration integrity is being maintained. The penetration protective device coordination analyses have been completed for every circuit which passes through a containment penetration. This analysis used accepted engineering practices to ensure that the thermal capability of the penetration was adequately protected thereby ensuring containment penetration integrity. Extremely conservative assumptions were utilized to ensure adequate safety margins. The method employed calculated the maximum allowable short circuit thermal capacity for different conductor sizes using ICEA P32-383-1969. This formula calculates the current withstand capability as a function of the I^2t value and the temperature differential from the initial conductor temperature to the maximum allowable final temperature. This formula was very conservatively applied to ensure adequate margin because it assumes no heat transfer during the short circuit condition.

The electrical penetrations were reviewed for short circuit and short time fault currents and total heat load in each penetration module to ensure that the nozzle/concrete interface temperature and the thermal capability of the individual modules were not exceeded. It must be noted that in all cases, even with conservative assumptions, the calculated heat load of the penetration modules is well below the rated maximum watts per foot values furnished by the penetration manufacturers. To calculate the maximum heat load, the continuous current was assumed equal to the rated trip value of the protective device for each conductor which is again very conservative. Additionally, an estimated current was assumed for the spare conductors and all conductors were assumed energized. It shall be noted that in all cases the penetration conductors are of equal or larger wire size than the associated field cables. Therefore, continuous overload capability of the conductors was not considered a concern.

Concern

22. A pressure transmitter (2-PT-1-72) was downgraded to a nonsafety-related status. This downgrading violates IEEE Standard 279-1971 and NRC Branch Technical Position ICSB-26, "Requirements for Reactor Protection System Anticipatory Trips," (NUREG-0800). TVA should reexamine the other 69 unit 2 instruments to ensure that no other problems exist.

TVA Revised Response

Pressure transmitters 2-PT-1-72, -76, -82, and -86 monitor main steam line pressure at the inlet to the main turbine. The transmitters and their associated switches are utilized to detect abnormal transient events such as inadvertent opening of the turbine bypass valves when the reactor is operating above 825 psig pressure. Sudden depressurization is the basis for closure of the main steam isolation valves (MSIV) to prevent undesirable transients on the reactor internals. The four transmitters and switches operate in 1-out-of-2 twice, fail-safe logic.

This corresponds to a standard GE BWR design for plants of the Browns Ferry vintage. This design was established prior to issuance of IEEE Standard 279-1971. However, the design was evaluated against and does comply with the intent of IEEE Standard 279-1971. This evaluation is documented in General Electric Topical Report NEDO-10139 dated June, 1970.

Branch Technical Position ICSB-26 addresses reactor protection system (RPS) anticipatory trips and establishes the position that all inputs to the RPS must meet the requirements of IEEE Standard 279-1971; i.e., the effects of credible faults or failures in these anticipatory trip functions must not be capable of propagating back to the RPS and degrading its performance or reliability. The pressure transmitters provide input to the primary containment isolation system (PCIS) logic cabinets via analog trip units (ATU). The ATUs are a safety-related interface and meet the intent of IEEE Standard 279-1971. The TVA quality information release (QIR) which downgraded the QA status of the subject instruments has been revised to upgrade the instruments and require accuracy calculations for the PTs. In addition, TVA has revised the main steam system design criteria (BFN-50-7001) to ensure that the quality of pressure transmitters 2-PT-1-72, -76, -82, and -86 is maintained as safety grade.

The other sixty nine (69) instruments mentioned in this concern have been evaluated by the Design Baseline Verification Program (DBVP) as part of its effort to identify essential calculations. None of these instruments were identified as requiring instrument accuracy calculations prior to unit 2 restart or as providing input to the protection systems (RPS/PCIS or ECCS). These instruments are not required to mitigate design basis accidents, abnormal transients, or special events.

Concern

26. The NRC team reviewed Ebasco's setpoint calculation for Reactor Building Closed Cooling Water (RBCCW) System time delay relays (ED-Q2070-88069). The calculation was listed as being essential (i.e., it addresses a safety-related component) but the calculation (Pages 1, 5, and 8) stated that the RBCCW pumps and the time delay relays performed no safety-related function. This statement appears to be inconsistent. The time delay relays do perform a safety-related function as the RBCCW system is converted from two pump to one pump operation. The Ebasco calculation should be modified to clarify that postaccident RBCCW flow is not safety-related but that the electrical controls needed to reconfigure the system are safety-related.

TVA Revised Response

The previous TVA response stated that the safety-related function of these relays was to preclude an unanalyzed loading of the standby diesel generators. Since this response was originally submitted, TVA has completed an essential calculation (ED-Q2000-88069) which considers the loading of both RBCCW pumps onto the standby diesel generators at the same time. The calculation clearly indicates that there are no degrading effects to the diesel generators if this condition should occur. Thus, if the relays misoperated such that both pumps were loaded at the same time, there would be no adverse effect on the diesel generator loading as previously assumed.

TVA and NRC personnel have reviewed the original General Electric (GE) design specification (GE Design Specification 22A1110 Table 1) for the RBCCW system. This specification requires that at least one RBCCW pump start to protect critical equipment upon loss of AC power. The specification does not define this to be a safety-related function. TVA has further amplified the requirement in its design criteria (BFN-50-7070) where the function is defined to be required but not for safety purposes. The original intent of the GE design specification was to ensure that critical equipment was not damaged due to overheating during a loss of AC power event. Therefore, the requirement for RBCCW pump operation is based upon operational/financial reasons rather than safety considerations.

TVA concludes that the subject time delay relays are Class 1E because of their association with vital diesel generator busses and not because of any safety function which they perform. The calculation in question provides the necessary information to ensure that the relays are properly set for operation of the pumps. Since the calculation correctly identifies the function of the relays, no other revisions to the calculation are planned based upon the foregoing discussion.

Enclosure 2

Summary List of Commitments

1. The accuracy calculations for pressure transmitters 2-PT-1-72, -76, -82, and -86 will be completed before unit 2 restart.

The following commitments are addressed by this submittal and in the Nuclear Performance Plan Volume 3 commitments for DBVP post-restart calculation review and restart miscellaneous steel framing modifications. As such, TVA will track, followup, and ensure completion in accordance with Volume 3 and does not consider these as new commitments.

RESTART

2. The miscellaneous steel support framing for core spray supports R-5, H-7, and H-8 will be modified to adequately handle the loads from the pipe anchor on the 10-inch diameter containment inerting pipe.

POST-RESTART

3. Complete the identification of nonrigid valves and evaluate their impact on seismic qualification of piping.