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SUBJECT: Forwards revs to description of util program for ensuring that any seismic Class II item does not degrade integrity of any seismic Class I item. Written statement documenting acceptability of program requested.

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Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 2 - SEISMIC CLASS II FEATURES OVER SEISMIC CLASS I FEATURES

This letter revises the description of the BFN program for ensuring that any seismic Class II item does not degrade the integrity of any seismic Class I item. This letter supplements the information provided by Section III.3.11 of revision 1 to the BFN Performance Plan which was transmitted by letter from S. A. White, dated July 1, 1987.

Enclosure 1 to this letter describes the BFN program for resolving this issue. Enclosure 2 provides the BFN evaluation criteria for seismic-induced spray hazards. TVA requests your review of this program and the issuance of a written statement documenting the acceptability of the program.

Please refer any questions regarding this submittal to M. J. May, Manager, BFN Site Licensing, (205) 729-3570.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. Gridley, Director
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Regulatory Affairs

Enclosures
cc: See page 2

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U.S. Nuclear Regulatory Commission

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ENCLOSURE 1

BROWNS FERRY NUCLEAR PLANT UNIT 2
CLASS II FEATURES OVER CLASS I FEATURES

Background

The Browns Ferry Nuclear Plant (BFN) Final Safety Analysis Report (FSAR) states that any item designated as Class II shall not degrade the integrity of any item designated as Class I. BFN non-Class I systems and components were often installed by field routing procedures with commercial grade hardware, following normal industrial practice, as was common in plants of BFN's vintage.

Definition of Problem

Significant Condition Report (SCR) BFNMEB8605 R1 states that objective evidence has not been found to indicate that engineering evaluations were performed to ensure that nonsafety-related components cannot degrade the integrity of safety-related components due to a seismic event. This SCR was written to serve as a single collective SCR on the seismic systems interaction subject.

The Browns Ferry Nuclear Performance Plan, Volume 3, states that a program to address the seismic systems interaction issue will be developed before restart of BFN unit 2 taking into account and consistent with the Unresolved Safety Issue (USI) A-46 program. Seismic systems interaction is included in the A-46 review to the extent that equipment within the scope must be protected from seismically induced physical interaction with all structures, piping, and equipment located nearby. Because seismic-induced fluid spray is currently not addressed by Seismic Qualification Utilities Group (SQUG) and the A-46 program, prerestart efforts will focus on the water spray issue.

A related study was conducted at BFN in 1973-1974 to assess dynamic and environmental effects resulting from postulated piping failures outside of primary containment. The study focused on the ability to place and maintain the plant in a cold shutdown condition. Certain nonqualified piping was included. Modifications were made to protect several shutdown components due to potential water spray hazards as the result of this study. The seismic induced spray program will utilize these results and consider plant modifications since the study timeframe.

TVA Position

Systems interaction in nuclear power plants was initiated by the Advisory Committee on Reactor Safeguards (ACRS) in 1974 and is currently being addressed by NRC as a USI. USI A-17 as defined by NRC task action plan encompasses functional, spatial, and human-induced interactions. Seismic system interactions covered by A-17 include Class II over Class I (II/I) falling, seismic impact, and seismic-induced spray/flooding events.

Spatial seismic interactions (i.e., falling and impact) are included in the USI A-46 program scope. BFN is an A-46 plant, and TVA is an active member of the SQUG. Evaluations for spatial seismic interactions will be conducted in a timeframe consistent with SQUG member utilities as committed in TVA's letter dated December 1, 1987.

The seismic-induced spray evaluation program is currently underway at BFN. The schedule has been developed so that walkdown screening of plant areas within the program's scope, evaluation of outliers, and plant modifications will be completed before startup of BFN unit 2.

Description of Program

TVA's plan for resolution of seismic systems interaction issues at BFN includes ongoing work as well as work that will be done after unit 2 restart. Seismic spatial interactions, such as falling and impact, will be conducted during A-46 program plant reviews. The seismic-induced spray evaluation program is ongoing. The evaluation criteria, which includes screening and outlier evaluation acceptance criteria, is included as enclosure 2 to this letter.

Criteria development tasks included detailed review of experience data on the performance of piping systems and fluid pressure boundary components in past major earthquakes. The damage data were categorized and evaluation acceptance criteria were established to address the failure modes identified by the experience data. The acceptance criteria are based on shake table test data, component test data, and engineering calculations.

Walkdowns are conducted on an area-by-area basis; 100 percent of each area is screened. Items not meeting the screening acceptance criteria are identified as outliers, then evaluated in more detail. Configurations not accepted will be analyzed for consequence to safety system function. Required modifications will be made to preclude adverse interactions between Class II and Class I features. All required modifications will be installed before restart of BFN unit 2.

Requested Action

TVA requests NRC approval of the criteria for seismic-induced water spray evaluations. The plant screening acceptance criteria is based on seismic experience data in conjunction with test data and analytical techniques. The plant screening will be conducted by trained and experienced degreed engineers.

TVA also requests NRC approval of the program schedule. The program schedule was developed assuming deferral of spatial seismic interaction evaluations until after plant restart. The evaluation of spatial seismic interactions will be included in the A-46 review program consistent with the other A-46 review plants.

Conclusion

This program will ensure that the integrity of Class I systems and components are not degraded by seismic-induced interactions with Class II features. This program satisfies FSAR Appendix C commitments regarding seismic systems interaction.

ENCLOSURE 2

EVALUATION CRITERIA FOR SEISMIC-INDUCED SPRAY HAZARDS
BROWNS FERRY NUCLEAR PLANT

1.0 PURPOSE

The purpose of this instruction is to provide engineering guidelines for evaluation of potential seismic-induced fluid spray hazards that may arise from failure of non-seismic Class I fluid pressure boundaries.

2.0 SCOPE

These guidelines apply to non-seismic Class I piping and component fluid pressure boundaries whose failure may result in interaction with fluid spray sensitive Class I components and degradation of required safety functions.

3.0 PLANT SCREENING ACCEPTANCE CRITERIA

A walkdown shall be conducted of piping and components within scope to identify credible water spray hazards as described below. Items not meeting the screening acceptance criteria shall be evaluated for acceptability in more detail as described in section 4.

3.1 Equipment Anchorage Screening

Unanchored, unrestrained, and inadequately anchored equipment components that provide some form of a fluid pressure boundary shall be screened.

3.2 Piping System Screening

Piping systems shall be screened for position retention and pressure retention capabilities, as follows.

3.2.1 Position Retention Screening

Piping spans shall meet B31.1 (Reference 1) dead load support criteria. Supports shall be screened for a dead load factor of safety of 2.0.

Short rod hangers with fixed-end connection details such as rods threaded directly into shell anchors shall be screened for fatigue failure. Plant screening shall consider component ultimate test data mean values less one standard deviation on number of cycles to failure for DBE time history response. Screening performance parameters shall

include rod connection detail, rod diameter, rod length, and supported weight.

Plant screening shall identify eccentric details that may induce significant prying loads on anchorages. Support configurations which may exhibit nonductile behavior for horizontal seismic induced motion shall be screened.

3.2.2 Pressure Retention Screening

Potential seismic anchor movement and seismic interaction scenarios shall be screened, utilizing conservative, approximate deflection estimation methods. Seimic proximity interactions with fragile pressure boundary appurtenances such as taps, vents, drains, and instrumentation shall be screened. Deflections shall be estimated considering first mode response of approximate simple pipe spans and 5% damped building floor response spectra. Seismic anchor movement induced piping stresses shall be screened using flexibility charts for $2.4S_h$.

Building deflection estimates for evaluation of seismic anchor movement induced by differential building motion shall be obtained from the building floor response analysis and relative motions shall be combined by absolute summation.

Mechanical pipe coupling details such as victaulic and bell and spigot, without idependent support, shall be screened. Piping of nonductile material such as PVC and cast iron shall be screened. Severe corrosion shall be screened.

4.0 OUTLIER EVALUATION CRITERIA

Configurations identified by plant screening shall be evaluated for acceptability as described below. If a configuration is not accepted, an evaluation may be conducted to assess the effect on system safety function.

4.1 Equipment Anchorage Acceptance Criteria

Equipment anchor bolt seismic demand and dead load shall be accepted based on allowable loads derived from ultimate test mean/2 values for wedge bolts and ultimate test mean/4 values for shell anchors. Seismic demand shall be estimated considering 5 percent damped building floor response spectra for the vertical and horizontal earthquake components. Piping attached to equipment components with flexible support systems shall be evaluated for seismic anchor movement.

4.2 Piping System Acceptance Criteria

Acceptance criteria for piping and pipe support systems are provided below.

Pipe system analysis for seismic demand shall be estimated considering 5 percent damped building floor response spectra for the vertical and horizontal earthquake components. Piping systems analyses may consider the realistic effects on non-linear behavior due to design features and phenomena such as proximity/impact with other non-seismic Class I systems, interferences and small clearances to stiff structures, geometric restoring forces, wall penetration sealants, and support ductile behavior.

4.2.1 Pipe Position Retention Acceptance Criteria

Pipe support anchor bolt loads shall be accepted based on allowable loads derived from ultimate test mean/2 values for wedge bolts and ultimate test mean/4 values for shell anchors.

Piping supports that may exhibit nonductile behavior shall be accepted based on stress allowables or test data as follows. Acceptable flexural/and tensile stresses shall be the lesser of $0.7S_u$ and $1.2S_y$. Acceptable shear stresses shall be the lesser of $0.42S_u$ and $0.72S_y$. Acceptable bolt stresses shall be the greater of $0.7S_u$ and minimum specified S_y. Acceptable loads based on test data shall consider mean less one standard deviation capacity.

Pipe supports not meeting the above criteria may be accepted if adjacent supports and resulting pipe span can resist dead load with a factor of safety of 2.0. Inplant considerations regarding other consequences of support failure such as falling and excessive deflection shall be made when using this provision.

4.2.2 Pipe Pressure Retention Acceptance Criteria

Acceptable pipe stresses induced by DBE inertial loads, seismic anchor movements, dead load and pressure shall be $2S_y$. In cases where piping stress exceeds $2S_y$, an augmented fatigue evaluation may be utilized (reference 2).

5.0 DOCUMENTATION

Engineering evaluations shall be performed in accordance with the applicable requirements of 10 CFR 50 Appendix B.

6.0 REFERENCES

1. ASME Code for Pressure Piping, B31, An American National Standard, ANSI/ASME B31.1--1983 Edition.

2. NUREG/CR-3243, Comparisons of ASME Code Fatigue Evaluation Methods for Nuclear Class I Piping With Class 2 or # Piping, 1983.