Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609

February 4, 2000

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 2 AND 3 - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING TECHNICAL SPECIFICATIONS (TS) CHANGE NO. 399 - INCREASED MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE RATE LIMITS AND EXEMPTION FROM 10 CFR 50 APPENDIX J - REVISED TS PAGES FOR INCREASED MSIV LEAKAGE LIMITS -(TAC NOS. MA6405, MA6406, MA6815 AND MA6816)

This letter responds to the November 23, 1999, Request for Additional Information (RAI) regarding (TS-399) change request 399. TS-399, which was submitted on September 28, 1999, proposes changes to the Unit 2 and 3 TS to increase the allowable leakage rate criteria for the MSIVs. In addition, in the September 28, 1999, submittal, TVA requested exemption to specific portions of 10 CFR 50, Appendix J to allow the exclusion of MSIV leakage from the summation of containment leak rate test results.

Enclosure 1 of this letter provides the TVA response to the nine RAI questions. Enclosure 2 contains supporting calculations for the condenser seismic assessment associated with RAI Item 7.

Enclosure 3 provides additional details regarding RAI Item 8 which addresses specific NRC staff questions on dose analysis methods. Additionally, as discussed in Enclosure 3, TVA has performed specific MSIV dose calculations rather than using extrapolation factors for the MSIV leakage. This revised

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analysis resulted in a reduction of the requested MSIV allowable leakage rate requested in the September 28, 1999 letter. Accordingly, a revised change request is provided in Enclosure 4. Enclosure 5 contains marked-up copies of the appropriate pages from the current Units 2 and 3 TS showing the proposed revisions.

The revised pages provided in Enclosure 5 do not alter the original determination that there are no significant hazards considerations associated with the proposed changes, nor does it alter the originally submitted Environmental Assessment and Finding of No Significant Impact provided by the September 28, 1999 letter. The BFN Plant Operations Review Committee and the BFN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of BFN Units 2 and 3 in accordance with the proposed change will not endanger the health and safety of the public.

Pursuant to 10 CFR 50.12, an exemption to 10 CFR 50, Appendix J containment leakage requirements was requested in the September 28, 1999, submittal which would allow exclusion of the MSIV leakage from the summation of containment leak rate test results. This exemption request supports the TS change to increase the MSIV leakage criteria and is still being requested. Additional information regarding the need for the exemption is provided in the response to RAI Item 9.

Enclosure 6 provides a listing of commitments made in this submittal. If you have any questions, please contact me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures
cc: see page 3

Enclosures
cc (Enclosures):

Mr. Paul Frederickson, Branch Chief
U.S. Nuclear Regulatory Commission
Region II
61 Forsyth Street, S.W.
Suite 23T85
Atlanta, Georgia
30303

Mr. William O. Long, Project Manager
U.S. Nuclear Regulatory Commission
One White Flint, North
11555 Rockville Pike
Rockville, Maryland 20852

NRC Resident Inspector
Browns Ferry Nuclear Plant
10833 Shaw Road
Athens, Alabama 35611

ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

TECHNICAL SPECIFICATIONS (TS) CHANGE - 399
INCREASED MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
DATED NOVEMBER 23, 1999

Below are responses to the nine NRC items from the subject RAI on TS-399. The NRC questions are repeated along with the TVA responses for each item.

TS-399, which was submitted on September 28, 1999, proposed changes to the Unit 2 and 3 TS to increase the allowable leakage rate criteria for the MSIVs and requested an exemption to specific portions of 10 CFR 50, Appendix J to allow the exclusion of MSIV leakage from the summation of containment leak rate test results.

Enclosure 2 contains supporting calculations for the condenser seismic assessment associated with RAI Item 7. Enclosure 3 provides additional details regarding RAI Item 8 which addresses specific NRC staff questions on dose calculation methods. Commitments made in these responses are presented in Enclosure 6.

NRC ITEM 1

Section 5.2 of the March 3, 1999, safety evaluation for NEDC-31858, states that a secondary ALT path to the condenser, having an orifice, should exist. Your application states that in the event that FCV-1-58 were to fail to open, the leakage flow would split, with part of the flow going to the condenser via a 0.1875 inch diameter orifice in a normally open bypass around FCV-1-58, and the remainder going to the condenser via normal leakage paths through the main steam stop/control valves and through the high pressure turbine. It is noted that NEDC-31858 para. 6.1.1(2) states that the ALT flow path should, based on the radiological dose methodology, be at least 1-square inch in internal cross sectional area. Please describe the effect on offsite dose and control room habitability, of this single failure. In particular, will dose consequences remain acceptable in the event of single-failure of FCV-1-58?

TVA RESPONSE TO ITEM 1

The BFN alternate leakage treatment (ALT) flow path is shown in Figure 3-1 of Attachment 4 of the September 28, 1999, TS-399 submittal. The ALT path is from the outboard side of the MSIVs through Flow Control Valve (FCV)-1-58 to the condenser and satisfies the sizing requirements of NEDC-31858 paragraph 6.1.1(2) which states that the ALT flow path should, based on the radiological dose methodology, be at least 1 square inch for internal cross sectional area. The orificed bypass path around FCV-1-58 shown in Figure 3-1 addresses Section 5.3 of the NRC safety evaluation dated March 3, 1999, which states that a secondary path to the condenser, having an orifice, should exist. This secondary path is considered a contingency alignment in the event of the unlikely failure of FCV-1-58 and is not sized to meet the 1-inch path provision discussed in the NEDC specified for the credited ALT path. Moreover, NEDC-31858 does not prescribe that a secondary ALT path be available which is fully redundant to the credited ALT path in terms of sizing.

The failure of FCV-1-58 is unlikely to result from a loss of offsite electrical power. For example, 2-FCV-1-58 is powered by 480-V Reactor Motor Operated Valve (RMOV) Board 2C. RMOV Board 2C is normally aligned to 480-V Shutdown Board 2B which is Division II essential power. The alternate feed to RMOV Board 2C is 480-V Shutdown Board 2A which is Division 1 essential power. These 480-V Shutdown Boards have separate Emergency Diesel Generators (EDGs) as back-up power supplies through their respective 4160-V Shutdown Boards.

If the normal feeder (480-V Shutdown Board 2B) to RMOV Board 2C is lost, it can be transferred to its alternate power supply (480-V Shutdown Board 2A) by remote breaker operation. Therefore, it is an easy operation to transfer 480-V RMOV Board C to its alternate emergency power supply. As noted above, the two 480-V Division I and II Shutdown Boards both have their own (separate) EDG power supplies. The power arrangement for 3-FCV-1-58 is similar. Refer to the Final Safety Analysis Report (FSAR) Figures 8.4-1.b and 8.4.2 for a diagram of this electrical distribution arrangement. For reasons stated above, it is highly unlikely that power will not be available to FCV-1-58 in the event of loss of offsite power.

As discussed in the response to NRC RAI item 4, FCV-1-58 will be periodically tested as part of the Inservice Test Program (IST) to ensure the valve is operable. In addition, the functionality of the ALT path has been made highly reliable through the efforts to ensure the line is seismically rugged as discussed in the TS-399 submittal.

TVA considers that the proposed ALT path configuration using FCV-1-58 is consistent with the NEDC criteria to provide a reliable ALT path. TVA is also providing a secondary orificed contingency path in the unlikely event of a failure of FCV-1-58. With the 0.1875-inch orificed path around FCV-1-58, it is calculated that the majority of MSIV leakage would still be directed to the condenser with a smaller remainder through the closed Main Steam Stop/Control Valves to the high pressure turbine. The Main Steam Stop/Control Valves are currently in the preventative maintenance program. As such, one Main Steam Stop and one Control valve is refurbished each outage. Consequently, the Main Steam Stop/Control valves are refurbished once every 96 months. These valves are tested each refueling outage for leak tightness and have historically been highly reliable. Therefore, even in the unlikely event of the failure of FCV-1-58, the bulk of the MSIV leakage would still be routed to the condenser, hence, reducing potential control room and offsite doses.

NRC ITEM 2

Your application indicates that sealing steam supply valve, PCV 1-147, will be modified so that it fails closed instead of open. Assuming that fails-open was the original "safe" fail position, please confirm that the new fail position will not adversely affect the capability to mitigate design basis accidents and other postulated events.

TVA RESPONSE TO ITEM 2

Pressure Control Valve (PCV) 1-147 is used during reactor startup to provide steam seals to the main turbine. At higher reactor powers (above approximately 25% power), the BFN turbine is self-sealing and PCV-1-147 is maintained closed by the valve controller.

The existing failure position (open) of PCV-1-147 presents an operational problem in potentially overwhelming the capacity of the seal steam subsystem to "unload" (self-regulate) the seal steam header pressure. Therefore, in the event of an "open" failure, to continue normal power operation, it would likely be necessary to supplement the automatic seal steam unloader valves, PCV-1-148A and B, by opening the manual unloader valve, FCV-1-149, and/or by closing the high pressure steam supply valve, FCV-1-146.

The new failure position (closed) would present an operational problem only at low reactor powers (below approximately 25 percent power). This would result in a slow loss of condenser vacuum if not corrected. Low seal steam pressure is alarmed in the control room and the associated Alarm Response Procedure directs the operator to open the steam seal bypass valve (FCV-1-145) to restore steam seal pressure. This is a simple task that can be performed from the main control room, and there is ample time to respond before condenser vacuum is lost.

From the above discussion it is seen that the failure of PCV-1-147 to either an open or closed position results in a operational problem dependent on the power level of the reactor. Either end state is readily remediable by operator action. Since the reactor is almost always at high power except for brief periods of start-up and shutdown operations, the new fail-closed mode is preferable from an operational and safety point of view.

PCV-1-147 is not a safety-related valve and its operation is not currently assumed in the mitigation of design basis accidents (DBA) or transients. Therefore, it is concluded that the new fail-closed mode to maintain the ALT boundary is satisfactory and does not adversely affect normal reactor operation.

NRC ITEM 3

Your application indicates that check valves are to be added to preheater steam lines to ensure ALT boundary integrity. Please describe any proposed measures surveillance tests for these valves. Does the use of these valves create a single-failure concern?

TVA RESPONSE TO ITEM 3

The subject check valves will be located in the steam supply lines to the Offgas Preheaters as shown in Figure 3-1 of Attachment 4 of the September 28, 1999, TS-399 submittal. These new valves are within the current scope of the BFN American Society of Mechanical Engineers (ASME) Inservice Test (IST) Program. These check valves will be inspected and tested in accordance with the requirements for ASME Class 2 valves. As such, these valves are nominally required to be exercised to their safety position (closed) once each quarter.

If quarterly or cold shutdown testing is not practical, the IST program allows that check valves may be disassembled and inspected each refueling outage as an alternative. TVA has concluded that it is not practical to exercise these valves on a quarterly or cold shutdown basis. Position 2 of Generic Letter (GL) 89-04, Guidance on Developing Acceptable Inservice Testing Programs, allows identical check valves to be grouped together (four valves per group maximum) and disassembled on a rotating basis (one valve each refueling outage) when normal testing is not practical. Therefore, Section XI surveillance testing will consist of disassembly and inspection on a rotating basis (one check valve each refueling outage) in accordance with Position 2 of GL 89-04. The valves will also be verified to open after disassembly and inspection by proper operation of the Offgas Preheaters.

Regarding single failure considerations, these check valves are particularly well suited for this application of providing a boundary for the ALT path. They are highly reliable and provide positive isolation through their design. Alternate configurations such as fail-closed air operated valves and motor operated valves were considered, but were rejected in favor of the use of check valves.

Use of check valves is considered more reliable than air valves since operation of the check valve depends only on the system process (differential steam pressure), and not the external devices such as controllers, solenoids, switches, etc. In addition, a fail-close pneumatic valve would have a potential to negatively interfere with normal operations. Motor operated valves (MOV) would be dependent on electrical power availability and relay logic. Hence, in this application, the use of check valves is considered the best choice of components which minimizes potential interferences with plant operation while providing high reliability for retention of the ALT path boundary. As noted above, these check valves are within the scope of the BFN IST program and will be inspected and tested as described to provide assurance of proper component operation.

NRC ITEM 4

In allowing nonseismic piping to perform an engineered safety feature (ESF) function, it is expected that licensees will include the ALT system in the ASME Section XI inservice inspection (ISI) and inservice testing programs, and perform augmented ISI and motor-operated valve inspections in a manner consistent with ongoing ASME and approved risk-based programs applicable to ESF piping systems. Please confirm if this is your intention.

Also, your application states that the most limiting single active failure would be failure of valve FCV-1-58 to open. Please describe any augmented periodic testing (i.e., Generic Letter (GL) 89-10/GL 96-05 diagnostics) that will be performed on this valve.

TVA RESPONSE TO ITEM 4

The piping and components within the boundaries of the MSIV ALT path are considered to be within the scope of the BFN Section XI

IST and ISI programs, and, accordingly, will be inspected and tested in accordance with the IST/ISI programs. Additional detail is provided below for certain aspects of the program pertaining to the RAI questions.

The IST program will test the power operated valves within the ALT path boundary on a periodic basis. The specific test requirements will be based on the function of the individual valve (e.g., passive versus active). Testing of the check valves (considered active check valves) to the offgas preheaters is discussed in RAI Item 3. Certain valves that serve as part of the ALT path boundary (for example, Main Turbine Stop and Bypass valves) are specifically excluded from the IST program in accordance with Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants. Currently, some of these excluded valves are tested during power operations to ensure their functionality and are in the preventive maintenance program for periodic refurbishment. These valves will be included as part of the IST program, but as non-Code valves.

The ALT path boundary piping does not meet the criteria for inclusion in the augmented Intergranular Stress Corrosion Cracking weld inspection program. This piping is, however, part of the Flow Accelerated Corrosion (FAC) program which periodically monitors pipe wall thickness degradation.

FCV-1-58 was considered for inclusion in the augmented MOV test programs such as GL 89-10, Safety-Related Motor-Operated Valve Testing And Surveillance, and GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves. The design basis for establishing the ALT path is a Loss-of-Coolant Accident (LOCA) with assumed major core damage and the MSIVs closed. With the MSIVs closed, the ALT path boundary is physically isolated from the reactor vessel and primary containment except through leakage through the MSIVs. The ALT path piping will depressurize through the orifice around FCV-1-58.

In order to establish the ALT path, FCV-1-58 will not have to open against a large differential pressure and the post-accident system conditions will be less severe than the conditions which the FCV-1-58 valve would experience during IST testing during normal power operations (with full reactor pressure). Therefore, the FCV-1-58 will not be included in the GL 89-10/96-05 MOV program since the periodic IST program testing on this valve is considered to be adequate to ensure its functionality.

NRC ITEM 5

Section 4.1.2 of your EQE Report identifies the load combinations and stress allowables utilized in seismic assessments. Please provide a discussion of the extent to which the criteria used are consistent with the licensing basis requirements for other engineered safety features.

TVA RESPONSE TO ITEM 5

The load combinations and stress allowables utilized in the seismic assessments for the resolution of outliers and the evaluation of ALT piping, related components, and supports as presented in Section 4.1.2 of the EQE Report (Attachment 4 of the September 28, 1999, TS-399 submittal) are consistent with plant licensing basis requirements used to address Class II piping, and pipe supports and components for pressure boundary integrity and position retention at BFN. These seismic evaluation criteria are contained in TVA Design Criteria BFN-50-C-7306, Qualification Criteria for Seismic Class II Piping, Pipe Supports and Components, which is Reference 9 of the EQE report. The objective of the subject seismic assessments was to provide assurance that the ALT pathway would maintain pressure boundary integrity and would not be adversely affected by such factors as (1) differential displacements of structures, equipment, and piping (2) pipe support integrity issues and (3) seismic interaction issues such as the impact of piping with equipment, structural features, and other piping.

Additionally, valves that are classified as active in establishing the ALT path must be functional following the Design Basis Earthquake (DBE) and were evaluated in accordance with the General Implementation Procedure (GIP) methodology as referenced in Section 4.1.2 of the EQE report. Qualification in accordance with GIP provides reasonable assurance the required valves will be functional.

The loading combinations and stress allowables utilized in the design or assessment of Class I systems (ESFs - piping, pipe supports, components, etc.) are described in Appendix C of the BFN FSAR, Structural Qualification of Subsystems and Components. These requirements are specified in TVA Design Criteria BFN-50-C-7103, Structural Analysis and Qualification of Mechanical and Electrical Systems, for Class I piping and tubing, in TVA Design Criteria BFN-50-C-7107, Design of Class I Seismic Pipe And Tubing Supports, and in TVA Design Criteria BFN-50-C-7105, Pipe Rupture, Internal Missiles, Internal Flooding, Seismic Equipment Qualification and Vibration Qualification of Piping, for Class I and Class II equipment. The load combinations and stress allowables for ESFs were developed to assure not only pressure boundary integrity and position retention, but also for full functionality of equipment following a DBE.

In summary, the load combinations and stress allowables used for the ALT path seismic assessment discussed in the EQE Report are based on assuring that the system will maintain pressure boundary integrity and position retention and, in some cases for valves, maintain functionality. Since the main steam piping system housed in the Turbine Building was not originally designed to include seismic loading, a seismic verification walkdown to identify potential piping concerns was performed of the leakage pathway to provide assurance that pressure boundary integrity and position retention would be maintained. The load combinations and stress allowables in Section 4.1.2 are the bases used to resolve, by calculation, or maintenance/modifications, all identified outliers. These resolutions are summarized in Tables 4.1 and 4.2 of the EQE Report.

NRC ITEM 6

Referring to Page 10 of the EQE Report, and noting that different Class I buildings at Browns Ferry Nuclear Plant have different vertical soil amplification factors, please explain the basis for the specific scaling factors selected for the Turbine Building.

TVA RESPONSE TO ITEM 6

The methodology to determine the soil amplification factors for the various Class I structures at BFN is defined in TVA Design Criteria BFN-50-C-7102, Seismic Design, which requires that structures founded on soil consider soil amplification. The soil amplification factors for applicable Class I structures are shown in BFN FSAR Chapter 12, Structures and Shielding. The horizontal soil amplification factors range from 1.0 for rock-founded structures such as the Reactor Building to a maximum of 1.6 for soil-founded structures such as the Diesel Generator Building (DGB). Similarly, the vertical soil amplification factors range from 1.0 to 1.3. Seismic demand for equipment in a particular structure is determined by scaling the site design basis response spectrum, i.e., the Housner spectrum for 5% damping and anchored at 0.2g, by the appropriate horizontal and vertical soil amplification factors.

Since the Turbine Building is designated as a Class II structure in the FSAR, no soil amplification provisions were originally specified and no dynamic seismic analysis results were available to define seismic demand on the structure or components. It was determined that the soil amplification factors for the DGBs would be most representative for the Turbine Building. The foundation materials are similar as are the foundation depths. In addition, the DGB horizontal soil amplification factor of 1.6 is known to be conservative, so this conservatism will be extended to the specification of the seismic demand for the equipment in the Turbine Building for the seismic evaluation. The primary foundation difference is that the Turbine Building is supported on steel H-piles to bedrock. However, it is considered that the primary effect of the pile foundation would be to increase the foundation stiffness in the vertical direction relative to a similar foundation without piles. Therefore, the horizontal soil amplification for the Turbine Building would have a more significant effect than that of the vertical in the overall seismic verification efforts.

Accordingly, seismic demand for equipment in the Turbine Building and for the seismic assessment of components is based on the same horizontal soil amplification factor of 1.6 and vertical soil amplification factor of 1.1 as was used for the DGBs. These factors were used to scale the BFN design basis DBE response spectrum (0.2g Housner spectrum, 5% damping) to determine seismic demand.

NRC ITEM 7

In Table 4-8 and Figures 4-2 thru 4-5 of the EQE Seismic Evaluation Report, only Moss Landing Units 6 & 7 condensers are provided for comparison with the Browns Ferry condensers. This is too limited to support a finding that the earthquake experience database demonstrates the seismic adequacy of Browns Ferry's condensers. Please provide additional condenser data.

As stated in the staff's March 3, 1999 safety evaluation, there is no standard at the present time, endorsed by NRC, that provides guidance for determining the required number of piping and equipment items, that should be referenced in the earthquake experience database when utilizing the BWROG methodology. Therefore, you are responsible for ensuring the sufficiency of the above data submitted for staff review and determination. If sufficient data are not provided for the condenser, the NRC may require that the condenser be analytically evaluated against all the pertinent operating and design loadings, in accordance with the plant's design basis methodology and criteria.

TVA RESPONSE TO ITEM 7

The BFN condenser design attributes are shown to fall within the bounds of the Moss Landing database site as discussed in Section 4.4 and depicted on Table 4-8 of the EQE Report. To provide additional assurance that BFN condensers would maintain structural integrity, a specific analysis was performed on the condenser subject to BFN seismic demand. Results of the analyses demonstrate that the condenser shell stresses are small, with maximum stress ratios based on American Institute of Steel Construction (AISC) allowables of 0.12 for combined axial and bending and 0.10 for shear (Reference section 4.4 of EQE Report). Additionally, the condenser anchorage was also compared with the performance of condensers of the database site. The anchorage was demonstrated by seismic experience and by analytical methods to be acceptable. Maximum stress ratios from the condenser support anchorage evaluation including BFN seismic demand, based on AISC allowables, are 0.70 for bolt tension in the perimeter support feet and 0.86 for shear in the center support built-up section (Reference: Section 4.4 of EQE Report). Based on the above, it was concluded that the BFN condensers were acceptable.

Refer to Enclosure 2 for a copy of the calculations used to determine the stress ratios given above.

NRC ITEM 8

The radiological analysis description provided in the application does not provide an adequate basis for the staff to determine whether or not those analyses are acceptable. The staff notes that the reported increase in doses appears to be inconsistent with the proposed eight fold increase in the allowable MSIV leakage. Please provide the analysis assumptions, methods, and input parameters used in your calculations, in sufficient detail for the staff to resolve the apparent inconsistency and, if deemed necessary by the staff, to perform independent calculations to confirm your reported results. Your response should identify any changes made to the assumptions, methods, and inputs used in analyses previously approved by the NRC for Browns Ferry Units 2 and 3.

TVA RESPONSE TO ITEM 8

Refer to Enclosure 3 for a response to this item and to additional NRC staff questions on dose methodology.

After further review, we agree that the linearity assumption used in TVA's initial calculation is not always conservative. Therefore, TVA reperformed the MSIV leakage dose calculations rather than use extrapolation factors to determine the MSIV leakage contribution to dose. These were completed in accordance with the NEDC methodology as reviewed in the NRC SER.

This recalculation resulted in a reduction of the requested MSIV allowable leakage rate. Therefore, TVA is providing an amended TS change request as part of this response (See Enclosures 4 and 5).

NRC ITEM 9

Your application requests an exemption from the requirement that MSIV leakage be included the overall Type A leakage limit (in addition to the 0.6 L_a limit for the sum of Types B and C penetration leakage). Is it your understanding that this is consistent with NEDC-31858? Is there a valid need for this exemption?

TVA RESPONSE TO ITEM 9

10 CFR 50, Appendix J testing ensures primary containment leakage following a design basis LOCA will be within the allowable leakage limits specified in plant TS and assumed in the safety analyses for determining radiological consequences. For BFN, the acceptance criteria for the Type A test Containment Integrated Leakage Rate Test (CILRT) is 0.75 L_a for return to power following performance of the CILRT. This limit is shown in BFN TS 5.5.12, Primary Containment Leakage Rate Testing Program.

The 0.75 L_a acceptance criteria allows for a 25% margin for degradation during plant operation. The CILRT currently includes leakage through the closed MSIVs. The proposed increase in MSIV leakage, if not excluded from the 0.75 L_a acceptance criteria for the CILRT, could account for approximately 18% of the 0.75 L_a acceptance criteria and significantly reduces the margin available for all other primary containment leakage paths. Inclusion of MSIV leakage in the CILRT would effectively reduce the CILRT acceptance criteria to approximately 0.62 L_a.

In analyzing the use of the ALT path, the radiological consequences of MSIV leakage are being determined separately from other primary containment leakage, since MSIV leakage is released directly into the Turbine Building, which is not treated by the Standby Gas Treatment System. The MSIV leakage rates are measured as part of the 10 CFR 50, Appendix J Program, to verify this leakage will not exceed the proposed maximum leakage in the TS and assumed in the safety analyses for radiological consequences. Therefore, since the effects of MSIV leakage are being explicitly accounted for in the dose analysis, it is appropriate that MSIV leakage be excluded from the Type A testing results.

Exclusion of MSIV leakage from the Type A test acceptance criteria is necessary to provide adequate margin for leakage of the remaining primary containment leakage paths tested during the CILRT. This exclusion is justified because of the separate treatment of MSIV leakage as previously discussed. The radiological consequences of primary containment leakage and MSIV leakage will continue to be maintained within allowable limits and the intent of 10 CFR 50, Appendix J will continue to be satisfied.

NEDC-31858P, Section 6.3.2.1, discusses the need for Appendix J exemptions for both Type A and Type C tests. Therefore, the exemption request is consistent with the NEDC.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 and 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-399
INCREASED MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
DATED NOVEMBER 23, 1999

CONDENSER SEISMIC STRESS CALCULATION

- Excerpt from calculation CD-N0001-980038, R1 Main Steam line Ruggedness - Shows Condenser Anchorage Calculation
- Excerpt from calculation CD-N0001-990113, R0 Seismic Evaluation Report - Shows Condenser Shell Calculation

QA Record

TVAN CALCULATION COVERSHEET

Title Main Steam Seismic Ruggedness Evaluation		Page i of	Plant BFN Unit 3
Preparing Organization CE-Civil	Key Nouns (For RIMS) Seismic, Component Qual, Piping, Pipe Support		
Branch/Project Identifiers CD-N0001-980038	Each time these calculations are issued, preparer must ensure that the original (R0) RIMS accession number is filled in.		

Applicable Design Document(s)	Rev	(for RIMS use)	RIMS Accession Number
BFN-50-C-7100, BFN-50-C-7102 BFN-50-C-7306	R0	344	R14 980915 106
	R 1		R14 990930 106
SAR affected: ☐Yes ☒No Section(s): — UNID System(s): 001, 006, 008, 012, 043, 071, 073, 303	R		
	R		

	Rev 0	R 1	R	R			
Design Change Document No.	T40871A	T41019A			Quality Related?	Yes ☒	No ☐
Prepared	W. Lugardes 9-11-98	W. Lugardes 9-28-99			Safety related?	Yes ☐	No ☒
Checked	9-12-98	F. CARAMANTE 09/29/1999			These calculations contain unverified assumption(s) that must be verified later?	Yes ☐	No ☒
Reviewed	9-12-98	09/29/1999			These calculations contain special requirements and/or limiting conditions?	Yes ☐	No ☒
Approved	J R Glass	L.R. Wadium, Jr.			These calculations contain a design output attachment?	Yes ☐	No ☒
Date	9-15-98	9-30-99			Calculation Revision: Entire Calculation ☒ Selected pages ☐ Not Applicable ☐		

Statement of Problem:
The Main Steam piping downstream of the outboard MSIVs is needed to be capable of with standing an earthquake so that any leakage through the MSIVs from the Reactor side can be contained and diverted to the main condensers. A walkdown was performed by EQE to verify the seismic adequacy of the piping. Problems found during the walkdown were identified as outliers. The outliers which were found acceptable as-is by performing detailed engineering evaluations are documented in this calculation. Also included in this calculation is the EQE report documenting the outliers and the final resolution for each item. The outliers that required a plant modification as a solution are documented in calculation CD-N0001-980039.

Abstract
This calculation is a collector for the reports, calculations, etc. done by EQE International for the seismic ruggedness verification of the BFN Main Steam piping in the Turbine Building. The scope of the seismic ruggedness walkdown performed by EQE was based on preliminary isolation boundary locations as shown in Table 2-2 and Figure 2-1 of the 'Summary Report" contained in this calculation as Attachment A. A review of this preliminary isolation boundary has shown that the valves at these locations may not close as desired. Once the isolation boundary review is complete, any additional walkdowns determined necessary to ensure the seismic ruggedness of the Main Steam piping will be done in support of DCN T41019A.

ORIGINAL

Calculation Classification - D |R1

Note: This calculation is to provide a retrievable source for the summary report and supporting calculations performed by EQE International, Inc. TVA signatures are not attesting to the technical accuracy of the included documents of this calculation.

Total Pages: ~~344 R0~~ 345 |R1

☐ Microfilm and return calculation to Calculation Library. Address:
☐ Microfilm and return calculation to:
☐ Microfilm and destroy.