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

July 9, 2004

TVA-BFN-TS-436

10 CFR 50.12
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop: OWFN P1-35
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-259
Tennessee Valley Authority)

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1 - TECHNICAL SPECIFICATION (TS) 436 - INCREASED MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE RATE LIMITS AND EXEMPTION FROM 10 CFR 50, APPENDIX J

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting a request for a TS change to license DPR-33 for BFN Unit 1. The proposed TS change increases the allowable leakage rate criteria for the MSIVs. In addition, in accordance with 10 CFR 50.12, TVA is requesting 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. TVA has previously requested, and NRC approved, similar changes for Browns Ferry Units 2 and 3 (References 1 through 4).

This request is based on the Boiling Water Reactor Owners' Group (BWROG) methodology described in NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems". NRC has previously determined that this report is acceptable for direct reference for individual licensee applications as documented in the associated NRC Safety Evaluation Report, dated March 3, 1999.

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This proposed change reduces the frequency of MSIV rebuilds during outages that are required to achieve the leakage rates specified in the current TS (11.5 standard cubic feet per hour). This will extend the service life of the MSIVs as well as reduce the radiation exposure to personnel involved in MSIV maintenance activities.

Specifically, TVA proposes to utilize the main steam drain lines to preferentially direct any MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. The components and piping systems used in this treatment path have been evaluated to be capable of performing their function following a Design Basis Earthquake (DBE). A seismic ruggedness verification program has been conducted in accordance with NEDC-31858P-A to ensure that the main steam piping from the outboard MSIV's to the turbine stop valves and the main steam drain line from the outboard MSIV's to and including the condenser itself are capable of being a pressure retaining boundary following a DBE.

A plant-specific radiological analysis has also been performed to assess the effects of the proposed increase in MSIV leakage acceptance criteria in terms of off-site and main control room operator doses. This analysis uses the holdup and plate-out factors described in NEDC-31858P-A. The analysis shows the dose contribution from the proposed increase in leakage acceptance criteria is acceptable compared to doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

This proposed TS change is necessary to support the restart of Unit 1. Therefore, TVA requests that the change be approved by July 1, 2005 and that the implementation of the revised TS be within 60 days of NRC approval.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Alabama State Department of Public Health.

Enclosure 1 provides TVA's evaluation of the proposed TS change. Enclosures 2 and 3 provide mark-ups of the proposed change to the TS and TS Bases, respectively. Enclosures 4 and 5 provide copies of the TS and TS Bases pages, respectively, with the proposed changes incorporated. The TS Bases changes in Enclosures 3 and 5 are provided for information and do not require NRC approval.

A request for an exemption to 10 CFR 50, Appendix J containment leakage requirements is provided as Enclosure 6 in accordance with 10 CFR 50.12. This exemption request supports the TS change to increase the MSIV leakage acceptance criteria.

Enclosure 7 provides a report summarizing the seismic ruggedness verification program for increasing the Unit 1 MSIV leakage rate limits. Enclosure 8 provides the supporting seismic ruggedness evaluation. The material contained in these enclosures was previously provided to NRC in support of TS 405, Alternative Source Term (Reference 5). A summary of the dose calculations and results were previously submitted in support of TS 405, Alternative Source Term, in Reference 6.

Enclosure 9 provides a listing of commitments made in this submittal. If you have any questions about this change, please contact me at (256) 729-2636.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 9, 2004.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures:
1. TVA Evaluation of Proposed Change
2. Proposed changes to the Technical Specifications (mark-ups)
3. Proposed changes to the Technical Specification Bases (mark-ups)
4. Proposed changes to the Technical Specifications (clean pages)
5. Proposed changes to the Technical Specification Bases (clean pages)
6. Request for an exemption to 10 CFR 50, Appendix J containment leakage requirements
7. Summary of the seismic ruggedness verification program
8. Seismic ruggedness evaluation
9. Summary of commitments

References:

1. TVA letter, T.E. Abney to NRC, dated September 28, 1999, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits and Exemption from 10 CFR 50 Appendix J."

2. NRC letter, W.O. Long to J.A. Scalice, dated November 23, 1999, "Browns Ferry Units 2 and 3, Main Steam Isolation Valve Leak Rate Limits, Request for Additional Information (TAC Nos. MA6405, MA6406, MA6815 and MA6816)."

3. TVA letter, T.E. Abney to NRC, dated February 4, 2000, "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)."

4. NRC letter, W.O. Long to J.A. Scalice, dated March 14, 2000, "Issuance of Amendments Regarding Limits on Main Steam Isolation Valve Leakage (TAC Nos. MA6405 and MA6406)."

5. TVA letter, T.E. Abney to NRC, dated July 2, 2004, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Response to Request for Additional Information (RAI) Related to Technical Specifications (TS) Change No. TS-405 - Alternative Source Term (AST) (TAC Nos. MB5733, MB5734, MB5735).

6. TVA letter, T.E. Abney to NRC, dated May 17, 2004, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Response to Request for Additional Information (RAI) and Unit 1 Analysis Results Related to Technical Specifications (TS) Change No. TS-405 - Alternative Source Term (AST) (TAC Nos. MB5733, MB5734, MB5735)."

Enclosures
cc: (Enclosures):

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ENCLOSURE 1

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION (TS) CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

INDEX

1.0 DESCRIPTION

This enclosure provides a request to amend Operating License DPR-33 for BFN Unit 1. The proposed TS change increases the allowed MSIV leak rate from 11.5 standard cubic feet per hour (scfh) per valve to 100 scfh for individual MSIVs with a maximum 150 scfh combined leakage for all four main steam lines.

TVA has previously requested, and NRC approved, similar changes for Browns Ferry Units 2 and 3 (References 1 through 4). The content of this submittal is based on both the Boiling Water Reactor Owners' Group (BWROG) methodology described in NEDC-31858P-A, *BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems* (Reference 5), and TVA's previous applications for Units 2 and 3. This proposed change is necessary to support the restart of Unit 1. Therefore, TVA requests that the TS change be approved by July 1, 2005 and that the implementation of the revised TS be within 60 days of NRC approval.

2.0 PROPOSED CHANGE

2.1 Proposed Technical Specification Change

The proposed change to Surveillance Requirement 3.6.1.3.10 increases the allowed MSIV leak rate from 11.5 scfh per valve to 100 scfh for individual MSIVs with a 150 scfh combined leakage for all four main steam lines is shown below.

Current Surveillance Requirement 3.6.1.3.10:

> "Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig."

Proposed Surveillance Requirement 3.6.1.3.10:

> Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.

2.2 Associated Technical Specification Bases Changes

The associated changes to the TS Bases are shown below. The associated TS Bases changes are provided for information and do not require NRC approval.

Current Bases description of Surveillance Requirement 3.6.1.3.10:

> The analyses in References 1 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ Pt (25 psig). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

Revised Bases description of Surveillance Requirement 3.6.1.3.10:

> The analyses in References 1 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 100 scfh when tested at ≥ Pt (25 psig). The combined leakage rate for all four main steam lines must be ≤ 150 scfh when tested at ≥ 25 psig in accordance with the Primary Containment Leakage Rate Testing Program. If the leakage rate through an individual MSIV exceeds 100 scfh, the leakage rate shall be restored below the alarm limit value as specified in the Containment Leakage Rate Testing Program referenced in TS 5.5.12. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

3.0 BACKGROUND

3.1 Reason for the Proposed Change

Refurbishment of an MSIV to meet the 11.5 scfh criteria is a man-hour intensive effort which results in a cumulative worker radiation dose of approximately 4.5 man-REM during a complete rebuild. With a 100 scfh limit for individual MSIVs and 150 scfh combined limit, the amount of rework on the MSIVs would be significantly reduced. The change would lower personnel radiation exposure and improve the overall performance integrity of the MSIVs by reducing the number of maintenance activities associated with restoring the leakage to an overly strict lower limit. Approval of this proposed change would also be an economic benefit to TVA in terms of direct costs and a reduction in outage activities.

3.2 Regulatory Background

This change request is based on the utilization of the methodology described in NEDC-31858P-A, BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems (Reference 5) and TVA's previous applications for Units 2 and 3 (TS 399). NRC has previously determined that the subject BWROG report is acceptable for direct reference for individual licensee applications as documented in the associated NRC Safety Evaluation Report (SER), dated March 3, 1999 (Reference 6). The licensee specific items identified in the staff's SER are addressed in Section 4.2. The BFN plant design does not include a MSIV leakage control system as do later vintage BWRs, therefore, there are no changes required in this area.

3.3 Comparison with Previous BFN Units 2 and 3 Change

TVA has compared the proposed change, reason for change, background information, and technical analysis submitted in support of this proposed TS change with the information provided by TVA and approved by NRC in References 1 through 4 for the previous Units 2 and 3 TS changes which increased the MSIV leakage rate limit. The comparison for each of these areas is provided below:

- The proposed changes to the Unit 1 TSs are the same changes that were proposed and approved for Units 2 and 3.

- The reasons for the Unit 1 TS change is the same as that which was previously submitted for the Units 2 and 3 TS (e.g., reduce the frequency of MSIV rebuilds during outages in order to extend the service life of the MSIVs as well as reduce the radiation exposure to personnel involved in MSIV maintenance activities). In addition, TVA needs to maximize consistency between the Unit 1 and Units 2 and 3 TS, operations and maintenance procedures and practices prior to restarting Unit 1.

- The background information provided in support of the Unit 1 TS change incorporates the same elements previously submitted in support of the Units 2 and 3 TS change.

- The technical analysis submitted for this Unit 1 TS change incorporates the same elements previously submitted in support of Units 2 and 3 (References 1 and 3). However, the following specific differences were incorporated into the Unit 1 submittal:

- In March 1999, NRC issued a Safety Evaluation accepting the BWROG topical report for increasing MSIV leakage limits. This report was approved for direct reference in individual plant submittals, subject to the plant unique conditions and limitations described in Section 6 of the Safety Evaluation. Section 4.2 of this submittal contains the licensee specific items required to be addressed and a summary / cross-reference to the required information. This section was added to assist NRC in their review of this application.

- The Rated Thermal Power assumed in the Unit 1 analyses that support this submittal is 3952 MWt versus a value of 3293 MWt for the analyses that supported the Units 2 and 3 TS change.

- The dose calculation results performed in support of this submittal utilize an Alternative Source Term, instead of the TID 14844 based source term utilized in the Units 2 and 3 submittal. The Unit 1 dose calculations assumed the proposed maximum allowable main steam line leakage of 150 scfh instead of the conservative 168 scfh value assumed in the Units 2 and 3 calculations that supported TS 399.

- The unit specific inputs for reactor building effective mixing free volume and turbine building release X/Q values differ between Unit 1 and those utilized in the Units 2 and 3 calculations.

In addition, a summary of the Unit 1 dose calculations were previously submitted in support of TS 405, Alternative Source Term, in Reference 7. The report summarizing the seismic ruggedness verification program and the supporting seismic ruggedness evaluation were previously provided to NRC in support of TS 405, Alternative Source Term (Reference 8).

4.0 TECHNICAL ANALYSIS

4.1 Summary

There are four steam lines installed on each BFN unit and each line is provided with dual quick-closing MSIVs. These valves serve to isolate the reactor coolant system in the event of steam line breaks outside primary containment, a design basis Loss of Coolant Accident (LOCA), or other events requiring containment isolation. For a detailed discussion of the system components and operating characteristics, refer to Section 4.6.3 of the BFN Updated Final Safety Analysis Report (UFSAR).

For a steam line break, the MSIV isolation function prevents damage to the fuel barrier by limiting the loss of reactor coolant. For the LOCA event, the valves limit the release of radioactive materials by closing the primary containment barrier to contain a leak from the nuclear system.

Although the MSIVs provide a low leakage barrier, industry and BFN experience indicates that it is difficult to maintain the valves to meet the current TS Surveillance Requirement 3.6.1.3.10 performance criteria of 11.5 scfh per valve.

TVA proposes the utilization of the main steam drain lines to preferentially direct any MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this approach, the main steam lines, steam drain piping, and the main condenser are used to mitigate the consequences of an accident:

- To limit potential off-site exposures below those specified in 10 CFR 50.67(b)(2)(i) for the exclusion area and, 10 CFR 50.67(b)(2)(ii) for the low population zone, and

- To limit control room operator doses below those specified in 10 CFR 50.67(b)(2)(iii) for control room personnel.

Because the original BFN design basis of the Turbine Building main steam piping and components was not Seismic Class I, the components and piping systems used in this treatment path have been evaluated to be capable of performing their function following a Design Basis Earthquake (DBE) in accordance the technical methodology in the BWROG report, NEDC-31858P-A. Seismic verification walkdowns and evaluations of piping/supports have been performed to demonstrate the main steam line piping and components that comprise the Alternate Leakage Treatment (ALT) path to the condenser are rugged and able to perform the safety function of MSIV leakage control and treatment following a DBE. In addition, TVA retained the services of Facility Risk Consultants (FRC), Inc. (which includes personnel formerly with EQE International, who performed this work on Units 2 and 3) to perform the walkdowns and evaluations of the MSIV seismic ruggedness. As previously documented in References 1 and 3 for BFN Units 2 and 3, based on past earthquake experience, welded steel piping and condensers constructed to normal industrial practices have been found to be seismically rugged and not susceptible to a primary collapse mode of failure as a result of the seismic vibration modes experienced at sites examined in the earthquake database.

Using the earthquake experience-based methodology, supplemented by the walkdowns and evaluations, it has been concluded the components in the ALT path can be relied upon to maintain structural integrity during a DBE.

Off-site dose calculations and control room dose calculations have been performed using the ALT path and an increased MSIV leakage term in accordance with NEDC-31858P-A. The revised calculations show that the doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel are maintained with the proposed MSIV leakage acceptance criteria.

4.2 Response to Licensee Specific Items in NRC Safety Evaluation of the Topical Report

In March 1999, NRC issued a Safety Evaluation accepting the BWROG topical report for increasing MSIV leakage limits (Reference 7). This report was approved for direct reference in individual plant submittals, subject to the plant unique conditions and limitations described in Section 6 of the Safety Evaluation. Provided below are the licensee specific limitations required to be addressed in this application and a summary / cross-reference to the required information.

Limitation 1:

> Individual licensees should provide a detailed description of the ALT drain path and the basis for its functional reliability, commensurate with its intended safety-related function. The licensee should also describe their maintenance and testing program for the active components (such as valves) in the ALT path.

Response:

> A detailed description of alternate flow path to condenser is provided in Section 4.3. In general, the piping and components within the boundaries of the MSIV ALT drain path are considered to be within the scope of the BFN Section XI In Service Testing (IST) and In Service Inspection (ISI) programs. Accordingly, they will be inspected and tested in accordance with the IST/ISI programs. A detailed discussion is provided in Section 4.4.

Limitation 2:

Individual licensees should provide plant-specific information for piping design parameters (e.g., uniqueness of piping configurations, pipe span between supports, and diameter-to-thickness ratios for each pipe size), to demonstrate that they are enveloped by those associated with the earthquake experience database.

Response:

As previously mentioned, TVA retained the services of FRC to perform the walkdowns and evaluations to document the ALT pathway seismic ruggedness as was done for BFN Units 2 and 3. The in-plant screening walkdown evaluations focused on key design attributes and seismic issues such as pipe spans and support integrity; seismic interaction issues (including proximity impact and falling concerns; differential displacement of structures, equipment and piping, performance of seismic verification boundary components; etc). Details are provided in Section 4.6.

Limitation 3:

Individual licensees should demonstrate that the plant condenser design falls within the bounds of design characteristics found in the earthquake experience database. This should include a review of as-built design documents and/or a walkdown to verify that the condenser has adequate anchorage.

Response:

In general, the condensers in the earthquake experience database exhibited substantial seismic ruggedness, even when they typically were not designed to resist earthquakes. This is a common conclusion in studies of this type on other plant items such as welded steel piping in general, anchored equipment such as motor control centers, pumps, valves, structures, and so forth. The actions taken to verify the seismic performance of the condenser included a Unit 1 specific walkdown and analysis of the seismic ruggedness of the condenser and its anchorage. Details are described in Section 4.7.

Limitation 4:

Individual licensees should perform a plant-specific seismic evaluation for representative supports and anchorages associated with affected piping and the condenser.

Response:

As discussed in Section 4.1, plant-specific seismic evaluations were performed for supports and anchorages associated with affected piping. The specific actions taken to verify the seismic performance of the condenser are described in Section 4.7. These actions included a Unit 1 specific analysis of the seismic ruggedness for the condenser and its anchorage.

Limitation 5:

Individual licensees should confirm that the condenser will not fail due to seismic II/I type of interaction (e.g., structural failure of the turbine building and its internals).

Response:

Structures, piping or equipment adjacent or above the components under review were evaluated for seismically induced II/I or impact failures. Additional information is provided in Section 4.6.1.

Limitation 6:

Individual licensees of plants whose FSARs or UFSARs reference Appendix A to 10 CFR Part 100 should perform a bounding seismic analysis for the ALT path piping. Those licensees committed to Part 100 should discuss the basis for selecting a particular portion of the bypass/drain line for the bounding analysis.

Response:

The BFN UFSAR does not reference Appendix A to 10 CFR 100. Unit 1 specific seismic verification walkdowns and the previous evaluations of bounding Units 2 and 3 piping/supports were performed to demonstrate the main steam line piping and components that comprise the ALT path to the condenser are rugged and able to perform the safety function of MSIV leakage control and treatment following a DBE.

Limitation 7:

The methodology and criteria used for the analytical evaluations should be those which are in compliance with the design basis methodology and criteria, or those which are acceptable to the staff.

Response:

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 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. 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 DBE and were evaluated in accordance with the Generic Implementation Procedure (GIP) methodology[1]. Qualification in accordance with GIP provides reasonable assurance the required valves will be functional.

Details regarding the efforts and criteria used to ensure structural integrity are provided in Section 4.6.

Limitation 8:

The facility ground motion estimates shown in Figures 1 through 13 of this attachment have been reviewed and accepted by the staff for inclusion in BWROG's earthquake experience database. These 13 facility ground motion estimates may be used to verify the seismic adequacy of equipment in the alternative MSIV leakage pathway for plants referencing the BWROG's Topical Report, NEDC-31858P, Revision 2.

1 J. Partlow (NRC) to All Unresolved Safety Issue (USI) A-46 Plant Licensees Who Are Members of the Seismic Qualification Utility Group (SQUG), "Supplement No. 1 to Generic Letter (GL) 87-02 that Transmits Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure, Revision 2, as Corrected on February 14, 1992 (GIP-2)," May 22, 1992.

Response:

Comparisons of the ground response spectra of selected database facilities with BFN design basis ground spectrum were made to establish the applicability of NEDC-31858P-A. It was concluded that the BFN DBE ground spectrum is generally bounded by the earthquake experience database sites at the frequencies of interest. Details are included in Section 4.6.2.

Limitation 9:

At the present time, there is no standard, endorsed by NRC, that provides guidance for determining what constitutes an acceptable number of earthquake recordings and their magnitudes and 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, individual licensees are responsible for ensuring the sufficiency of the data to be submitted for staff review and determination. When a revision of the QME Standard that incorporates specific criteria for use of experience data in the qualification of mechanical equipment is endorsed by NRC, such criteria should be followed in future applications involving MSIV ALT pathway evaluations.

Response:

The earthquake databases have been verified applicable to BFN as discussed in Section 4.6.2.

4.3 Detailed Description of Alternate Flow Path to Condenser

Figure 1 (Page E1-33), MSIV Seismic Verification Boundary, provides a flow diagram schematic which shows the ALT pathway from the MSIVs to the condenser and the boundary valves associated with the ALT pathway for BFN Unit 1. As discussed in NEDC-31858P-A, the ALT pathway establishes a seismically equivalent route to contain and direct leakage from the MSIVs to the condenser following a design basis LOCA.

The ALT path is from the outboard side of the MSIVs through four 3-inch lines which join a 4-inch drain line path to the condenser. The flow path is through normally open Flow Control Valves (FCVs) 1-168, 169, 170, and 171, and continues through FCV 1-57, FCV 1-58, and FCV 1-59 to the main condenser. FCV 1-57 is normally open and FCVs 1-58 and 1-59 are normally closed valves. FCV 1-59 has a 4" bypass line which also routes to the condenser. The bypass around FCV 1-59 is free of valves and orifices.

Establishment of the ALT path is based on valves FCV 1-58 and FCV 1-59 (although FCV 1-59 is not strictly required because of the bypass line) being opened by operators in response to those events taking credit for the availability of the ALT path to the condenser. Operating procedures will provide procedural requirements to establish the ALT path to the condenser.

FCVs 1-168, 169, 170, 171, and 57 are normally open motor operated valves which would remain open on loss of off-site power. FCV 1-58 and FCV 1-59 are normally closed valves which would require operator action to align the ALT path to the condenser. These two valves are powered from essential power buses with emergency diesel generator back-up. To further ensure valve reliability, FCV 1-58 and FCV 1-59 are in the IST program and will be periodically stroke tested. However, as stated above, FCV 1-59 has an open full bypass line, therefore, its operation is not essential to align the ALT path.

The ALT pathway includes the main steam and main steam drain piping, and branch lines, which act as a boundary volume to contain the MSIV leakage and direct it via the ALT pathway to the condenser. All of the boundary valves fall into one of the following categories:

1. Manual isolation valves that are normally closed;

2. Motor operated valves that are normally closed;

3. Air operated valves that are normally open, but fail closed on loss of power, loss of air, or loss of control signal;

4. Air operated valves that are normally open which will require an operator action to close; and

5. Valves isolated by a spring assisted in-line check valve, which has a opening pressure in excess of the ALT drain path differential pressure once the MSIVs have closed and the line has depressurized.

The following changes to boundary valves will be made prior to Unit 1 restart:

1. The steam line supplies to the Offgas Preheaters currently do not have adequate boundary valves. Therefore, in-line check valves (CKV) 1-CKV-1-742 and 1-CKV-1-744 are being added to these lines to serve as the boundary valves. Refer to Figure 1 for the location of these valves.

2. Pressure Control Valve (PCV) 1-147 is currently an air operated valve which fails open. Therefore, PCV-1-147 is being modified by changing the operator to an operator that will fail closed on loss of power, air, or control signal.

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) could present an operational problem only at low reactor powers (below approximately 25 percent power). This could 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 will direct 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 an operational problem depending 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.

A secondary passive flow path also exists from the MSIVs to the condenser. This flow path consists of:

1. Four 2-inch bypass lines, which contain 0.25″ orifices around FCV 1-168, 169, 170 and 171;

2. A 1″ bypass around the FCV 1-58 valve which has a 0.1875″ orifice and a normally open manual valve, Hand Control Valve (HCV) 1-525 and;

3. A 4″ bypass line around FCV 1-59.

The ALT path is from the outboard side of the MSIVs through FCV-1-58 to the condenser. This path satisfies the sizing requirements of NEDC-31858 paragraph 6.1.1(2) that 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 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.

4.4 <u>Reliability of the Alternate Flow Path</u>

As previously noted, Figure 1 provides a flow diagram that shows the ALT leakage pathway from the MSIVs to the condenser, and the boundary valves. The boundary valves and material properties of the drain line piping and branch lines have been documented. All boundary valves are either closed during normal system operation or fail closed upon loss of power, or loss of control air or hydraulic pressure.

To establish the primary ALT path to the condenser flow path to the condenser, FCV 1-58 and FCV 1-59 will be opened using hand switches in the main control room. FCV 1-58 will also auto-open for certain combinations of the MSIVs being closed and turbine speed conditions. As noted above, the opening of FCV 1-59 is not essential since it has a 4-inch non-orificed bypass line. Both FCV 1-58 and FCV 1-59 will be powered from essential power buses with emergency diesel backed power. Therefore, they are designed to be available during and after a LOCA event concurrent with a loss of off-site power.

The most limiting single active failure would be the failure of FCV 1-58 valve to open. In this condition, MSIV leakage flow would be diverted through the 1" orificed bypass line around FCV 1-58 and through normally open manual valve HCV 1-525. With the 0.1875" orifice, 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. Therefore, even in the unlikely event of this single active failure, the bulk of the MSIV leakage will still be routed to the condenser.

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

If the normal feeder (480-V Shutdown Board 1B) to RMOV Board 1C is lost, it can be transferred to its alternate power supply (480-V Shutdown Board 1A) by remote breaker operation. Therefore, it is an easy operation to transfer 480-V RMOV Board 1C to its alternate emergency power supply. As noted above, the two 480-V Division I and II Shutdown Boards both have their own (separate) Emergency Diesel Generator power supplies. Refer to UFSAR Figures 8.4-1.b and 8.4.2 for a diagram of this electrical distribution arrangement. For the reasons stated above, the current design ensures that power will be available to FCV-1-58 in the event of loss of offsite power.

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, TVA determined 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 will be in the preventative maintenance program. As such, one Main Steam Stop and one Control valve will be refurbished each outage. These valves will be tested each refueling outage for leak tightness and are 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.

In general, the piping and components within the boundaries of the MSIV ALT drain 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. The IST program will test the power operated valves within the ALT drain 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). Certain valves that serve as part of the ALT drain 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. Some of these excluded valves will be tested during power operations to ensure their functionality and will be in the preventive maintenance program for periodic refurbishment. These valves will be included as part of the IST program, but as non-Code valves. 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 Enclosure 8.

FCV-1-58 was considered for inclusion in the augmented motor operated valve (MOV) test programs such as Generic Letter (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 drain path is a LOCA with assumed major core damage and the MSIVs closed. With the MSIVs closed, the ALT drain path boundary is physically isolated from the reactor vessel and primary containment except through leakage through the MSIVs. The ALT drain path piping will depressurize through the orifice around FCV-1-58. In order to establish the ALT drain 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 programs since the periodic IST program testing on this valve is considered to be adequate to ensure its functionality.

As part of the ALT path isolation boundary, two in-line check valves (1-CKV-1-742 & 1-CKV-1-744) are being added to the steam supply lines for the Offgas Preheaters as shown in Figure 1. These check valves will be spring assisted to remain closed against less than a 5 pound per square inch (psi) differential pressure. This is based on the pressure downstream of the MSIVs being 1 or 2 psi once the steam lines have depressurized and the condenser has lost vacuum. The check valves will be ASME Section III, Class 2 valves. Spring assisted in-line check valves have been used in the nuclear and non-nuclear industry to resolve issues with other types of check valves. This type of in-line check valve has been very reliable, and provides leak tight closure. Therefore, a spring assisted check valve is well suited for this application.

These new valves will also be within the scope of the 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 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. MOVs 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.

The majority of the ALT drain path and boundary paths fall within the ISI ASME Section III Class 2 Program. Inclusion of this piping in the ISI program further ensures the reliability of this piping through periodic inspections.

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.

4.5 Radiological Dose Assessment

The dose calculation results performed in support of this submittal are same alternate source term calculation results submitted as part of TS 405, Alternative Source Term. The results of the analyses for the Unit 1 offsite dose and the Units 2/3 control room dose are bounding for all three units. The Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and control room calculated doses are within the regulatory limits. The following table presents the results of the bounding LOCA radiological consequence analysis.

TABLE BFN UNITS 1, 2 AND 3 LOCA RADIOLOGICAL CONSEQUENCE ANALYSIS (REM TEDE)			
	Offsite Dose[1]		Control Room Dose[2]
Dose Component	EAB	LPZ	
Base of Stack	-	1.14E-2	4.49E-3
Top of Stack	-	6.14E-1	2.43E-1
Turbine Building Roof	-	3.02E-1	1.13E-1
ECCS Leakage - Base of Stack	-	1.27E-2	1.21E-2
ECCS Leakage - Top of Stack	-	3.64E-1	1.12E-1
Shine	-	N/A	7.62E-1
TOTAL	**1.11**	**1.30**	**1.25**
Regulatory Limit	25	25	5

[1] **Bounding results (Unit 1) are shown.**

[2] **Bounding results (Units 2/3) are shown.**

As shown in the above table, the calculations demonstrate that the dose contribution from the increased MSIV leak rate is far below the doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

4.6 Structural Integrity

Portions of the main steam piping and components in the Turbine Building at BFN were not originally designed as Seismic Class I. Therefore, seismic verification walkdowns and evaluations of piping/supports were performed to demonstrate the main steam line piping and components that comprise the ALT path were rugged, and would be able to perform the safety function of MSIV leakage control following a DBE.

As previously documented in References 1 and 3 for BFN Units 2 and 3, a review of the earthquake experience data on the performance of nuclear power plants and condensers was conducted. Using the experience-based methodology, supplemented by the walkdowns and evaluations, it has been concluded the components in the ALT leakage path can be relied upon to maintain structural integrity. The seismic evaluation results are summarized in this section.

In summary, the seismic verification walkdowns and evaluations of piping/supports demonstrated the main steam line piping and components that comprise the ALT path were rugged, and would be able to perform the safety function of MSIV leakage control following an DBE. Therefore, it has been concluded the components in the ALT leakage path can be relied upon to maintain their structural integrity.

4.6.1 Seismic Walkdowns

The seismic walkdown scope included the ALT path from the outboard MSIVs to the main condenser and includes the piping, instrumentation, valves, and equipment that would be required to maintain the primary ALT drain pathway and secondary path boundaries. The seismic verification boundary was established to envelop the ALT pathway shown on Figure 1.

BFN Unit 1 walkdowns were conducted on the ALT pathway system piping, equipment and components consistent with the scope of the ALT path as shown as documented in Enclosure 7. Various design attributes of the as-installed equipment, piping, and tubing were reviewed and evaluated by a Seismic Walkdown Team to ensure that BFN installations were representative of the earthquake comparison database design practices and that components were free of known seismic vulnerabilities. Earthquake experience had previously identified that certain specific conditions were more prone to failure of piping and tubing systems and components during seismic events. Some of the conditions evaluated in the seismic walkdown reviews included:

- Piping, pipe supports and equipment design attributes;

- Seismic anchor movements;

- Seismic interaction issues (Class II/I issues and proximity issues); and

- Valve design attributes.

The above design attributes and conditions are discussed in more detail in Attachment 1, Section 3.1, of the enclosed Seismic Evaluation Report (Enclosure 8). Conditions which did not conform with the walkdown screening guidelines or which were judged by the Seismic Walkdown Team to require further review were documented as open items and listed as "outliers".

As a result of the walkdowns by the Seismic Walkdown Team, 54 outliers were identified. A brief summary of the identified outliers is listed on Table 5.1 of Enclosure 7. Examples of the outliers identified by the Seismic Walkdown Team are:

- Spans exceeding USA Standard Code for Pressure Piping (USAS) B31.1 recommendations,
- Differential displacements of main pipe and branch lines,
- Equipment anchorage deficiencies,
- Valve performance evaluation,
- Condenser and condenser anchorage evaluation, and
- Proximity and potential impact of piping with equipment, structural features, and other piping.

As documented in the table, 24 of the outliers were further evaluated by calculations and determined to be acceptable. The remaining outliers will be resolved prior to Unit 1 restart.

4.6.2 Verification of Earthquake Databases

To establish the applicability of NEDC-31858P-A, regarding usage of earthquake experience-based methodology for demonstrating the seismic ruggedness of the main steam piping and associated components for BFN, comparisons of the ground response spectra of selected database facilities with the BFN design basis ground spectrum were made. In general, the earthquake experience database sites have experienced strong ground motions that are in excess of the BFN DBE at the frequency range of interest (i.e., about 1 Hertz and above for piping and rigid range of equipment). Many of the database site ground motions envelope the conservatively estimated BFN DBE ground spectrum by large factors in various frequency bands within the 1 Hertz and above range. Of the 13 database facilities reviewed and approved by the NRC as documented in Reference 7, ten were selected for comparison to BFN and are shown in Attachment A, Figures 2-2 through 2-11, of the Seismic Evaluation Report (Enclosure 8).

Based on the comparison, it was concluded that the BFN DBE ground spectrum is generally bounded by the earthquake experience database sites at the frequencies of interest. Hence, the use of an earthquake experience-based approach for demonstrating the seismic ruggedness of non-seismically analyzed main steam piping and associated components at BFN is consistent with NEDC-31858P-A recommendations and with the limitations of the NRC Safety Evaluation.

The main steam piping and condensers in the earthquake experience database exhibited substantial seismic ruggedness, even when they typically were not designed to resist earthquakes. This is a common conclusion in studies of this type on other plant items such as welded steel piping in general, anchored equipment such as motor control centers, pumps, valves, structures, and so forth. That is, with limited exceptions, normal industrial construction and equipment typically have substantial inherent seismic ruggedness, even when they are not designed for earthquakes. No failures of the main steam piping were found. Anchored condensers have also performed well in past earthquakes with damage limited to minor internal tube leakage.

NEDC-31858P-A contains detailed discussions and comparisons of main steam piping and condenser design in several earthquake experience database sites and example General Electric (GE) Mark I, II, and III plants in the United States. The general conclusions of these comparisons are as follows:

- GE plant designs are similar to or more rugged than those in the earthquake experience database that exhibited good earthquake performance;
- The possibility of significant failure in GE BWR main steam piping or condensers in the event of an eastern United States DBE is highly unlikely; and that
- Any such failure would also be contrary to a large body of historical earthquake experience data and thus unprecedented.

Additional detailed discussions of the seismic experience database comparisons and applicability to BFN are in Attachment A, Section 2, of the Seismic Evaluation Report (Enclosure 8).

4.6.3 Seismic Analyses for ALT Drain Path

The majority of the MSIV ALT path piping systems and related components at BFN (those segments downstream of the outboard MSIVs and the outboard drain isolation valve) are located in the Turbine Building and are not designated as Seismic Class I systems. Thus, these piping systems were not typically seismically analyzed, but rather they were designed and installed to the requirements of USAS B31.1-1967.

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 5.1.2 of FRC Report (Enclosure 7) 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. The objective of the 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 DBE were evaluated in accordance with the GIP methodology as referenced in Section 5.1.2 of the FRC 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 (Engineered Safety Features [ESFs] - piping, pipe supports, components, etc.) are described in Appendix C of the BFN UFSAR, Structural Qualification of Subsystems and Components. 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 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 discussed above are the bases used to resolve, by calculation, maintenance and/or modifications, the identified outliers. These resolutions are summarized in Table 5.1 of the FRC Report.

As part of the plant specific seismic verification of the non-seismic designed ALT piping, and related supports and components using the earthquake experience-based approach outlined in NEDC-31858P-A, the following reviews were performed to demonstrate that the piping and related supports fall within the bounds of the earthquake experience database:

- Review of the design codes and standards, piping design parameters, and support configurations; and

- Seismic verification walkdowns to identify potential piping concerns.

The BFN ALT piping systems consist of welded steel pipe and standard support components and have support spacing that generally meet the USAS B31.1 recommended spans. Design bases for the portions of piping associated with the ALT pathway are further tabulated in Attachment A, Table 4-3, of the Seismic Evaluation Report (Enclosure 8). Attachment A, Table 4-4, also presents a general summary of the piping data that constitute the seismic experience data. Comparisons of BFN and selected database piping parameters are presented in Attachment A, Table 4-5. Overall, BFN piping design is similar to and well represented by those designs found in the experience database sites that have been shown to perform well in past earthquakes.

The seismic adequacy of the ALT piping was addressed by performing seismic verification walkdowns to identify specific design attributes associated with poor seismic performance following the guidelines outlined in Attachment A, Section 3.1, of the Seismic Evaluation Report. Additionally, bounding evaluations were previously performed for typical support configurations on BFN Units 2 and 3 using evaluation criteria discussed in Attachment A, Section 4.1, of the report. Table 4-6 summarizes the results of the support and anchorage evaluations for the selected bounding configurations.

In summary, the seismic evaluations, which consist of verification walkdowns, bounding support evaluations, and the resolution of the identified walkdown outliers, will provide reasonable assurance that the ALT drain path piping, related supports, and components will remain functional in the event of a DBE at BFN.

4.6.4 Seismic Dynamic Analysis of the Turbine Building

The BFN Turbine Building is classified as a Class II structure. Class II structures are generally defined as structures which are important to reactor operation, but are not essential for preventing an accident which would endanger the public health and safety, and are not essential for the mitigation of the consequences of accidents. Class II designated structures are also required to not degrade the integrity of any Class I structure.

The methodology to determine the soil amplification factors for the various Class I structures at BFN is defined in TVA Design Criteria, which require that structures founded on soil consider soil amplification. The soil amplification factors for applicable Class I structures are shown in BFN UFSAR 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 UFSAR, 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.

The Turbine Building houses the main turbines, generators, condensers, other auxiliary systems and balance-of-plant systems, and related piping. It is a reinforced concrete structure below the operating floor, elevation 617 feet, and is supported on steel H-piles. The turbine foundation is separated from the rest of the Turbine Building foundations and frames by means of a one inch expansion joint. The steel super-structure above elevation 617 feet is framed by transverse welded steel rigid frames which span approximately 107 feet. An expansion joint is provided between a two-bay frame for the first two units and a single bay frame for Unit 3.

Rolled shapes, plates, and bars used for structural steel are in accordance with ASTM A36. ASTM A325 bolts were used for field structural bolt connections. For welding, E70XX electrodes were used. The steel super-structure frames are braced to provide rigidity in the direction of the Reactor Building and provide support for the turbine cranes as well as the elaborate girt system. Frames are designed with fixed bases to resist lateral forces from the overhead cranes and wind loads due to 100 mile per hour wind in addition to supporting the vertical dead and live loads. The design of the steel super-structure is based on American Institute of Steel Construction (AISC) "Manual of Steel Construction", 6th Edition.

The compressive strength of the structural concrete is 3000 psi at 28 days cure time except that turbine building columns are 4000 psi. For evaluation/reanalysis of the structure, long-term concrete strength gain may be used. Reinforcing steel used is in accordance with ASTM A432, Grade 60 or ASTM A615 with f_y = 60,000 psi and E_s = 29 x 10^6 psi. Beams and slabs have been designed by American Concrete Institute (ACI) working stress methods and columns designed by working stress method, and checked by ACI ultimate strength design method using a load factor of 1.8.

Where masonry walls exist in the Turbine Building, they are generally used as removable shield walls or non-load bearing partition walls. Since non-reinforced masonry walls do not perform well during seismic events, masonry walls were specifically reviewed during the seismic verification walkdowns. Resolution of any masonry wall outliers typically focuses on relocating the ALT path components outside the potential zone of influence of the masonry wall.

Performance of the Turbine Building and other non-seismic structures during a seismic event is relevant to the modeling of MSIV leakage, only to the extent that the building structure and its internal components survive and not degrade the capabilities of the ALT pathway and condenser. The NEDC-31858P-A survey of this type of industrial structure has, in general, confirmed that excellent past seismic performance exists. There are no known cases of structural collapse of either turbine buildings at power stations or structures of a similar construction.

Based on the above design bases for the BFN Turbine Building, and the excellent seismic performance of similar types of industrial structures in past strong-motion earthquakes as documented in NEDC-31858P-A, it was determined that the BFN Turbine Building will remain structurally intact following a DBE. Additional discussions of the seismic assessment of the Turbine Building is included in Attachment A, Section 4.3, of the Seismic Evaluation Report (Enclosure 8). Table 4-7 of the report summarizes the design basis of the BFN Turbine Building and the applicable design codes used.

4.7 Seismic Analysis of the Condenser

The BFN condensers consist of three single-pass, single pressure, radial flow type surface condensers. Each condenser is located beneath each of the three low pressure turbines, and is structurally independent. Attachment A, Table 4-8, of the Seismic Evaluation Report lists the design data for the BFN condensers and an earthquake experience database site listed in the NEDC-31858P-A. In addition, design characteristic comparisons of the BFN condensers with the selected database condensers are shown in Figures 4-2 through 4-5. The BFN condenser design data is comparable to the data for the database site. The BFN condensers were also evaluated for structural integrity subject to seismic DBE loads (Attachment B of the Seismic Evaluation Report). Results of the evaluation indicate that the condenser shell stresses are small. Maximum stress ratios, based on AISC allowables, are 0.12 for combined axial and bending and 0.10 for shear.

The condenser support anchorage consists of a center key and six support feet that are arranged as shown in Attachment A, Figure 4-6 of the Seismic Evaluation Report. The center support is a fixed anchor and consists of a built-up wide flange H section embedded 4 feet into the concrete pedestal, which is connected to the Turbine Building base mat and welded to the bottom plate of the condenser. The support plates consist of 2 to 3 anchors of 2 to 2-1/2 inch diameter bolts. Each anchor bolt has greater than 5 feet nominal length with approximately 48 inches of embedment into the concrete pedestal, which is connected to the Turbine Building base mat. These supports are designed to resist vertical operating loads and are slotted radially from the center key to allow for thermal growth. Shear forces are transferred to the wide flange shaped anchor in the center and to the anchor bolts and shear keys to the support feet and carried through the concrete pedestal to the Turbine Building base mat.

The anchorage for the BFN condenser is comparable with the performance of the anchorages for similar condensers in the earthquake experience database. The shear areas of the condenser anchorage, in the directions parallel and transverse to the turbine generator axis, divided by the seismic demand, were used to compare with those presented in NEDC-31858P-A, and are shown in Attachment A, Figures 4-7 and 4-8 of the Seismic Evaluation Report, respectively. The BFN condenser anchorage shear area to seismic demand is substantially greater than the selected database sites. The condenser support anchorage was also evaluated and the results indicate that the combined seismic DBE and operational demand is less than the anchorage capacity based on the AISC allowables. Maximum stress ratios are 0.70 for bolt tension in the perimeter support feet and 0.86 for shear in the center support built-up section.

The above comparisons of the condenser seismic experience data and the anchorage capacity evaluations demonstrate that the conclusions presented in the NEDC-31858P-A, can be applied to the BFN condensers. That is, a significant failure of the condenser in the event of a DBE at BFN is highly unlikely and contrary to the large body of historical earthquake experience data.

To provide additional assurance that the BFN condensers would maintain structural integrity, a specific analysis was performed on the condenser subject to BFN seismic demand. Results of the analyses demonstrated that the condenser shell stresses are small, with maximum stress ratios based on AISC allowables of 0.12 for combined axial and bending and 0.10 for shear (Reference Attachment A, Section 4.4, of the Seismic Evaluation Report). Additionally, the condenser anchorage was also compared with the performance of condensers of the database site. The anchorage was demonstrated to be acceptable by seismic experience and by analytical methods. 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. Based on the above, it was concluded that the BFN condensers were acceptable.

5.0 REGULATORY SAFETY ANALYSIS

The Tennessee Valley Authority (TVA) is submitting a request to amend Operating License DPR-33 for the Browns Ferry Nuclear Plant (BFN) Unit 1. This Technical Specification (TS) would increase the allowable leakage rate criteria for the Main Steam Isolation Valves (MSIVs). This change request is based on the utilization of the Boiling Water Reactor Owners' Group (BWROG) methodology described in NEDC-31858P-A, BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems. TVA proposes to utilize the inherently rugged main steam drain lines to preferentially direct MSIV leakage to the main condenser.

This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this approach, the main steam lines, steam drain piping, and the main condenser are used to mitigate the consequences of an accident to limit potential exposures below the doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

5.1 No Significant Hazards Consideration

TVA has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment", as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

TVA proposes to utilize the main steam drain lines to preferentially direct MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this approach, the main steam lines, steam drain piping, and the main condenser are used to mitigate the consequences of an accident to limit potential doses below the limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

Seismic verification walkdowns and evaluations of bounding piping/supports were performed to demonstrate the main steam line piping and components that comprise the Alternate Leakage Treatment (ALT) path were rugged and able to perform the safety function of MSIV leakage control following a Design Basis Earthquake (DBE). Thus, it has been concluded the components in the MSIV alternate leakage treatment flow path can be relied upon to maintain structural integrity.

Therefore, the proposed amendment does not involve changes to structures, components, or systems which would affect the probability of an accident previously evaluated in the Browns Ferry Updated Final Safety Analysis Report (UFSAR).

A plant-specific radiological analysis has been performed to assess the effects of the proposed increase in MSIV leakage acceptance criteria in terms of off-site doses and main control room dose. The analysis shows the dose contribution from the proposed increase in leakage acceptance criteria is acceptable compared to doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed changes require the use of the main steam piping and the condenser to process MSIV leakage. This additional function does not compromise the reliability of these systems. They will continue to function as intended and not be subject to a failure of a different kind than previously considered. In addition, MSIV functionality will not be adversely impacted by the increased leakage limit. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change to Surveillance Requirement 3.6.1.3.10, to increase the allowable MSIV leakage, does not involve a significant reduction in the margin of safety. The allowable leak rate specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass containment. The results of the re-analysis supporting these changes were evaluated against the dose limits contained in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel. Sufficient margin relative to the regulatory limits is maintained even when conservative assumptions and methods are utilized. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The radiological consequences of the design basis LOCA accident at BFN Unit 1 will be less than the dose guidelines of 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

Therefore, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration; (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite; or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. TVA letter, T.E. Abney to NRC, dated September 28, 1999, "Browns Ferry Nuclear Plant (BFN) - Units 2 and 3 - Technical Specification (TS) Change TS-399 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits and Exemption from 10 CFR 50 Appendix J."

2. NRC letter, W.O. Long to J.A. Scalice, dated November 23, 1999, "Browns Ferry Units 2 and 3, Main Steam Isolation Valve Leak Rate Limits, Request for Additional Information (TAC Nos. MA6405, MA6406, MA6815 and MA6816)."

3. TVA letter, T.E. Abney to NRC, dated February 4, 2000, "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)."

4. NRC letter, W.O. Long to J.A. Scalice, dated March 14, 2000, "Issuance of Amendments Regarding Limits on Main Steam Isolation Valve Leakage (TAC Nos. MA6405 and MA6406)."

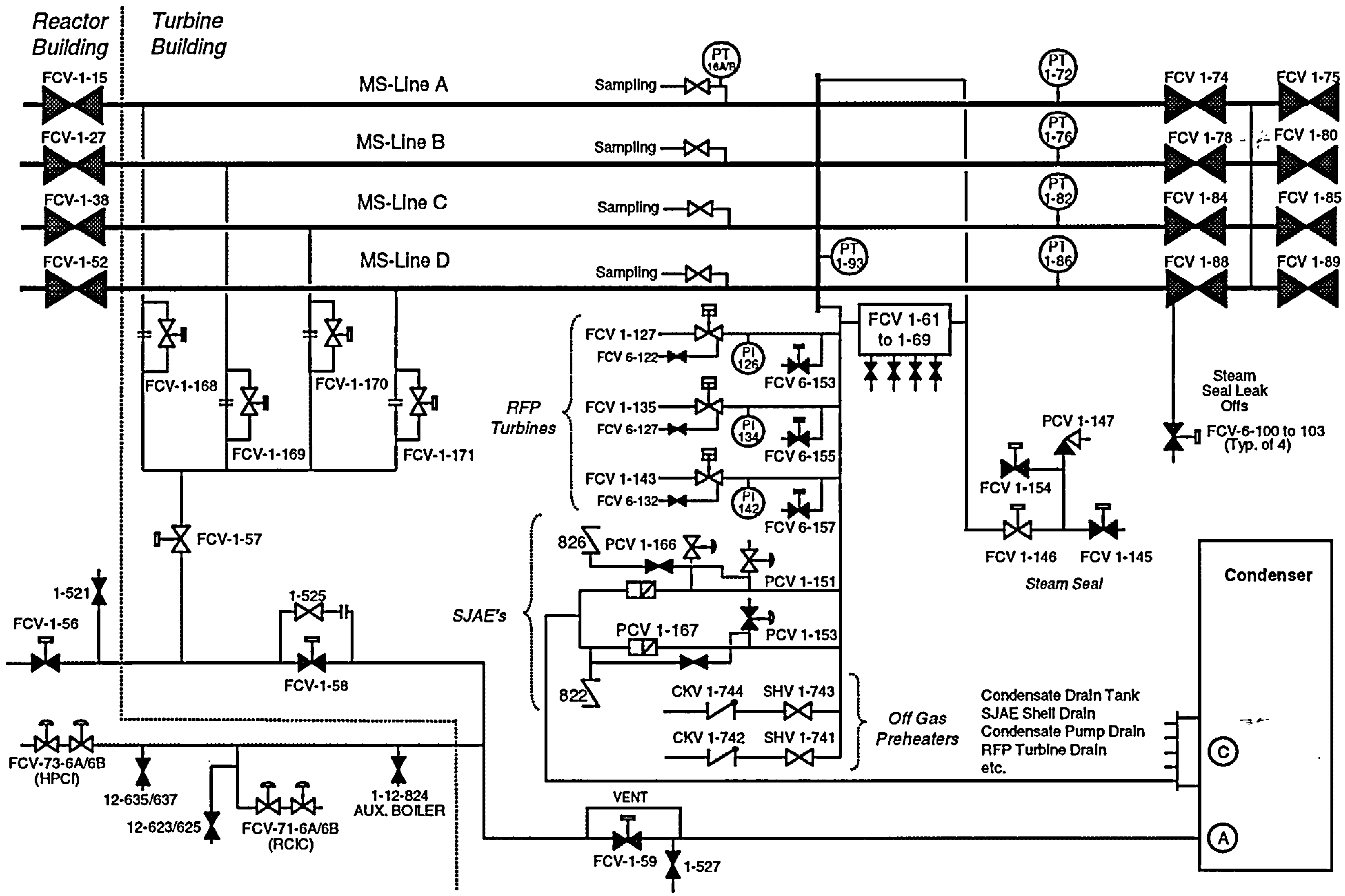
5. BWROG letter, W. G. Warren to NRC, dated November 22, 1999, "Transmittal of Approved GE Licensing Topical Report, NEDC-31858P-A dated August 1999.

6. NRC letter to the BWROG, F. M. Akstulewicz to T. A. Green, dated March 3, 1999, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems," September 1993."

7. TVA letter, T.E. Abney to NRC, dated May 17, 2004, Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Response to Request for Additional Information (RAI) and Unit 1 Analysis Results Related to Technical Specifications (TS) Change No. TS-405 - Alternative Source Term (AST) (TAC Nos. MB5733, MB5734, MB5735).

8. TVA letter, T.E. Abney to NRC, dated May 17, 2004, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2 and 3 - Response to Request for Additional Information (RAI) and Unit 1 Analysis Results Related to Technical Specifications (TS) Change No. TS-405 - Alternative Source Term (AST) (TAC Nos. MB5733, MB5734, MB5735)."

FIGURE 1 BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY



ENCLOSURE 2

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on a simulated instrument line break signal.	18 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~ **Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.**	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program

ENCLOSURE 3

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

ASSOCIATED TECHNICAL SPECIFICATION BASES CHANGES
(MARK-UP)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in References 1 and 5 are based on leakage that is less than the specified leakage rate. ~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate.~~ The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

Leakage through each MSIV must be ≤ 100 scfh when tested at ≥ P_t (25 psig). The combined leakage rate for all four main steam lines must be ≤ 150 scfh when tested at ≥ 25 psig in accordance with the Primary Containment Leakage Rate Testing Program. If the leakage rate through an individual MSIV exceeds 100 scfh, the leakage rate shall be restored below the alarm limit value as specified in the Containment Leakage Rate Testing Program referenced in TS 5.5.12. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate.

SR 3.6.1.3.11

Surveillance of water tested lines ensures that sufficient inventory will be available to provide a sealing function for at least 30 days at a pressure of 1.1 Pa. Sufficient inventory ensures there is no path for leakage of primary containment atmosphere to the environment following a DBA. Leakage from containment isolation valves that terminate below the suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available as described in 10 CFR 50, Appendix J, Option B.

(continued)

ENCLOSURE 4

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

PROPOSED TECHNICAL SPECIFICATION CHANGES (CLEAN PAGES)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.5	Verify the isolation time of each power operated, automatic PCIV, except for MSIVs, is within limits.	In accordance with the Inservice Testing Program
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	18 months
SR 3.6.1.3.8	Verify each reactor instrumentation line EFCV actuates to the isolation position on a simulated instrument line break signal.	18 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	18 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify leakage rate through each MSIV is ≤ 100 scfh and that the combined leakage rate for all four main steam lines is ≤ 150 scfh when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage through water tested lines that penetrate primary containment are within the limits specified in the Primary Containment Leakage Rate Testing Program.	In accordance with the Primary Containment Leakage Rate Testing Program

ENCLOSURE 5

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

ASSOCIATED TECHNICAL SPECIFICATION BASES CHANGES
(CLEAN PAGES)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.4).

SR 3.6.1.3.10

The analyses in References 1 and 5 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 100 scfh when tested at ≥ P_t (25 psig). The combined leakage rate for all four main steam lines must be ≤ 150 scfh when tested at ≥ 25 psig in accordance with the Primary Containment Leakage Rate Testing Program. If the leakage rate through an individual MSIV exceeds 100 scfh, the leakage rate shall be restored below the alarm limit value as specified in the Containment Leakage Rate Testing Program referenced in TS 5.5.12. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is specified in the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.11

Surveillance of water tested lines ensures that sufficient inventory will be available to provide a sealing function for at least 30 days at a pressure of 1.1 Pa. Sufficient inventory ensures there is no path for leakage of primary containment

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.1.3.11 (continued) atmosphere to the environment following a DBA. Leakage from containment isolation valves that terminate below the suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available as described in 10 CFR 50, Appendix J, Option B. Leakage through valves in closed loop seismic class I lines that are considered as extensions of primary containment present no potential for leakage to the environment. Leakage from these valves will be measured, but will be excluded when computing the total leakage. This leakage will be reported as required by the Primary Containment Leakage Rate Testing Program.
REFERENCES	1. FSAR, Section 14.6. 2. BFN Technical Instruction (TI), 0-TI-360. 3. 10 CFR 50, Appendix J, Option B. 4. FSAR, Section 5.2. 5. FSAR, Section 14.6.5. 6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993. 7. FSAR Table 5.2-2.

ENCLOSURE 6

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION (TS) CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

ASSOCIATED EXEMPTION TO 10 CFR 50, APPENDIX J

Pursuant to 10 CFR 50.12, an exemption is being requested which would allow exclusion of the MSIVs from the overall integrated leakage rate test acceptance criteria of Paragraph III.A of Appendix J, Option B, as measured during the performance of the Type A test and for the combined local leak rate tests (Types B and C) as defined in 10 CFR 50, Appendix J, Option B, Paragraph III.B.

I. APPLICABLE RULE

The pertinent applicable rule is 10 CFR 50, Appendix J, Option B, Paragraphs III.A and III.B.

II. REQUESTED EXEMPTION

TVA requests an exemption from the requirements of 10 CFR 50, Appendix J, Option B, Section III.A. This exemption would allow exclusion of the MSIV leakage from the overall integrated leakage rate measured when performing a Type A test.

TVA also requests an exemption from the requirements of 10 CFR 50, Appendix J, Option B, Section III.B. This exemption would allow exclusion of the MSIV leakage from the combined containment local leak rate tests (Types B and C).

III. JUSTIFICATION AND ASSESSMENT OF SAFETY IMPACT

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 La 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 La 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 La acceptance criteria for the CILRT, could account for approximately 18% of the 0.75 La 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 La. In analyzing the use of the Alternate Leakage Treatment (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.

The current TS allowable MSIV leakage rate (11.5 Standard Cubic Feet Hour [scfh]) is extremely small considering the valve's physical size and operating characteristics (large size and fast acting). TVA has proposed a TS change to increase the allowable leakage rate using the methodology described in Boiling Water Reactor Owners' Group (BWROG) report NEDC-31858P-A.

NEDC-31858P-A, 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.

Specifically, TVA proposes to utilize the main steam drain lines to direct MSIV leakage to the main condenser. This drain path takes advantage of the large volume of the steam lines and condenser to provide holdup and plate-out of fission products that may leak through the closed MSIVs. In this approach, the main steam lines, steam drain piping, and the main condenser are used to mitigate the consequences of an accident to limit potential off-site exposures below those specified in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

Paragraph III.A of Appendix J, Option B requires that the overall integrated leakage rate, as measured during the performance of the Type A test, be less than the performance criteria (La) with margin as specified in the TS. Appendix J, Option B defines the overall integrated leakage rate as the total leakage rate through all tested leakage paths, including containment welds, valves, fittings, and components that penetrate the containment system. For BFN, TS 5.5.12 defines the leakage rate acceptance criteria as follows: Type A leakage shall be less than or equal to La. During the first unit startup following testing in accordance with this program, acceptance criterion is 0.75La for the Type A test.

Paragraph III.B of Appendix J, Option B requires that the combined leakage rate for all penetrations and isolation valves, as measured during local leak rate tests (Types B and C) be less than the performance criteria (La) with margin as specified in the TS. For BFN, TS 5.5.12 defines the leakage rate acceptance criteria as less than or equal to 0.60 La for the Types B and Type C tests.

Limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the postulated design basis Loss of Coolant Accident (LOCA) analysis as evaluated in Chapter 14.6 of the Updated Final Safety Analysis Report (UFSAR). The radiological impact considers both the effect of containment leakage in terms of the doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel. In general, leakages from the containment volume are contained in the reactor building (secondary containment), filtered through the Standby Gas Treatment System (SBGT), and released at an elevated point through the plant stack. Leakages through the MSIVs are contained in the main steam piping before eventually entering the turbine building.

In support of the TS change, a separate dose calculation was performed which accounts for the radiological effects from the MSIV leakage and from other primary containment leakages following the worst case LOCA. The MSIV leakage was evaluated using the main steam line drain path as recommended in NEDC-31858P-A, which considers fission product removal by holdup and plate-out in the leak path and main condenser. Additionally, the treatment path was evaluated and determined to be seismically rugged, even though the probability of a LOCA and concurrent seismic event are extremely low.

To further justify the capability of the main steam drain piping and condenser as an alternate pathway, BWR design was compared with earthquake experience data on the performance of non-seismically designed piping and condensers during past earthquakes. It was concluded that the possibility of a failure which would cause a loss of main steam piping or condenser integrity in BWRs main steam systems was highly unlikely, and such a failure would be contrary to historical experience for earthquakes.

As previously noted, the basis for the leak rate test acceptance criteria is to establish the maximum leak rate assumed in the FSAR LOCA analysis. The pertinent LOCA analysis has been revised to incorporate the increased MSIV leak rate criteria as a separate source term release pathway. The MSIVs will continue to be tested in accordance with the Primary Containment Leak Rate Testing Program to ensure leakages remain within the revised LOCA analysis assumptions.

There is sufficient remaining dose margin based on the revised analysis to allow for possible degradation of the MSIV leakage barrier between leakage tests. The TS Bases will also be revised to require that any MSIV exceeding 100 scfh leak rate will be refurbished to a value prescribed in the Primary Containment Leakage Rate Testing Program.

Therefore, the proposed exemptions to the requirements of 10 CFR 50, Appendix J are acceptable and do not compromise the safety design basis of the primary containment or the overall purpose of performing leak rate testing.

10 CFR 50 authorizes the Commission to grant exemptions from the requirements of regulations provided that:

- The exemption is authorized by law;

- The exemption does not present an undue risk to the public health and safety;

- The requested exemption will not endanger the common defense and security; and

- Special circumstances are present as defined in 10 CFR 50.12(a)(2).

A discussion of how this exemption request satisfies each of these requirements is provided below.

IV. AUTHORIZED BY LAW

TVA was issued its Operating License for BFN under the provisions of Section 104.b of the Atomic Energy Act. Operating Licenses issued by the Commission pursuant to Section 104.b are not limited, by statute, to specific methods of testing primary containment integrity. Thus, the Commission is authorized by law to exempt TVA from the requirements of 10 CFR 50 Appendix J.

V. NO UNDUE RISK TO THE PUBLIC HEALTH AND SAFETY

The revised MSIV leakage rate has been incorporated into the radiological analysis for a postulated LOCA as an addition to the designed containment leak rate. The analysis demonstrates an acceptable increase in the dose exposures previously calculated for the control room and off-site. The revised calculations remain well within the guidelines of 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

In addition, TS 3.6.1.3.10 provides for allowable MSIV leak rates, which assure that the MSIV function is not compromised. Finally, potential MSIV leakage is subject to holdup and plate-out in the main steam piping and condenser, thus minimizing its affect on the total dose released.

The implementation of the proposed MSIV treatment method will provide BFN with a reliable capability to process MSIV leakage. Therefore, the proposed exemption presents no undue risk to public health and safety.

VI. BE CONSISTENT WITH THE COMMON DEFENSE AND SECURITY

The Commission's Statement of Considerations in support of the exemption rule note with approval the explanation of this standard as set forth in Long Island Lighting Company (Shoreham Nuclear Power Station, Unit 1) LBP-84-45, 20 NRC 1343, 1400 (October 29, 1984). There, the term "common defense and security" refers principally to the safeguarding of special nuclear material, the absence of foreign control over the applicant, the protection of Restricted Data, and the availability of special nuclear material for defense needs. The granting of the requested exemption will not affect any of these matters and, thus, the requested exemption is consistent with the common defense and security.

VII. MUST ENTAIL SPECIAL CIRCUMSTANCES

According to NRC regulations, special circumstances are present if any one of six different cases cited in 10 CFR 50.12(a)(2) are present. TVA submits that the existence of special circumstances (ii), (iii) and (iv) are applicable for this exemption.

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not needed to satisfy the underlying purpose of the rule.

The underlying purpose of the rule is to limit radiological releases to within the off-site and control room dose limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

Paragraph III.A of Appendix J, Option B requires that the overall integrated leakage rate, as measured during the performance of the Type A test, be less than the performance criteria (L_a) with margin as specified in the TS. Appendix J, Option B defines the overall integrated leakage rate as the total leakage rate through all tested leakage paths, including containment welds, valves, fittings, and components that penetrate the containment system. For BFN, TS 5.5.12 defines the leakage rate acceptance criteria as follows: Type A leakage shall be less than or equal to L_a. During the first unit startup following testing in accordance with this program, acceptance criterion is 0.75 for the Type A test. Paragraph III.B of Appendix J, Option B requires that the combined leakage rate for all penetrations and isolation valves, as measured during local leak rate tests (Types B and C) to meet the acceptance criterion as specified in the TS (0.60 L_a). Currently, MSIV leakage is included in the test results for these criteria.

Including this leakage in the test acceptance criteria is not necessary to achieve the underlying purpose of the rule because MSIV leakage is not directed into the secondary containment. Instead, the MSIVs leakage is directed through the main steam drain piping into the condenser. Compliance with 10 CFR 50, Appendix J, Type C acceptance criterion is not necessary because a specific leak rate is already in TS SR 3.6.1.3.10. The radiological safety analysis has been revised to assess the specific consequences of MSIV leakage following a design basis LOCA. The analysis has demonstrated that the revised LOCA doses are well within the doses limits prescribed in 10 CFR 50.67(b)(2)(i) for the exclusion area, 10 CFR 50.67(b)(2)(ii) for the low population zone, and in 10 CFR 50.67(b)(2)(iii) for control room personnel.

(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

Strict compliance with Paragraphs III.A and III.B of Appendix J, Option B requires that the combined leakage rate for Type A tests and leakage for all penetrations and isolation valves, as measured during local leak rate tests (Types B and C) meet the acceptance criterion specified in the TS. This results in undue hardship and other costs that are significantly in excess of those contemplated when the regulation was adopted. The proposed increase in the MSIV allowable leak rate will not be practical if the MSIV leak rate results are included in the overall test acceptance criteria. Compliance requires unnecessary repair and retesting of the MSIVs.

This significantly impacts the maintenance work load during plant outages and can result in extending outages.

Examples of these maintenance induced defects include machine-induced seat cracking, machining of guide ribs, excessive pilot valve seat leakage, and mechanical defects induced by assembly and disassembly. By not having to disassemble the valves and refurbish them for minor leakage, BFN avoids introducing one of the root causes of recurring leakage. Industrial experience suggests that, by attempting to correct non-existing or minimal defects in the valves, it is likely that some actual defects may be introduced that lead to later leak test failures.

In addition, the frequent maintenance work results in needless dose exposures to maintenance personnel which are inconsistent with As Low As Reasonably Achievable (ALARA) principles and also leads to additional economic burdens.

(iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption.

Enclosure 1 contains an application for a license amendment to change TS to increase the allowable MSIV leak rate. This application is partially based on the fact that the current limit is too restrictive, and results in excessive MSIV maintenance and repair, which may result in additional MSIV failures. The proposed change will benefit the public health and safety by reducing the potential for MSIV failures, and thus keeping the MSIVs within the radiological analysis values.

TVA proposes to implement an alternate reliable and effective treatment path (the main steam lines and condenser) for MSIV leakage treatment. This treatment method is effective to treat MSIV leakage over an expanded operating range without exceeding the off-site and control room dose limits. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any logic control or interlocks.

The method is consistent with the philosophy of protection by multiple leak tight barriers used in containment design for limiting fission product relapse to the environment. Therefore, the proposed method is highly reliable for MSIV treatment. The implementation will provide BFN with a capability to process MSIV leakage, and will also provide a uniform basis for establishing a plant-specific MSIV leakage limit.

The exemption from Appendix J requirements for MSIV leakage rates is required so that BFN can operate with the proposed TS increased allowable MSIV leakage rates. This results in reduced radiological exposure to plant personnel, greater overall MSIV reliability, and significant economic benefit to TVA and its customers as a result of reduced plant outage durations. These benefits will compensate for any decrease in safety that may result from the granting of the exemption.

Thus, as discussed above, special circumstances exist warranting the grant of the exemption.

VIII. ENVIROMENTAL IMPACT

The proposed exemption has been analyzed and determined not to cause additional construction or operational activities which may significantly affect the environment. It does not result in a significant increase in any adverse environmental impact previously evaluated, result in a significant change in effluents or power levels, or affect any matter not previously reviewed by the NRC that may have a significant environmental impact.

The proposed exemption does not alter the land use for the plant, any water uses or impacts on water quality, air or ambient air quality. The proposed action does not affect the ecology of the site and does not affect the noise emitted by the site. Therefore, the proposed exemption the does not affect the previous analysis of environmental impacts.

ENCLOSURE 7

BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1
TECHNICAL SPECIFICATION CHANGE (TS 436) -
INCREASED MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE RATE LIMITS

MAIN STEAM SEISMIC RUGGEDNESS EVALUATION

EXECUTIVE SUMMARY

This report summarizes the seismic ruggedness verification program for increasing the Main Steam Isolation Valve (MSIV) leakage rate limits at Browns Ferry Nuclear Plant Unit 1 (BFN-1). Key engineering attributes of the seismic verification program consisted of the following:

- Review of the MSIV seismic verification boundaries;
- In-plant screening walkdown evaluations and identification of potential outliers;
- Further evaluations and resolution of potential outliers;
- Recommendations for plant modifications to resolve outliers;
- Work order requests to address general maintenance and housekeeping items.

BFN-1 MSIV seismic ruggedness verification program was performed in accordance with the recommendations of the General Electric BWR Owners Group Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems. MSIV leakage boundaries were established to ensure that the main steam piping from the outboard MSIV's to the turbine stop valves and the main steam drain line from the outboard MSIV's to and including the condenser itself are capable of being a pressure retaining boundary following a Design Basis Earthquake (DBE).

In-plant screening walkdown evaluations of piping and associated components within the seismic verification boundaries were performed in accordance with Walkdown Instruction WI-BFN-0-CEB-07, *"Engineering Walkdown Instruction for MSIV Seismic Ruggedness Verification."* A total of 15 subsystems within the seismic boundaries were included. Screening evaluations focused on key design attributes and seismic issues such as pipe spans and support integrity; seismic interaction issues including proximity impact and falling concerns; differential displacement of structures, equipment and piping; performance of seismic verification boundary components, among others. Screening tools such as seismic deflection estimates and charts for various plant features; pipe flexibility and seismic anchor movement evaluation charts; support and anchorage capacity screening charts; and others were used in the in-plant screening walkdown evaluations. Certain configurations identified during the in-plant screening walkdowns as not meeting the screening criteria were documented in the Potential Outlier Sheet (POS) as potential

outliers and for further evaluation and disposition. Walkdown results, including a total of 54 potential outliers identified, are documented in the Walkdown Data Packages (WDP's) for the respective subsystems.

Potential outliers identified during the in-plant screening walkdowns were further evaluated to the acceptance criteria of TVA Design Criteria BFN-50-C-7306, *"Qualification Criteria for Seismic Class II Piping, Pipe Supports, and Components."* Further evaluations and bounding analyses of these potential outliers consisted of hand calculations using basic engineering mechanics techniques for simple configurations, and rigorous piping analyses (TPIPE computer program) for more complex piping configurations. A total of 15 outliers were found to have not met the acceptance criteria. Plant modifications were designed and several Design Change Notices (DCN's) were issued to implement the changes so that all of these concerns were resolved. Furthermore, 15 maintenance and/or housekeeping items were also identified for corrective actions. Maintenance work order requests were issued to address these items.

Overall program results for the MSIV seismic ruggedness verification program for Browns Ferry Unit 1 are summarized in the following table:

Sub-system No.	Subsystem Description	No. of Potential Outliers	No. of Design Mods	No. of Maint. Items
1	MS Drain Line – Turbine Bldg. Main Steam Tunnel	4	0	0
2	MS Lines – Turbine Bldg.	2	0	1
3	MS Drain Line – Reactor Bldg. Main Steam Vault	2	0	0
4	HPCI/RCIC/Aux. Boiler Drains	8	2	1
5	MS Pressure Transmitters PT 1-72, 76, 82, 86 & 93	6	4	2
6	MS Sample Lines to Sampling Station	4	2	2
7	MS Bypass	1	0	0
8	MS Stop Valve Above Seat Drains	3	0	3
9	MS to Steam Seal Regulator	6	1	1
10	Steam Supply to RFP Turbines	6	1	2

Sub-system No.	Subsystem Description	No. of Potential Outliers	No. of Design Mods	No. of Maint. Items
11	Steam Supply to Steam Jet Air Ejectors (SJAE's)	6	2	1
12	Steam Supply to Off-Gas Preheaters	1	1	0
13	SJAE's Drain to Condenser	1	1	0
14	MS Drain Line (Turbine Bldg.) to Condenser	3	1	2
15	Condenser	1	0	0
	Total	**54**	**15**	**15**

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1. INTRODUCTION

This report summarizes the work performed for supplemental plant-specific Main Steam Piping Seismic Verification. The verification program was performed in accordance with the recommendation of the General Electric BWR Owners Group Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems (Reference 7-1). The objective of the supplemental plant specific walkdown verification was to identify specific design conditions associated with poor piping and component seismic performance. The walkdown was directed toward identification of the following specific phenomenon and issues:

- Piping Support Integrity Issues
- Seismic Interaction Issues of:
    - Failure and Falling (II/I Concerns)
    - Displacement and Proximity Impact Issues
- Differential Displacement of Structures, Equipment and Piping
- Performance of Seismic Verification Boundary Components

1.1 Report Organization

This report is organized as follows. Chapter 2 presents the scope of the program. Technical bases and evaluation approach for in-plant screening evaluations, including walkdown implementation, are described in Chapter 3. Summary of in-plant screening walkdowns and evaluations, including open items identified, is presented in Chapter 4, while that of potential outlier resolution is in Chapter 5. Overall summary and recommendations, including plant modifications and maintenance work orders, are provided in Chapter 6. References are listed in Chapter 7 of the report.

There are 3 appendices to this report. Appendix A contains a listing of walkdown data packages (WDP) which document the in-plant screening evaluations performed by the Walkdown Teams. Appendix B contains a list of engineering calculations which document the resolution of potential outliers by further analyses. A list of calculations containing the engineering design of plant modifications for the resolution of BFN-1 MSIV outliers, as well as maintenance work order requests, are provided in Appendix C.

2. PROGRAM SCOPE

The walkdown scope included the drain path that will be established to convey leakage past the outboard Main Steam Isolation Valves (MSIV's) to the isolated condenser and includes piping, instrumentation, valves and equipment that would be required to maintain the drain pathway.

2.1 Seismic Verification Boundary

BFN-1 MSIV Leakage Containment Boundaries are described in Reference 7-2, and are shown in Figure 2-1. The associated flow diagrams are listed in Table 2-1, and the piping isolation boundaries defining the seismic verification boundary are shown in Table 2-2. Note that Unit 1 seismic boundaries are generally similar to those for Units 2 and 3.

The scope of the seismic verification walkdown presented in this report generally consists of the following portions of the Main Steam (MS) system beyond the outboard MSIV's:

1. Main Steam drain path to the condenser for any leakage past the isolated outboard MSIV's.
2. Main Steam piping from the outboard MSIV's to the Main Steam Stop Valves (MSV's).
3. Main Steam bypass piping from the Main Steam lines to the Bypass Valve Chest.
4. Main Condensers.
5. Additional piping and instrumentation within the Seismic Verification Boundary includes:
    - Stop Valve Above Seat Drains to Condenser
    - Steam Sample System
    - HPCI/RCIC Steam Drains to Main Steam
    - Auxiliary Boiler Drains to Main Steam
    - Main Steam Instrumentation
    - Steam to Steam Seal Regulator
    - Main Steam Supply to Reactor Feed Pumps
    - Main Steam Supply to Steam Jet Air Ejectors
    - Main Steam Supply to Off-Gas Preheaters

2.2 Seismic Walkdown Scope

The scope of the seismic verification walkdown included consideration of design conditions which in past earthquake experience have been associated with piping damage and could contribute to pressure boundary failure and inventory release. These conditions include support failure, falling of non-seismically designed plant features (II/I), proximity impact, and differential seismic anchor motion of structures, piping or equipment. The scope and extent of these conditions are consistent with the guidelines as specified in Reference 7-1 and are described in greater details in Chapters 3 and 4 of this report.

BFN-1 MSIV seismic ruggedness verification walkdowns were performed considering a total of 15 subsystems or portions within the MSIV seismic verification boundary, as follows:

1. Main Steam Drain Line – Turbine Bldg. Main Steam Tunnel
2. Main Steam – Turbine Bldg.
3. Main Steam Drain Line – Reactor Bldg. Main Steam Vault
4. HPCI/RCIC/Aux. Boiler Drains
5. Main Steam Pressure Transmitters PT 1-72, 76, 82, 86 & 93
6. Main Steam Sample Lines to Sampling Station
7. Main Steam Bypass
8. Main Steam Stop Valve Above Seat Drains
9. Main Steam to Steam Seal Regulator
10. Steam Supply to RFP Turbines
11. Steam Supply to Steam Jet Air Ejectors (SJAE's)
12. Steam Supply to Off-Gas Preheaters
13. SJAE's Drain to Condenser
14. Main Steam Drain Line (Turbine Bldg.) to Condenser
15. Condenser

Figures 2-2 to 2-16 highlight the extent of the seismic walkdown scope for each of the above 15 subsystems or portions within the BFN-1 MSIV seismic verification boundary

Table 2-1
BFN UNIT 1 MECHANICAL FLOW DIAGRAMS

Drawing Number	System Description
1-47E801-1	Main Steam
1-47E801-2	Main Steam
1-47E805-3	Heater Drains, Vents and Misc. Piping
`1-47E807-1	Turbine Drains and Miscellaneous Piping
1-47E807-2	Turbine Drains and Miscellaneous Piping
1-47E812-1	High Pressure Coolant Injection System
1-47E813-1	Reactor Core Isolation Cooling System
0-47E815-1 1-47E815-3	Auxiliary Boiler System
1-47E610-43-1	Sampling and Water Quality System (Mechanical Control Diagram)

Table 2-2
BFN UNIT 1 MSIV LEAKAGE BOUNDARY POINTS

Leakage Boundary Point	Flow Diagram/ Drawing	Comment
1-FCV-1-15	1-47E801-1	Outboard MSIV for Main Steam Line A
1-FCV-1-27	1-47E801-1	Outboard MSIV for Main Steam Line B
1-FCV-1-38	1-47E801-1	Outboard MSIV for Main Steam Line C
1-FCV-1-52	1-47E801-1	Outboard MSIV for Main Steam Line D
1-FCV-1-56	1-47E801-1	Outboard containment isolation valve for Primary Containment steam drains
1-1-521 1-1-527	1-47E801-1	Normally closed Main Steam Drain manual isolation valves
1-43-631	1-47E610-43-1	Normally closed Main Steam Sample System manual isolation valve
1-43-631A	1-47E610-43-1	Normally closed Main Steam Sample System manual isolation valve
1-43-632	1-47E610-43-1	Normally closed Main Steam Sample System manual isolation valve
1-FCV-1-74	1-47E801-2	Main Turbine Stop Valve for Steam Line A
1-FCV-1-78	1-47E801-2	Main Turbine Stop Valve for Steam Line B
1-FCV-1-84	1-47E801-2	Main Turbine Stop Valve for Steam Line C
1-FCV-1-88	1-47E801-2	Main Turbine Stop Valve for Steam Line D
1-FCV-1-61 1-FCV-1-62 1-FCV-1-63 1-FCV-1-64 1-FCV-1-65 1-FCV-1-66 1-FCV-1-67 1-FCV-1-68 1-FCV-1-69	1-47E801-2	Main Steam Bypass Valve Chest
1-FCV-1-127	1-47E801-2	RFP Turbine A Stop Valve
1-FCV-1-135	1-47E801-2	RFP Turbine B Stop Valve
1-FCV-1-143	1-47E801-2	RFP Turbine C Stop Valve
1-FCV-6-153	1-47E807-2	Normally closed motor operated isolation valve – RFP Turbine A
1-FCV-6-155	1-47E807-2	Normally closed motor operated isolation valve – RFP Turbine B
1-FCV-6-157	1-47E807-2	Normally closed motor operated isolation valve – RFP Turbine C
1-FCV-6-122	1-47E807-2	Normally closed motor operated isolation valve – RFP Turbine A
1-FCV-6-127	1-47E807-2	Normally closed motor operated isolation valve – RFP Turbine B

Table 2-2 (Cont.)
BFN UNIT 1 MSIV LEAKAGE BOUNDARY POINTS

Leakage Boundary Point	Flow Diagram/ Drawing	Comment
1-FCV-6-132	1-47E807-2	Normally closed motor operated isolation valve – RFP Turbine C
1-PCV-1-151	1-47E801-2	Normally open air operated isolation valve – SJAE A
1-PCV-1-166	1-47E801-2	Normally open air operated isolation valve – SJAE A
1-6-826	1-47E805-3	Check Valve – SJAE A
1-PCV-1-153	1-47E801-2	Normally open air operated isolation valve – SJAE B
1-PCV-1-167	1-47E801-2	Normally open air operated isolation valve – SJAE B
1-6-822	1-47E805-3	Check Valve – SJAE B
1-SHV-1-741	—	New manual isolation valve for Off-Gas Preheater – per DCN 51112
1-CKV-1-742	—	New check valve for Off-Gas Preheater – per DCN 51112
1-SHV-1-743	—	New manual isolation valve for Off-Gas Preheater – per DCN 51112
1-CKV-1-744	—	New check valve for Off-Gas Preheater – per DCN 51112
1-FCV-73-6B	1-47E812-1	Normally open air operated isolation valve - HPCI
1-FCV-71-6B	1-47E813-1	Normally open air operated isolation valve - RCIC
1-12-635	1-47E815-3	Normally closed manual isolation valve - Aux. Boiler
1-12-637	1-47E815-3	Normally closed manual isolation valve - Aux. Boiler
1-12-623	1-47E815-3	Normally closed manual isolation valve - Aux. Boiler
1-12-625	1-47E815-3	Normally closed manual isolation valve - Aux. Boiler
1-12-824	0-47E815-1	Normally closed manual isolation valve - Aux. Boiler
1-FCV-1-145	1-47E807-2	Normally closed motor operated isolation valve – Steam Seal Regulator
1-FCV-1-154	1-47E807-2	Normally closed motor operated isolation valve – Steam Seal Regulator
1-PCV-1-147	1-47E807-2	Air operated pressure control valve/relief valve – Steam Seal Regulator
1-FCV-6-100	1-47E807-1	Normally closed motor operated isolation valve - Stop valve above seat drains
1-FCV-6-101	1-47E807-1	Normally closed motor operated isolation valve - Stop valve above seat drains
1-FCV-6-102	1-47E807-1	Normally closed motor operated isolation valve - Stop valve above seat drains
1-FCV-6-103	1-47E807-1	Normally closed motor operated isolation valve - Stop valve above seat drains
Condenser A	—	The condenser is the ultimate boundary for the MSIV leakage path.
Condenser B	—	The condenser is the ultimate boundary for the MSIV leakage path.
Condenser C	—	The condenser is the ultimate boundary for the MSIV leakage path.
Misc. Components	1-47E801-1 1-47E801-2	Miscellaneous test, vent, drain and instrument connections

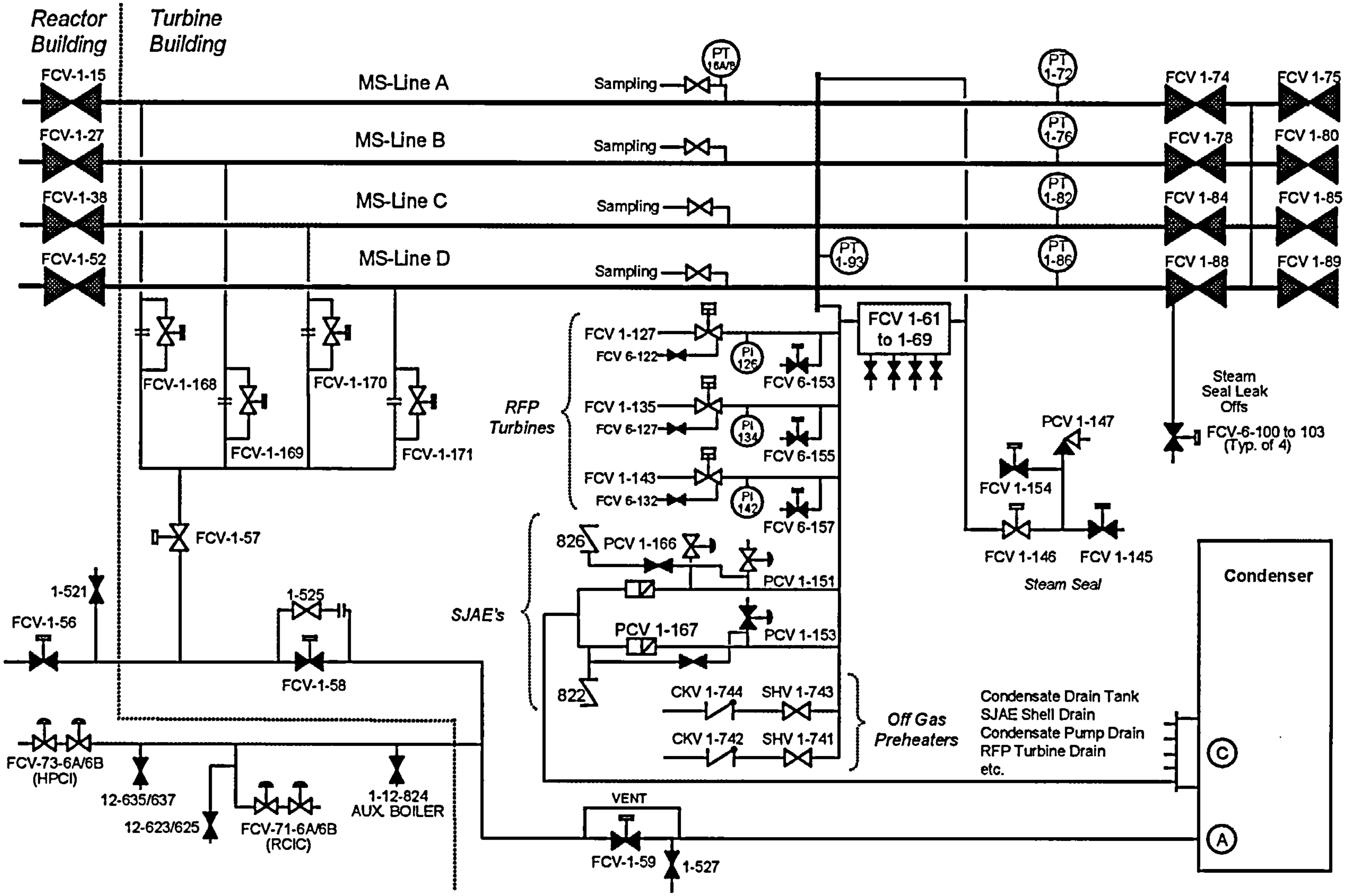
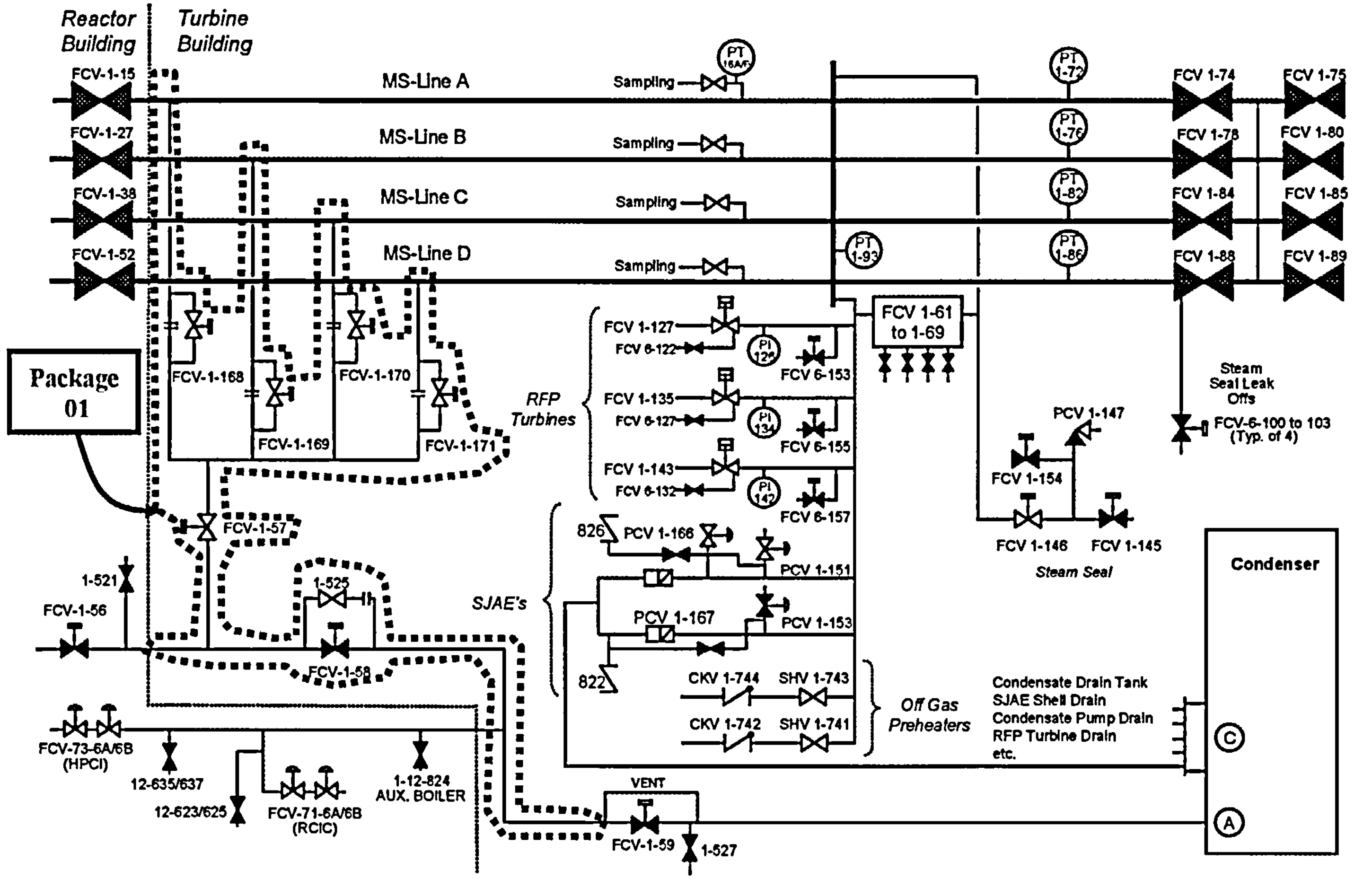
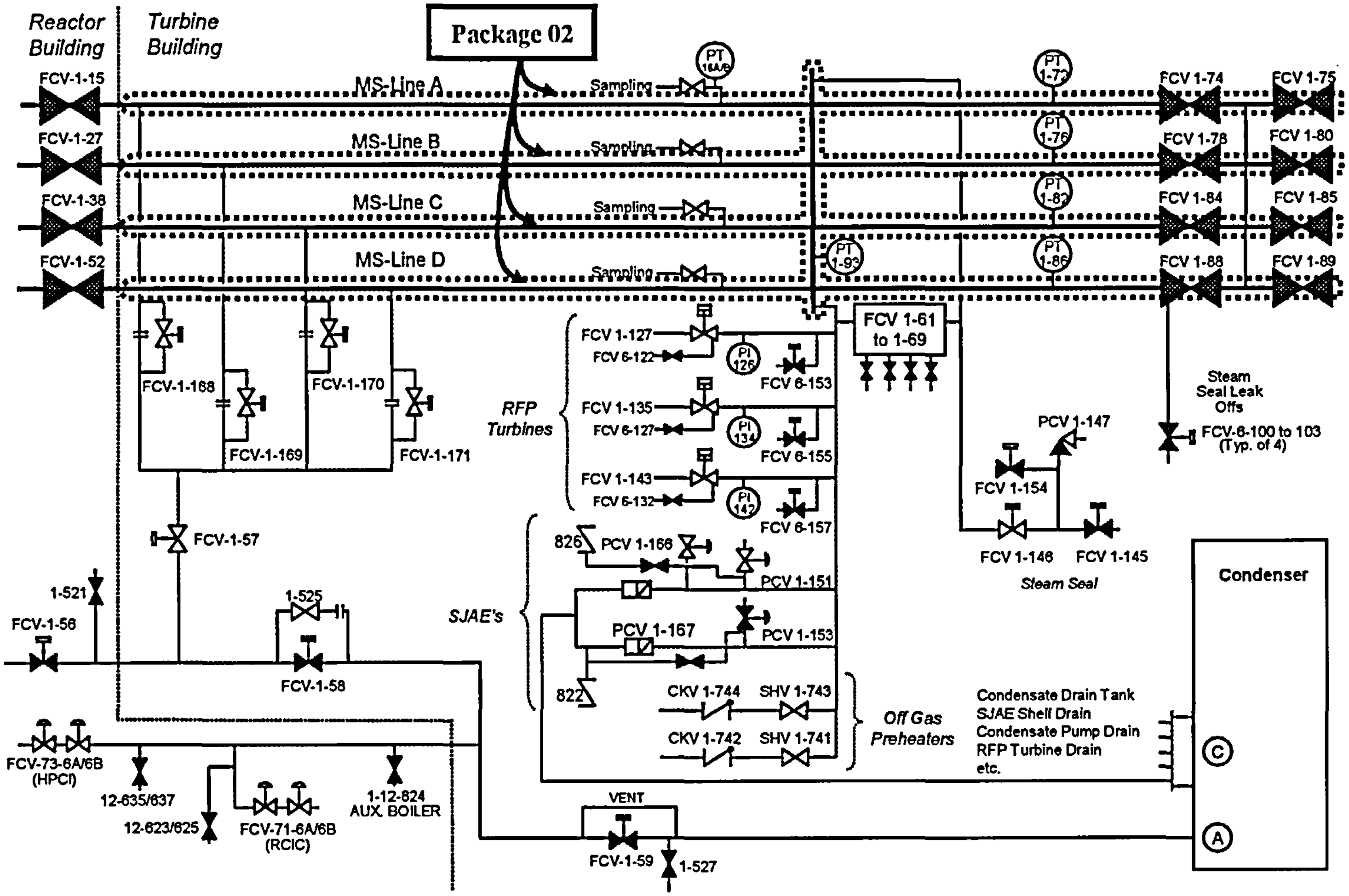


Figure 2-1: BFN Unit 1 MSIV Seismic Verification Boundary



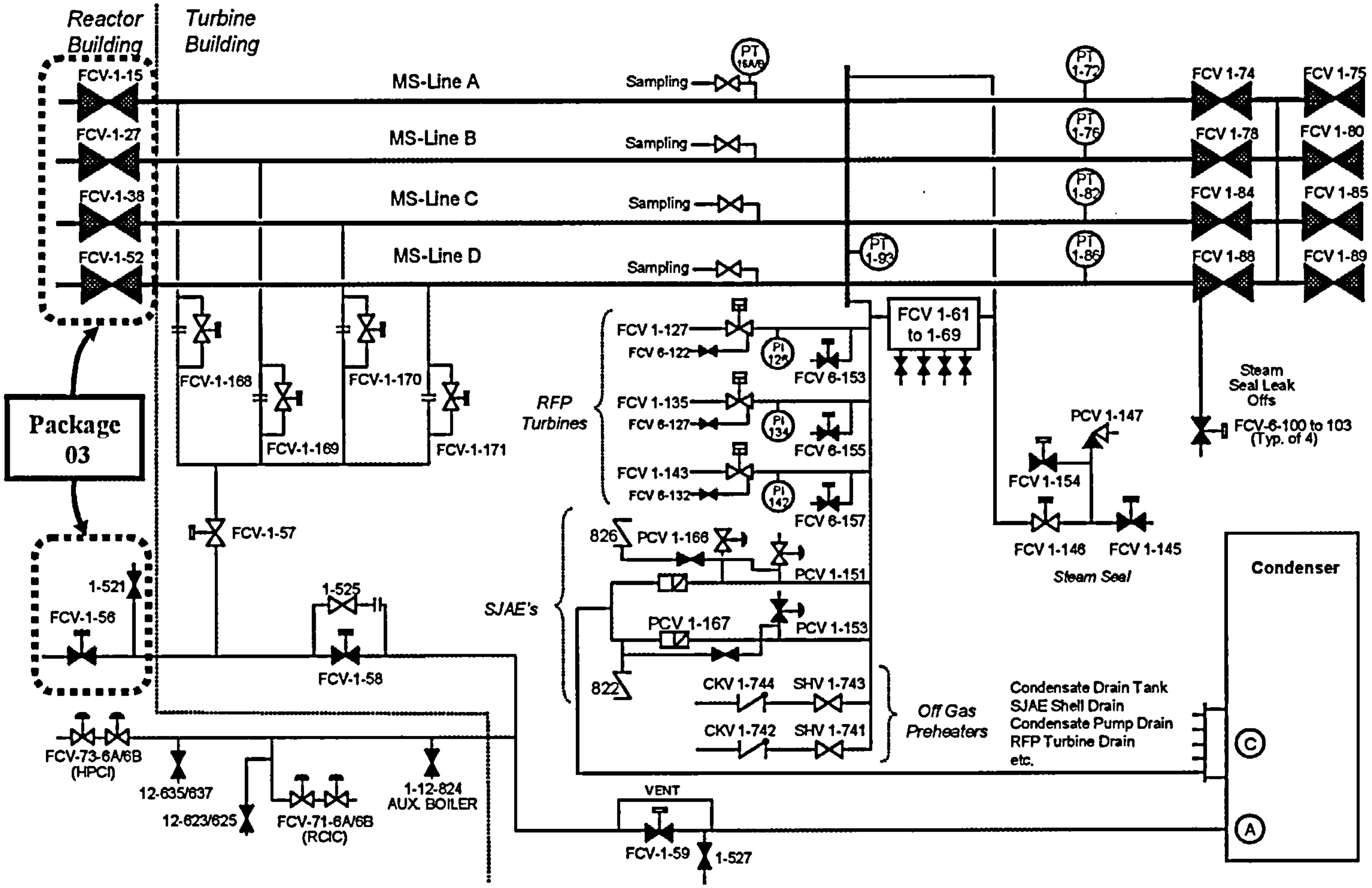
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-2: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 1



BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-3: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 2



BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-4: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 3

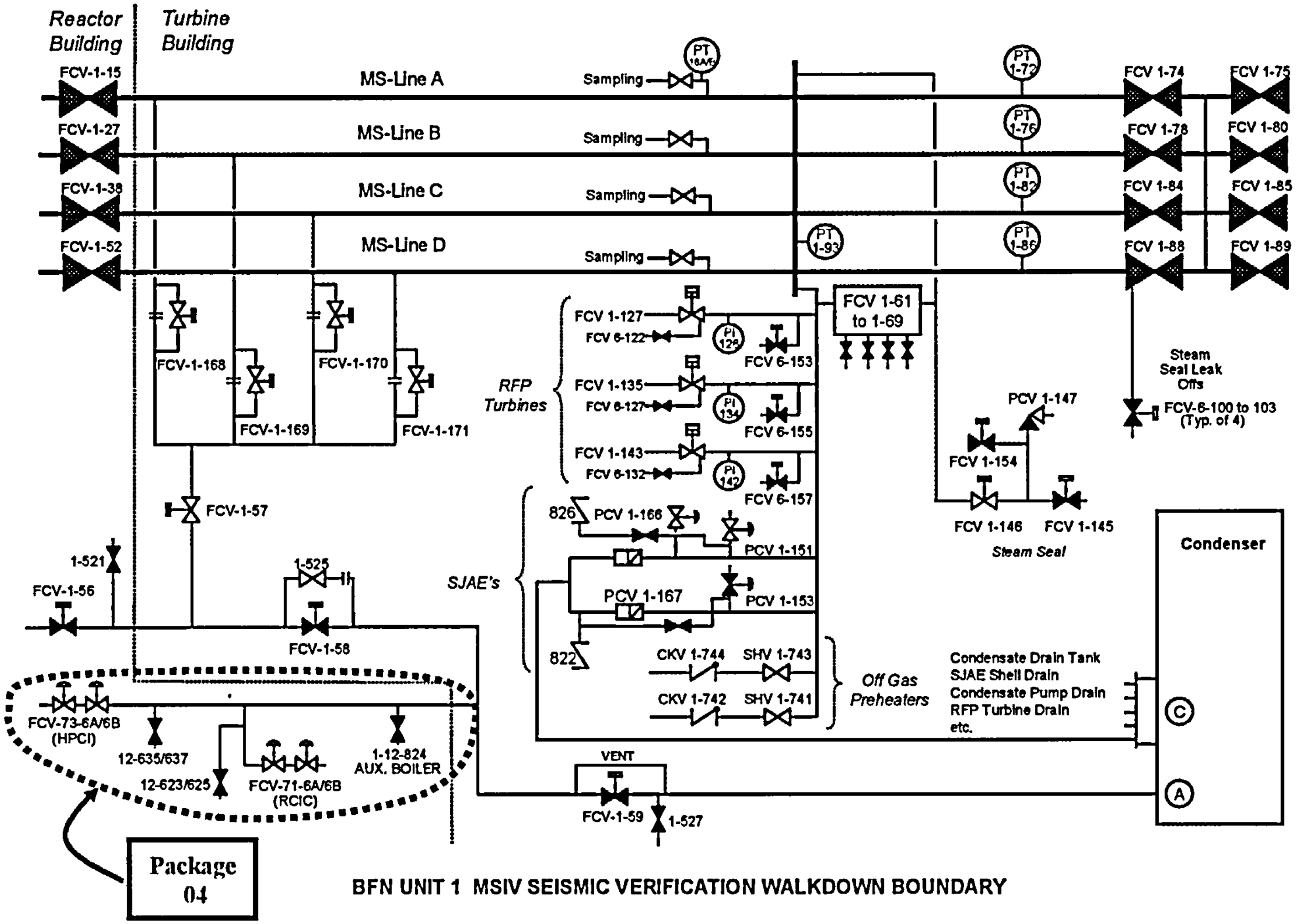
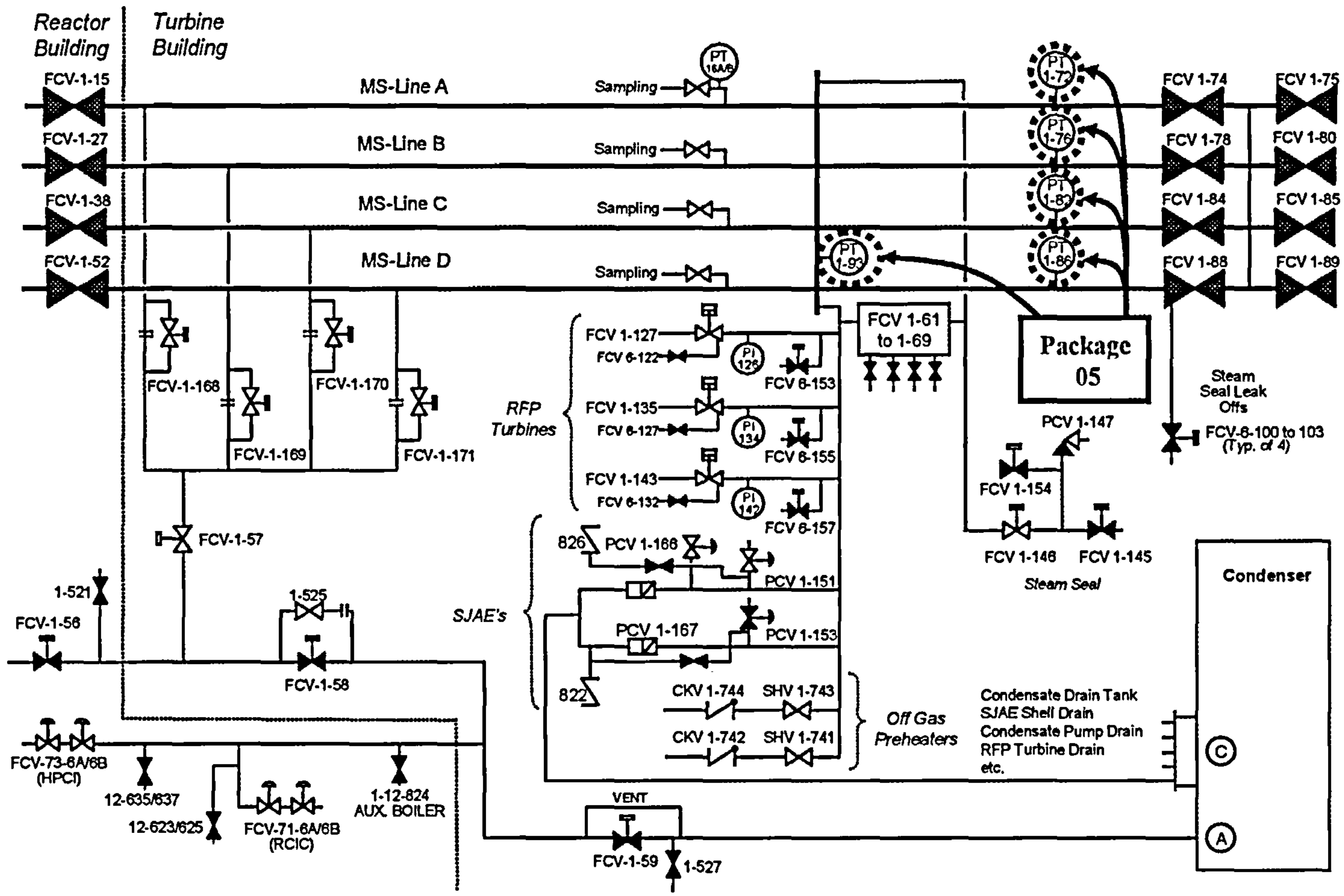
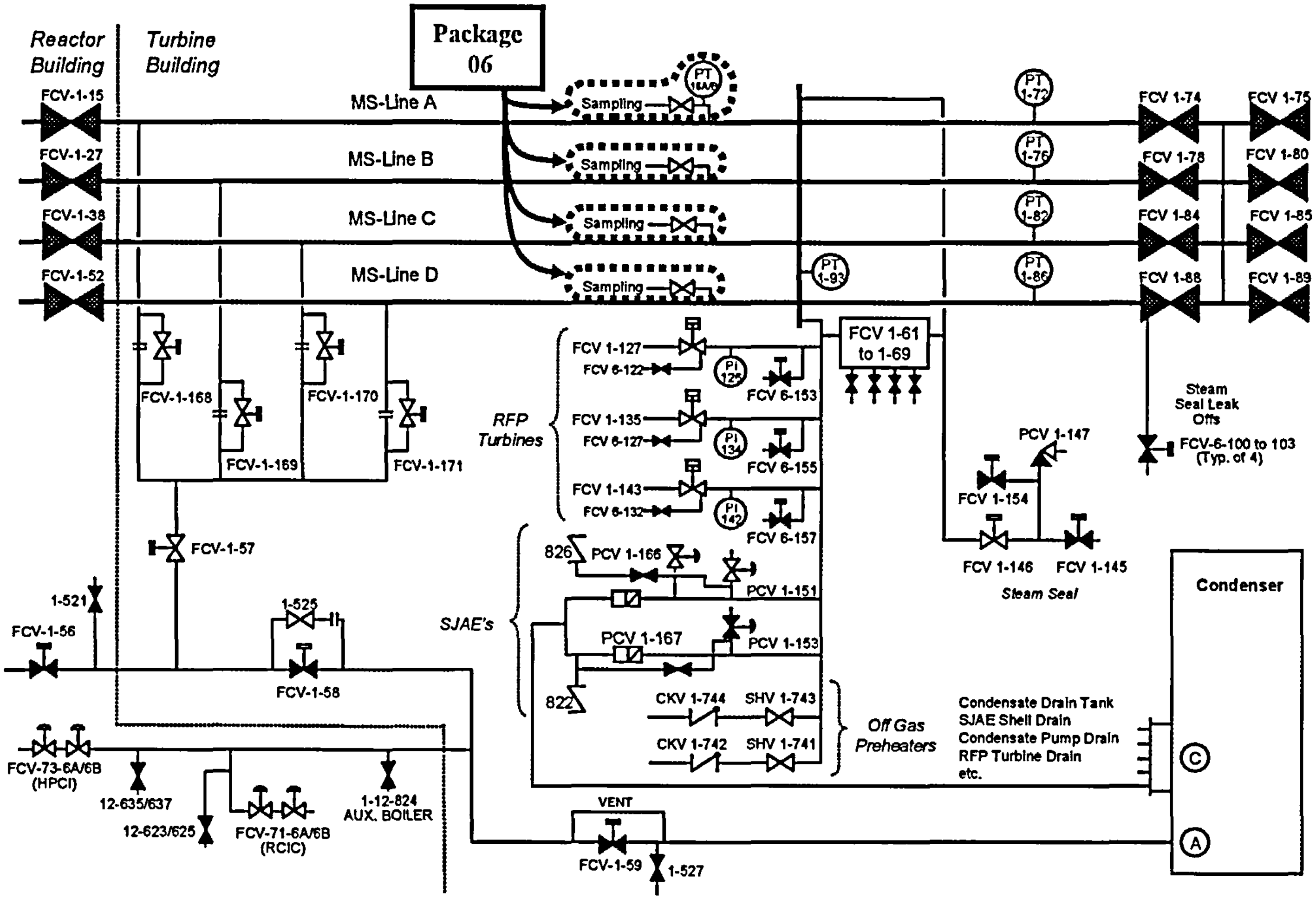


Figure 2-5: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 4



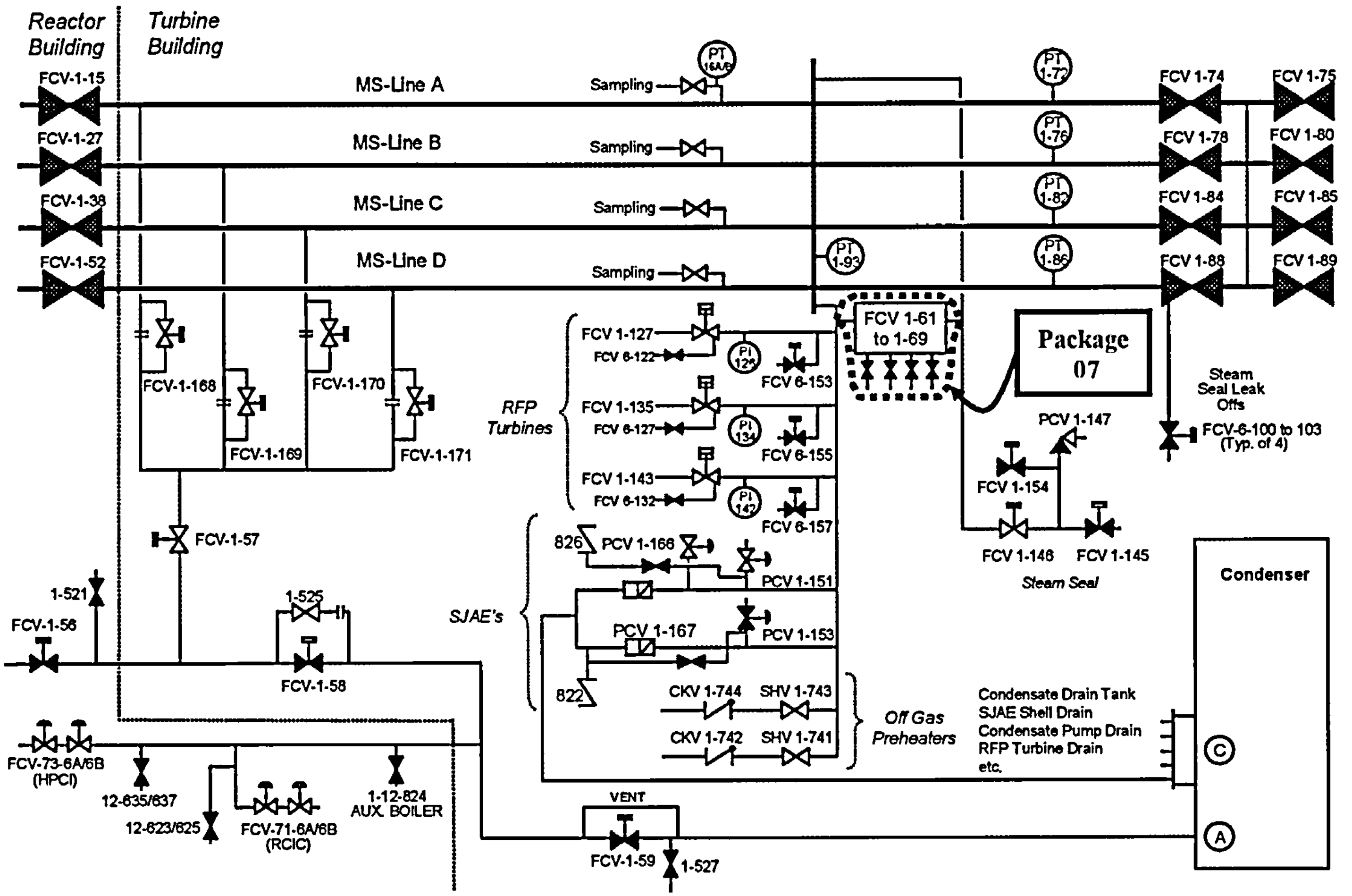
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-6: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 5



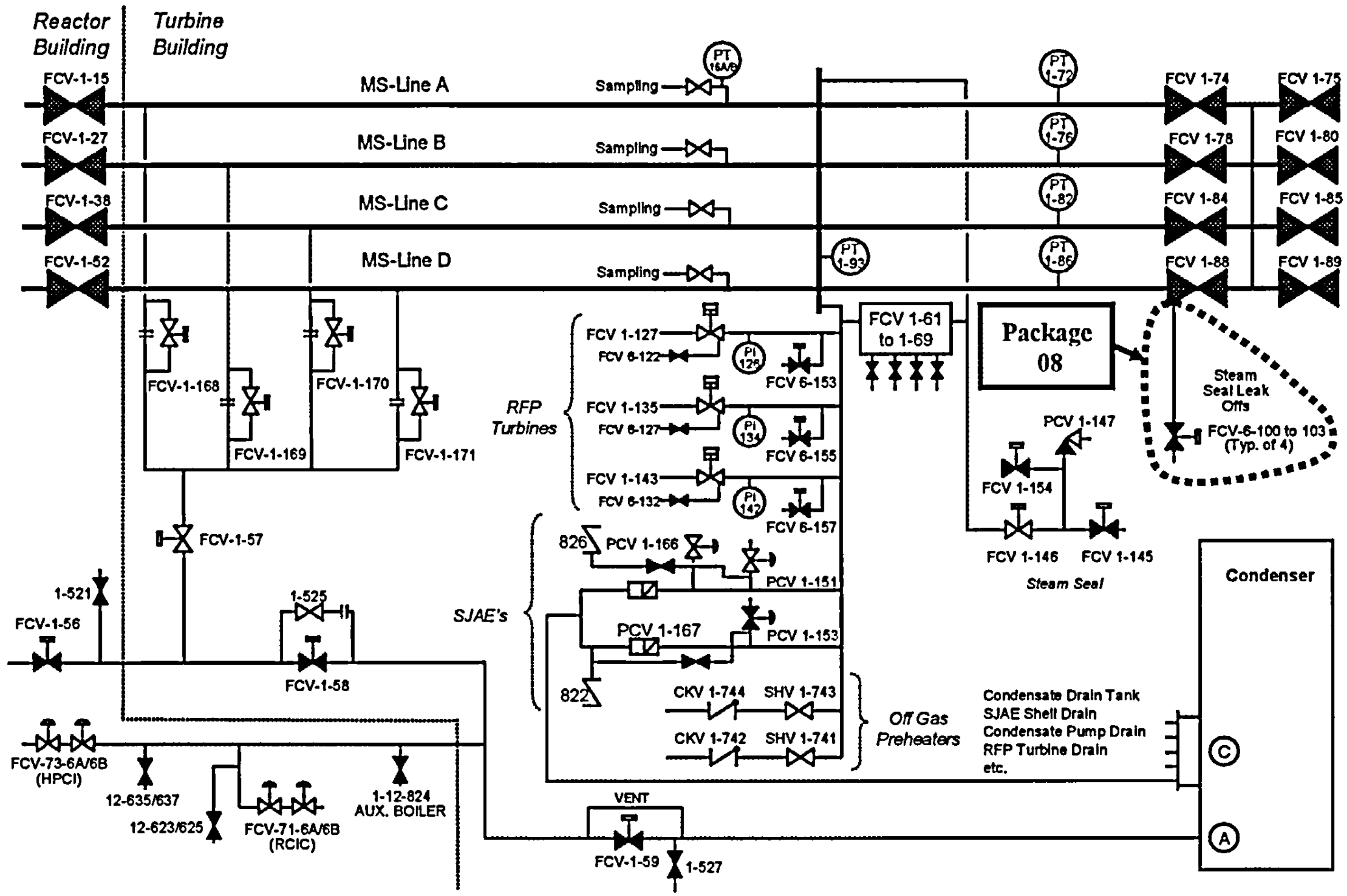
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-7: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 6



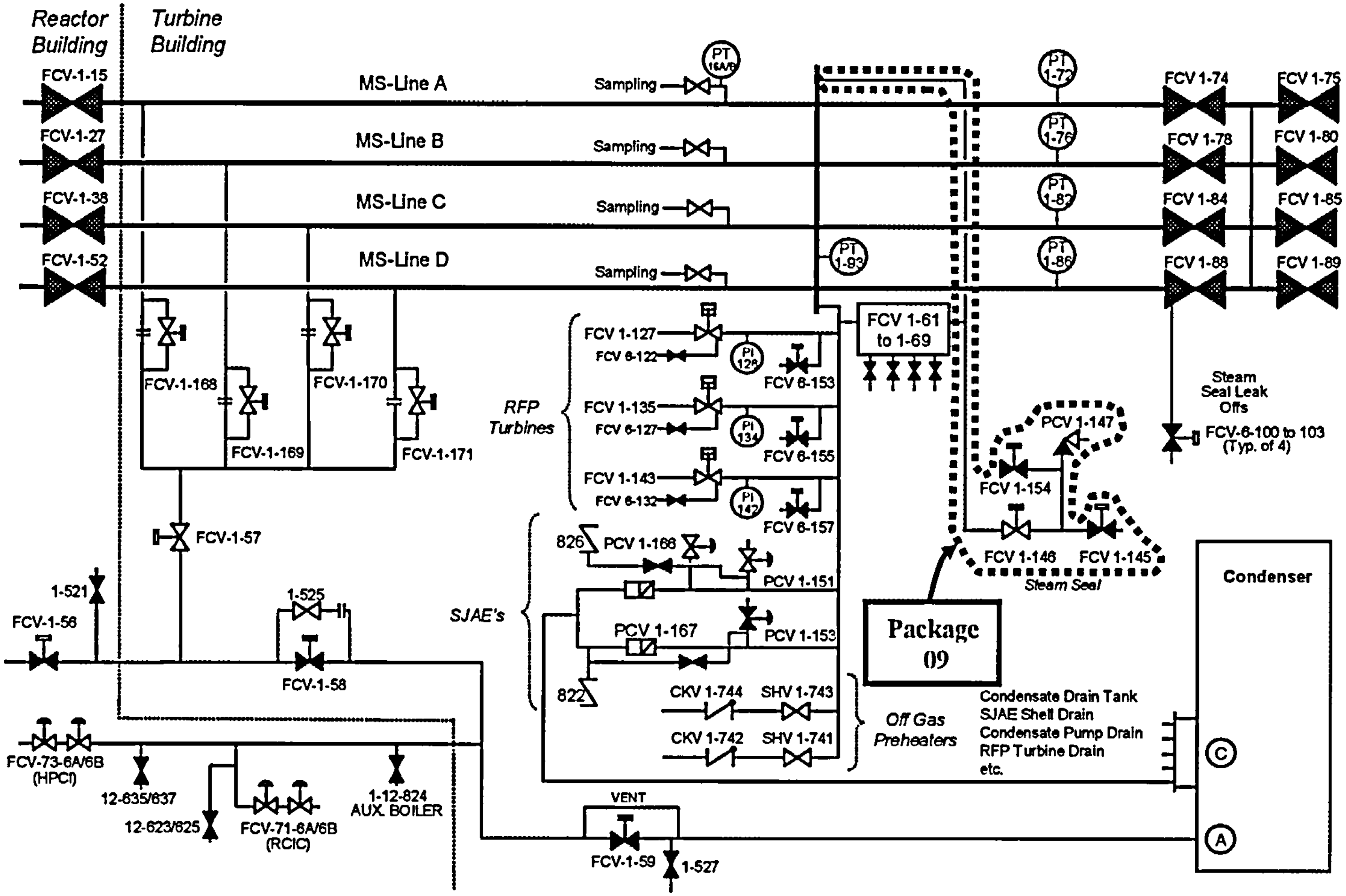
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-8: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 7



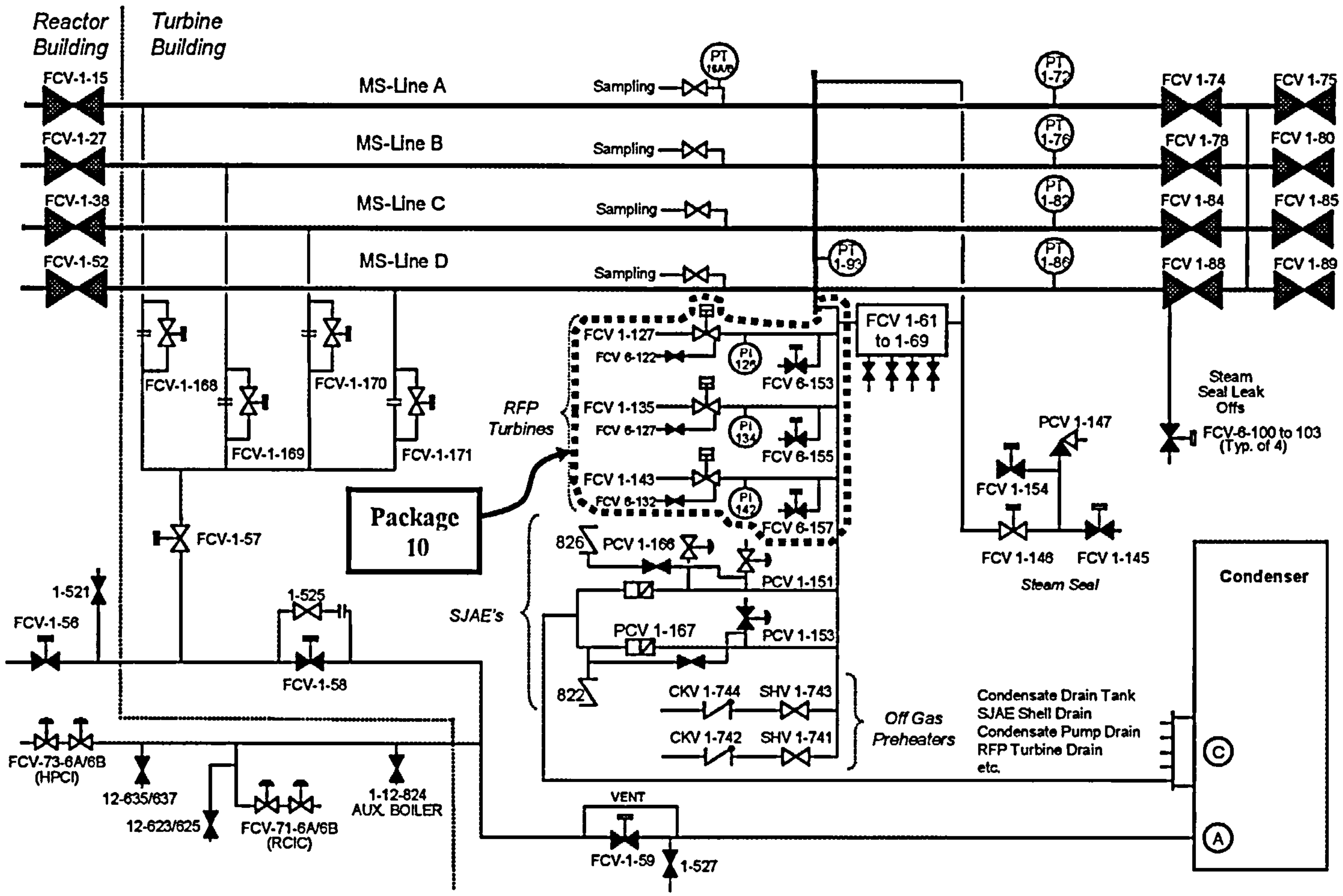
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-9: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems -- Package 8



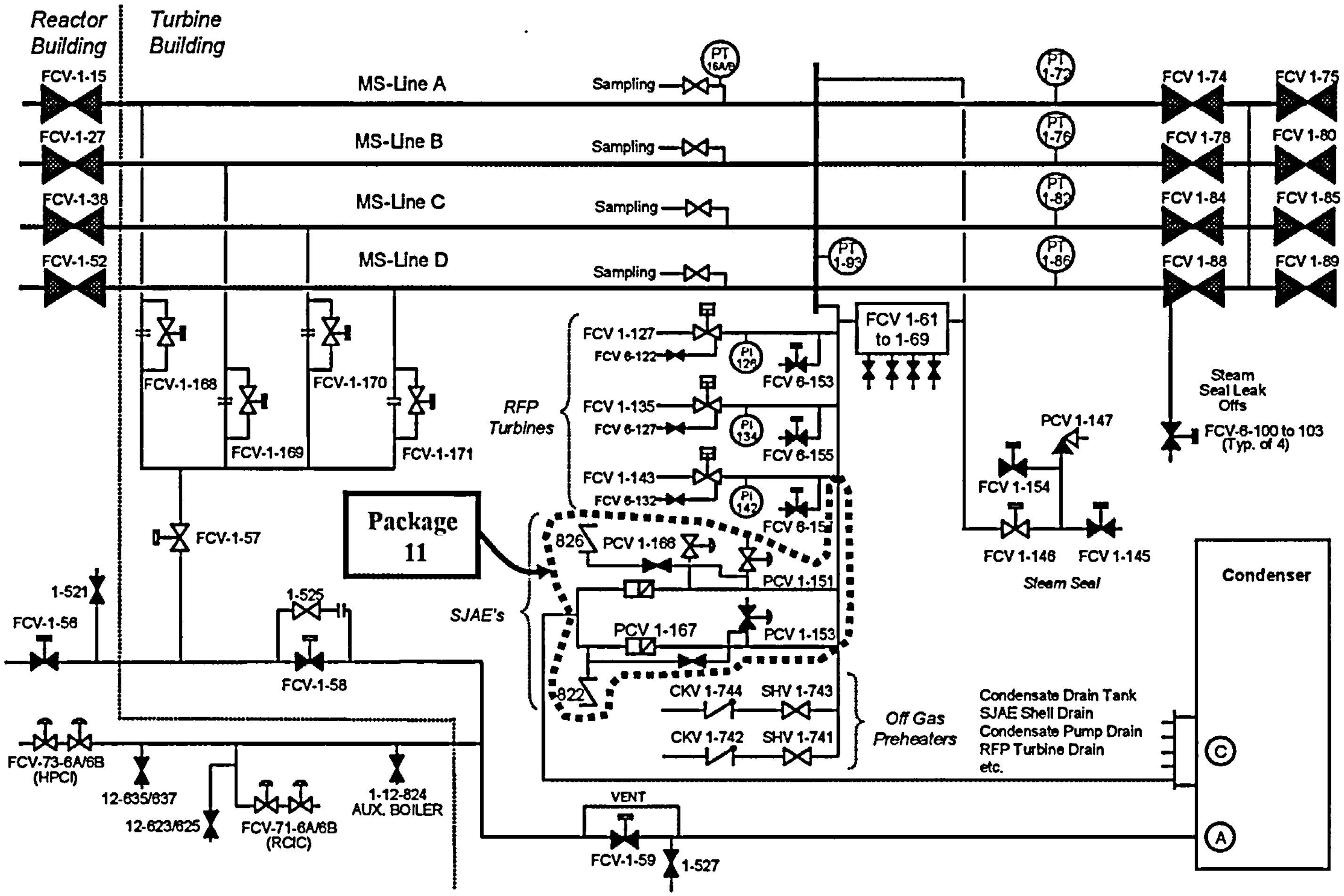
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-10: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 9



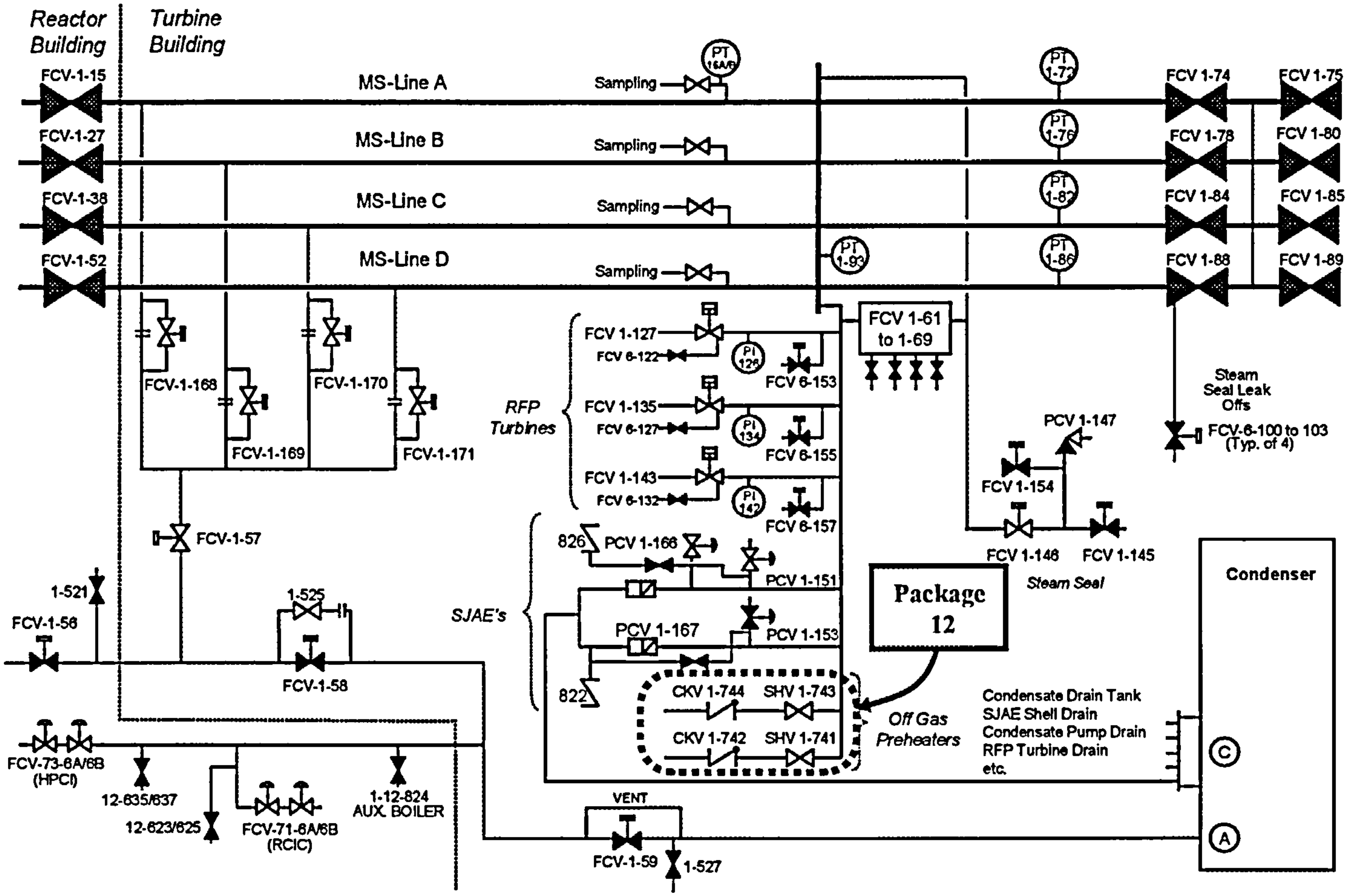
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-11: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 10



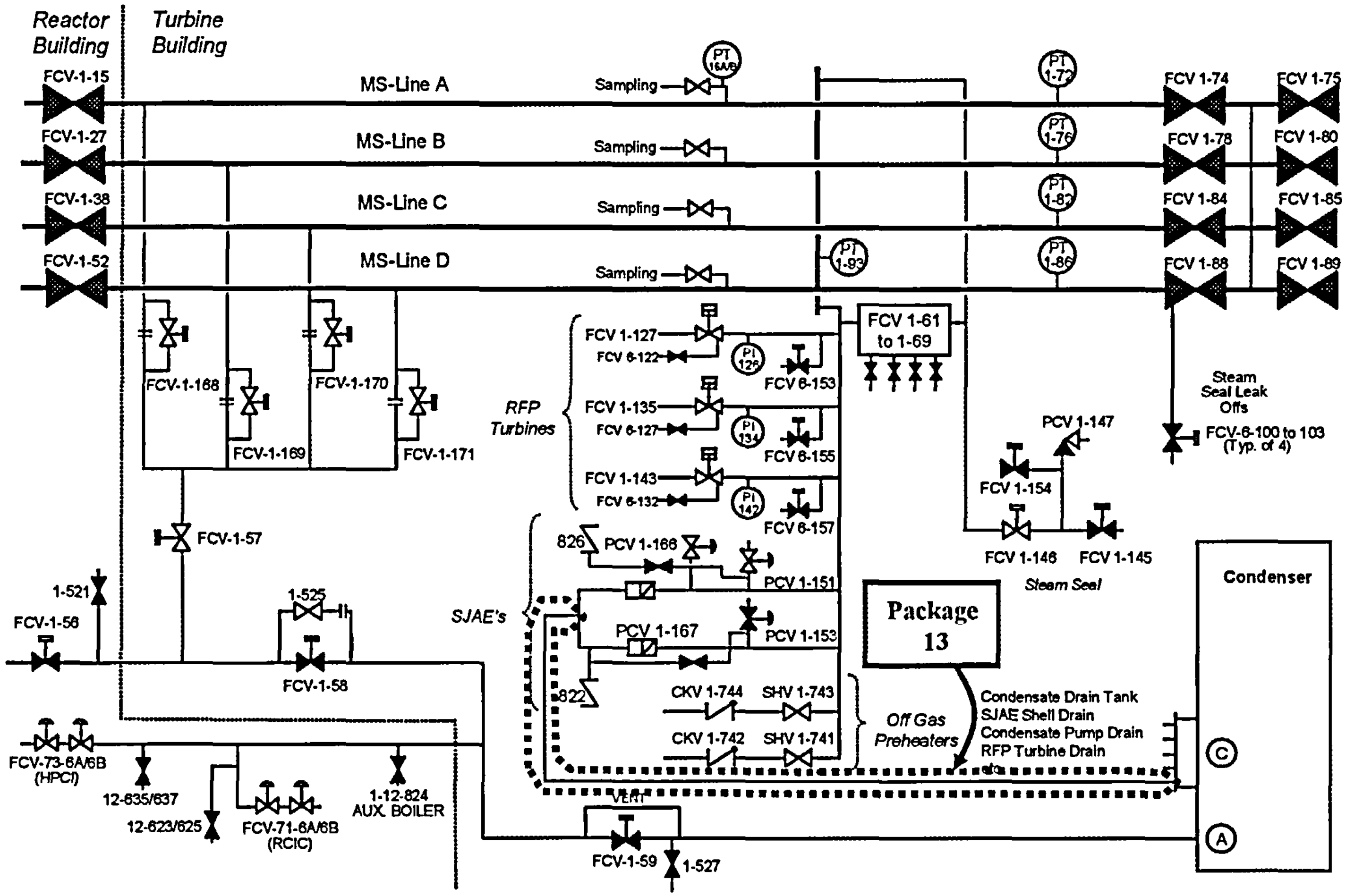
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-12: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 11



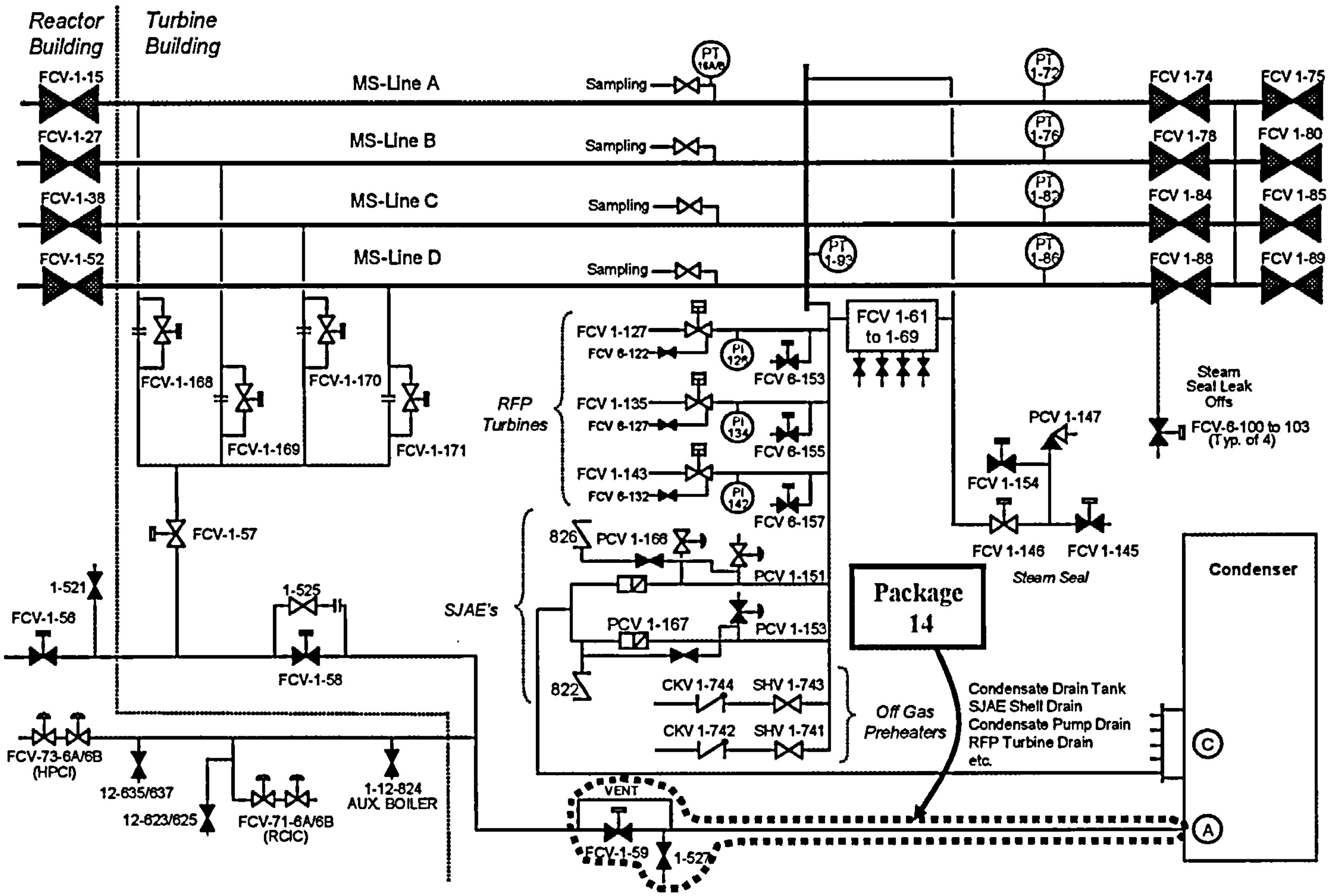
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-13: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 12



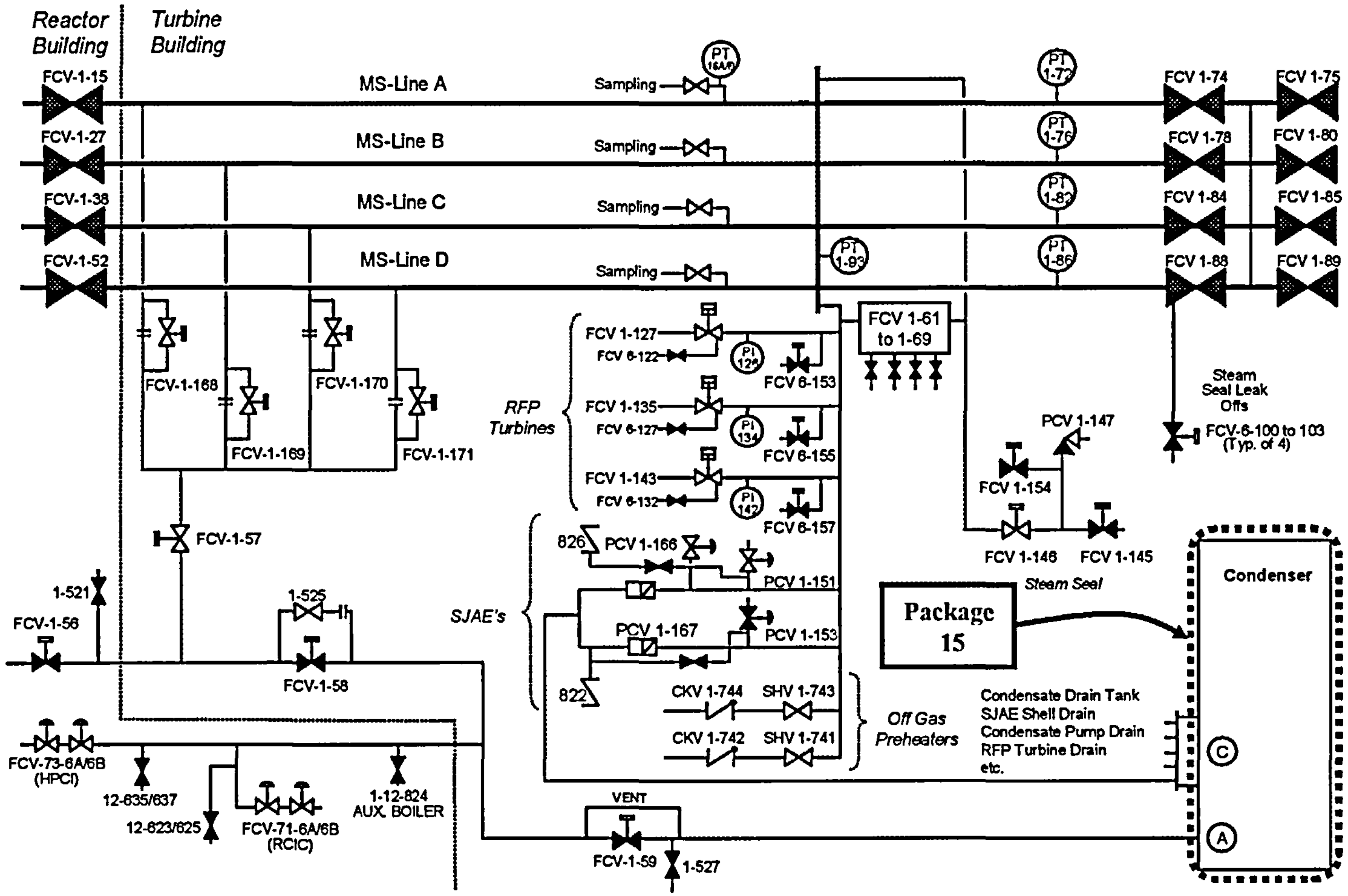
BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-14: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 13



BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-15: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 14



BFN UNIT 1 MSIV SEISMIC VERIFICATION WALKDOWN BOUNDARY

Figure 2-16: BFN Unit 1 MSIV Seismic Walkdown Scope by Subsystems – Package 15

3. SEISMIC VERIFICATION WALKDOWN

Very few components of nuclear plant systems are unique to nuclear facilities. Nuclear plant systems include piping, tubing, conduit, and many other items that are common components of conventional power plants and industrial facilities. Seismic experience data based methods have been developed which address the question of adequacy of seismic performance of equipment and commodities not designed, procured and installed to current nuclear seismic criteria (Reference 7-3). By reviewing the performance of facilities that contain equipment similar to that found in nuclear plants, conclusions can be drawn about the performance of nuclear plant equipment and associated components during and after earthquake events. Extensive work has been performed documenting the performance of power plant equipment and the common sources of seismic damage to equipment and piping (References 7-1, 7-4 and 7-5). These and other similar studies form the basis for the walkdown.

Equipment, piping and tubing systems in the seismic experience data base have performed very well in earthquakes, even though they were typically designed for deadweight and operating loads only, with little or no consideration for seismic loads (Reference 7-5). Earthquake experience data base methods provide the basis for review of the main steam piping and equipment.

The catalog of earthquake experience data on which the walkdown screening evaluation was based includes the El Centro Steam Plant, Valley Steam Plant, Glendale Power Plant, Burbank Power Plant, Humboldt Bay Plant, PALCO Co-generation Plant, Coolwater Plant, Ormond Beach Plant, Moss Landing Steam Plants and several other facilities affected by the 1987 Whittier earthquakes, and the 1992 Cape Mendocino (Humboldt Bay) and Landers-Big Bear earthquakes in California.

A comparison of the selected earthquake experience database site spectra with the Browns Ferry Design Basis Earthquake (DBE) ground spectrum, amplified by 1.6 to account for soil-founded structures such as the Turbine Building, is shown in Figure 3-1. The earthquake experience database plants have experienced strong motions substantially in excess of the soil-amplified Browns Ferry DBE (1.6 x DBE).

Further discussion on the appropriate seismic demand used for Turbine Building is presented in Section 5.1.1. Also, additional details regarding the earthquake experience database plants can be found in Appendix D, Section 4.1, of the BWROG Report GE NEDC-31858P (Reference 7-1).

3.1 Seismic Verification Review Guidelines

Various design attributes of the as-installed scope of equipment, piping, and tubing were reviewed and evaluated by the Seismic Walkdown Teams to insure that the BFN installations are representative of data base design practice and that components are free of known seismic vulnerabilities. Earthquake experience has identified conditions that have resulted in failure of piping and tubing systems and components. The conditions evaluated in this walkdown review included:

- Piping, Pipe Support and Equipment Design Attributes
- Seismic Anchor Motion Issues
- Seismic Interaction Issues (II/I & Proximity)
- Boundary Valve Design Attributes

3.2 Piping, Pipe Support and Equipment Attributes

Earthquake experience data base representation of the piping and tubing systems can be verified by the following design and installation attributes:

- Piping and tubing installations are in conformance with industry-standard practices (e.g., ANSI B31.1 spans for piping, standard industrial supports for piping and tubing).
- Piping or tubing system does not display known seismic vulnerabilities or exhibit seismically sensitive characteristics.

3.2.1 Piping and Pipe Support Design Attributes

The Seismic Walkdown Team reviewed the piping and tubing systems for conditions associated with past poor performance. Support failure in past earthquakes has rarely resulted in piping failure unless multiple supports over long runs fail. Support types which have demonstrated poor seismic performance include:

- One-way bracket or stanchion supports that could allow pipe to slide off the support and fall.
- Supports attached by beam clamps without restraining straps.
- Short threaded rods that are fixed against rotation may be vulnerable to low cycle fatigue.
- Other support integrity issues such as damaged, deteriorated or altered parts that could result in non-ductile behavior or significant weakness in the load path.

Rod hanger attachments that act as pinned members (e.g. clevis, eyes, etc.) and welded structural steel pipe supports have not been observed in seismic events to be vulnerable unless there are major design or construction flaws in their attachment to the supporting structure.

3.2.2 Vulnerable Piping Joints

Threaded piping connections, unsupported bellows or expansion joints and mechanical joints (e.g., mechanical type couplings, bell and spigot joints, etc.) have been observed in past earthquakes to be more vulnerable to seismic loads than welded piping joints. The Seismic Walkdown Team reviewed flexibly supported piping to identify segments which contain weak or brittle joints. Bolted flange connections have performed well and are not considered to be weak or vulnerable to seismic loads.

3.2.3 Other Potential Seismic Vulnerabilities

Other piping and tubing design attributes which are inconsistent with good design practice and which may contribute to poor seismic performance include:

- Piping with dead weight support spacing greatly in excess of ANSI/ ASME B31.1 suggested spans (Reference 7-16). Tubing with excessive sagging or support spans greatly in excess of 6'-0".

- Heavy in-line masses (e.g., valves, accumulators, filters, strainers).

- Piping inadequately restrained adjacent to expansion joints.

- Piping constructed of non-ductile materials such as cast iron or PVC.

- Non-standard fittings, such as mitered elbows or unreinforced branch connections, or unusual attachments that could cause excessive localized stresses.

- Presence of severe corrosion.

Walkdowns reviewed piping and supports to identify designs which contained any of these attributes.

3.3 Equipment Design Attributes

The equipment reviewed in the seismic verification walkdown includes the main condenser and equipment which acts as terminal anchor points for piping and tubing systems. Typical equipment components include heat exchangers, vessels and some measuring instrumentation such as transmitters, and gauges. In general the walkdown review included:

- Review the equipment for known failure modes and sources of seismic damage which may affect the seismic performance of the equipment and sub components.

- Check for unusual or non typical arrangements of the devices within the equipment or of items external to the equipment.

- Assess the anchorage and presence of an adequate load path. For instrumentation, this included evaluation of the instrument rack and/or other terminal equipment.

The details of the review varied according to the type of equipment and location within the plant. The extent of review and information gathering for evaluation of Seismic Verification Boundary components, equipment required for structural integrity, etc. were determined based on the judgment and experience of the Seismic Walkdown Team.

3.4 Anchorage Design Attributes

Anchorage of equipment and supports were reviewed for adequacy on a visual basis. Details of piping and equipment anchorage were documented in the walkdown notes only as required for review of outliers.

3.4.1 Expansion Anchor Bolt Review Guidelines

Visual review of expansion anchor bolts considered the following:

- The concrete appears sound with no significant cracks in the vicinity of the anchor bolt.
- Gaps between the equipment base and the concrete surface were negligible.
- The bolt spacing is greater than about 10 times the bolt diameter.
- The distance between the bolt and any free concrete surface is greater than approximately 10 times the bolt diameter.

3.4.2 Welded Anchorage Review Guidelines

Welded anchorages were reviewed for adequacy, good installation practices and workmanship on a visual basis. This visual review included:

- Review for weld burn-through on thin sections.
- Review of weld thickness, against the thickness of thinner part being connected.
- Review for plug welds subjected to tension loads.

3.5 Seismic Anchor Movement

The experience data base includes instances of seismic damage to piping, tubing and supports that were attributed to seismic anchor movement. Damage was the result of excessive movement of terminal end equipment, differential movement between supports in adjacent buildings, and excessive movements imposed on branch lines by flexible headers. As a result of these observed vulnerabilities, the following attributes were evaluated by the Seismic Walkdown Team during the piping walkdown:

- System configurations at building joints and between buildings were reviewed to insure adequate piping system flexibility to accommodate seismically-induced differential building movement.

- Fittings which can be adversely affected by seismically-induced differential movement (e.g., bellows, etc.) were evaluated for adequate flexibility. This included conditions with rigid connections at multiple structures.

- Piping attached to unanchored or poorly anchored equipment was considered an outlier. Stiff piping attached to flexible equipment was evaluated to verify that the piping will not act as an equipment anchorage.

- Conditions where stiffly supported branch lines were attached to flexibly-supported (e.g., rod-hung) mainlines or headers were identified as outliers. The Seismic Walkdown Team evaluated these configurations for potential damage due to seismically-induced differential movement.

3.6 Seismic Interaction Review (II/I and Proximity)

The seismic interaction review was a visual inspection of structures, piping, or equipment adjacent to the components under evaluation. The seismic interaction review identified seismically induced failures (II/I) and displacements of adjacent structures, piping, or equipment (proximity) that could adversely affect the required seismic performance of the system and components under consideration.

The Seismic Walkdown Team identified and evaluated all credible and significant interaction hazards in the immediate vicinity of the item being evaluated. Evaluation of interaction effects considered detrimental effects on the capability of equipment and systems to function, taking into account equipment attributes such as mass, size, support configuration, and material hardness in conjunction with the physical relationships of interacting equipment, systems, and structures. In the evaluation of proximity effects involving overhead or adjacent equipment failure and interactions, the effects of intervening structures and equipment which would preclude impact were also considered.

Unusual circumstances and details have led to damaging interaction of plant features during past earthquakes. In the interaction review, the Seismic Walkdown Team looked for unusual impact situations, and lack of proper anchorage or bracing of adjacent equipment. All credible interactions that could affect the required performance of the piping, tubing and equipment reviewed were identified and documented, as appropriate.

3.7 Seismic Verification Boundary Valves

Screening guidelines are provided for valves which are either relied upon to establish, or are within the Seismic Verification Boundary. The guidelines are consistent with the SQUG Generic Implementation Procedure (GIP, Reference 7-3) and include provisions for air-operated diaphragm valves, spring-operated pressure relief valves and piston-operated valves of light-weight construction. Screening guidelines for motor-operated valves and substantial piston-operated valves are also provided. Evaluation of valves included review of power and control utilities to insure adequate slack is provided to accommodate anticipated seismic motions. Supports located on the valve operator were reviewed to insure that they were accompanied by supports on the valve body or piping adjacent to the valve body. Reviews were also performed to insure that the valve body and operator were supported by a common structure to prevent differential displacement. Piping or tubing less than 1-inch in diameter with in-line eccentric masses such as motor or air operated valves were checked for support at or near the valve.

3.8 Walkdown

BFN-1 MSIV seismic ruggedness verification boundary was divided into 15 subsystems or portions for walkdown purposes (see Section 2.2 above). Seismic verification walkdown of the main steam lines, various drain paths, and associated components and appendages within the Seismic Verification Boundary were conducted as part of BFN Unit 1 restart project, and were performed by Seismic Walkdown Teams consisted of Messrs. John O. Dizon, Stephen J. Eder, Robert D. Hookway and Michael W. Whited of ***FACILITY RISK CONSULTANTS, Inc.*** All of the MSIV Seismic Verification Walkdown Team members are degreed engineers; each has over ten to twenty years of experience in structural or mechanical engineering and/or earthquake engineering application to nuclear power plants, and is familiar with the earthquake experience methodology.

The above engineers have performed complete MSIV Seismic Verification Walkdowns in accordance with the recommendations of the BWROG Report GE NEDC-31858P at several other plants, including BFN Units 2 and 3.

3.9 Documentation

The seismic verification walkdowns and evaluations were performed in accordance with applicable Walkdown Instruction WI-BFN-0-CEB-07 (References 7-6), and following the guidelines contained in the BWROG Report GE NEDC-31858P (References 7-1). Screening evaluation guidelines were based on TVA Design Criteria BFN-50-C-7306 (References 7-7) for Class II piping and components. Walkdown evaluations utilized existing plant documentation, as available, including:

- Systems flow diagrams or P&ID's identifying piping and equipment within the verification review boundaries;

- Piping isometric drawings;

- Piping support sketches and piping layout drawings, as needed;

- In-plant screening tools such as piping deflection charts, pipe flexibility charts, standard support hardware capacities, anchorage capacities, and others (Reference 7-8), as applicable;

- Walkdown and evaluation results from previous MSIV seismic ruggedness programs conducted for BFN Units 2 and 3 (References 7-9 and 7-10).

The walkdown review of piping and supports was primarily visual for qualitative attributes of the systems. Only physical system attributes which could be visually verified with available access, and without system disassembly were reviewed. Where indicated, additional details of the system design, installation and construction were collected and are documented on piping isometrics and in walkdown field notes, as appropriate.

Conditions which did not conform to the walkdown screening guidelines or which were judged by the Seismic Walkdown Team to require further review were identified and documented as potential outliers in the Potential Outlier Sheet (POS). Some photographs were taken for informational use in subsequent evaluation phases of the project.

Results of the walkdown for each of the 15 portions or subsystems within the BFN-1 MSIV seismic verification boundary are documented in the respective walkdown data package (WDP). A listing of the walkdown data packages with unique identifier number can be found in Appendix A of this report.

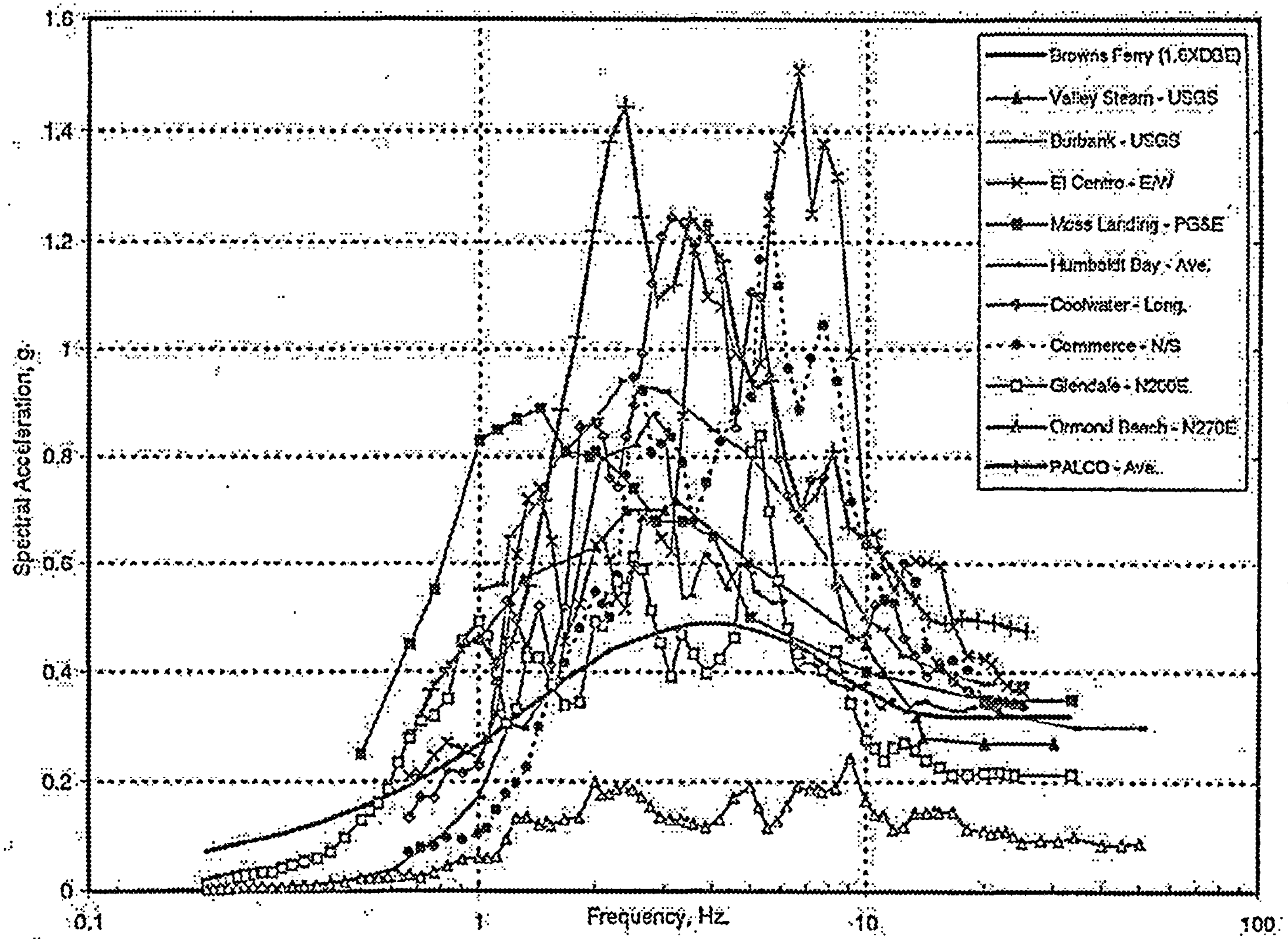


Figure 3-1 Comparisons of Database Site Spectra to Browns Ferry Design Basis Ground Spectrum

4. WALKDOWN OPEN ITEMS

Screening walkdown evaluations for the 15 subsystems or portions included in the BFN-1 MSIV Seismic Verification Boundary were performed in accordance with Walkdown Instruction WI-BFN-0-CEB-07 (Reference 7-6). Screening evaluations focused on certain key attributes of the associated piping and components, as discussed earlier in Chapter 3, to ensure pressure boundary integrity. Available screening tools (see Reference 7-8) such as seismic deflection estimates and charts for various plant features, pipe flexibility and seismic anchor movement evaluation charts, support and anchorage capacity screening charts, and others, that are developed based on the TVA Design Criteria BFN-50-C-7306 (Reference 7-7) were utilized during the walkdowns.

Results of in-plant screening walkdown evaluations for the 15 subsystems included in the MSIV seismic boundary are documented in details in the respective walkdown data packages (WDP's). Conditions which did not conform to the walkdown screening guidelines or which were judged by the Seismic Walkdown Team to require further review were identified and documented as potential outliers in the Potential Outlier Sheet (POS) which are also included in the WDP's. A listing of the walkdown data packages with unique identifier number can be found in Appendix A of this report.

The following sections provide a brief discussion of the walkdown evaluations and identified open items or potential outliers associated with each of the 15 subsystems.

4.1 Main Steam Drain Line – Turbine Bldg. Main Steam Tunnel (Pkg. 01)

The Main Steam Drain Line, which is the primary MSIV leakage drain path, originates in the Reactor Building MSIV Vault (Package 03, Section 4.3) and continues through the Turbine Building Main Steam Tunnel (Package 01, this section). The drain then enters a guard pipe which is embedded below the slab and crosses below the gap between the Steam Tunnel and the Turbine Pedestal Structure, continues into the Turbine Structure and terminates at the main condenser (Package 14, Section 4.14). The Main Steam Drain Line also has a 4-inch diameter vent line that begins in the Turbine Building Main Steam Tunnel, rises up to Elevation 593' and continues through the Turbine Structure to the south end of Condenser A where it runs down to rejoin the Main Steam Drain Line before it enters the condenser (also included in Package 14). The lines are shown on Flow Diagram 1-47E801-1.

The portion of piping in this package included the Main Steam drain lines in the Turbine Building Steam Tunnel on Elevation 565', i.e., from the blowout panels at column line N to just north of column line K. Several potential outliers were identified, as listed below:

Subsystem Description	POS #	Description of Potential Outlier Condition
Main Steam Drain Taps	1-1	Flexibility concerns associated with the MS lines and Drains.
MOV 's 1-FCV-1-57 & 1-FCV-1-58	1-2	Excessive mass and extended valve operators.
Main Steam and HPCI Drains	1-3	Flexibility concerns associated with the MS and HPCI Drains.
Main Steam Bypass Drain	1-4	Proximity interaction concerns between bypass line and adjacent 2-way support.

4.2 Main Steam – Turbine Building (Package 02)

This package includes the four Main Steam Lines (A, B, C & D) inside the Turbine Building that penetrate from the blowout panels at column line N and terminate at the Main Steam Stop Valves 1-FCV-1-74, -78, -84 and -88, which are considered as seismic boundary valves. These lines are shown on Flow Diagrams 1-47E801-1 and 1-47E801-2. The Main Steam lines come from the outboard MSIV's and are anchored downstream inside the Reactor Building MSIV vault. These lines are vertically supported by variable springs along their length until the Stop Valves, which are supported by vertical struts.

Two (2) potential outliers were identified and are listed below:

Subsystem Description	POS #	Description of Potential Outlier Condition
MS Balancing Header	2-1	Upper pipe clamp nut is missing.
MS Turbine Stop/Control Valves	2-2	MS Stop/Control Valves are not covered in the valve screening criteria, thus, seismic adequacy of these valves needs to be verified.

4.3 Main Steam Drain Line – Reactor Building Main Steam Vault (Pkg. 03)

Included in this package are the four Main Steam Lines which originate at the containment penetrations and run to the outboard MSIV's 1-FCV-1-15, -27, -38 and -52, which are considered as seismic boundary valves. The Main Steam Lines then run to anchors (one for each line) embedded in a reinforced concrete wall. The lines continue through the MSIV Vault and enter the Turbine Building through the blowout panel wall. The Main Steam Piping is shown on Flow Diagram 1-47E801-1. The Main Steam Lines in the MSIV Vault are seismically analyzed from the containment penetrations to the anchors in the concrete wall, which are also the only supports for the lines. The Main Steam lines beyond the anchors were found to be seismically adequate. The only potential outlier identified during the walkdown relates to the seismic adequacy of the MSIV's (not bounded by seismic experience data).

Also included in this package is a single 3-inch diameter steam drain that runs from its containment penetration through an anchor and to the normally closed outboard containment isolation valve for Primary Containment steam drains, 1-FCV-1-56, which is a seismic boundary valve. The piping continues through the Reactor Building MSIV Vault and exits to the Turbine Building through the blowout panel wall at the north end of the Vault (Package 04, Section 4.4). This drain line is shown on Flow Diagram 1-47E801-1. This drain line was found to be seismically adequate.

In the Reactor Building MSIV Vault, the 3-inch diameter drain line is primarily supported on stanchion supports that provide only vertical restraint and are detailed to accommodate thermal growth of the piping. Piping and supports in this area are considered to be acceptable. Valve 1-FCV-1-56 was identified as a potential outlier due to its extended valve operator not meeting the screening criteria for operator height.

Subsystem Description	POS #	Description of Potential Outlier Condition
MSIV 's 1-FCV-1-15, -27, -38 & -52	3-1	MSIV 's are not covered in the valve screening criteria, thus, seismic adequacy of these valves needs to be verified.
MOV 1-FCV-1-56	3-2	Excessive mass and extended valve operators.

4.4 HPCI/RCIC/Aux. Boiler Drains (Package 04)

HPCI and RCIC Steam Drains route steam leakage from the HPCI and RCIC Steam Supply Lines to the Main Steam Drain Line in the Turbine Building Steam Tunnel. These line segments are shown on Flow Diagrams 1-47E812-1, 1-47E813-1, and 1-47E801-1. The RCIC Steam Drain Line originates in the Northwest corner room of the Reactor Building (Boundary valve 1-FCV-071-06B) and is routed over the top of the Torus to its connection into the HPCI Steam Drain Line just below the penetration through the floor into the MSIV Vault. The HPCI Steam Drain Line starts in the HPCI room (Boundary valve 1-FCV-073-06B) and is routed above the Torus to the point where it penetrates the floor into the MSIV Vault. Both drains have branches near their points of origin that lead from Auxiliary Boiler Drain piping. The Auxiliary Boiler piping is shown on Flow Diagrams 0-47E815-1 and 1-47E815-3. The normally closed manual isolation valves (1-12-824, 1-12-635/637 and 1-12-623/625) that serve as the seismic verification boundary for the Auxiliary Boiler drains are located in the HPCI room and the Northwest corner room of the Reactor Building. These drain lines were found to be generally well supported in these areas.

From the penetration above the Torus through the floor of the Reactor Building MSIV Vault, the HPCI piping is routed through the MSIV Vault and into the Turbine Building Steam Tunnel through the blowout panels. The HPCI Steam Drain continues alongside the Main Steam Drain to the point where it connects with the Main Steam Drain between valves 1-FCV-1-58 and -59. The HPCI piping is supported throughout the MSIV Vault and the Turbine Building Steam Tunnel on floor-mounted, custom-fabricated supports which act as horizontal and vertical rigid restraints and are detailed to accommodate thermal growth of the piping.

Several potential outliers were identified on these drain lines. Conditions include pipe overspan, support deficiencies, proximity interactions and piping flexibility issues, as listed below:

Subsystem Description	POS #	Description of Potential Outlier Condition
RCIC Drain	4-1	Excessive pipe overspan condition.
RCIC Drain	4-2	Pipe support with questionable capacity.
Aux. Steam Boiler Drain	4-3	Missing rod hanger hardware (eye nut).

HPCI Drain	4-4	Multiple pipe overspan conditions on the 1"ø drain line.
HPCI Line	4-5	Multiple pipe overspan conditions on the 2"ø line.
HPCI Line and Valve 1-73-222	4-6	Proximity interaction of the 2"ø HPCI line and the valve.
HPCI Drain	4-7	Several floor-mounted supports were noted to be damaged (potentially due to thermal effects). Also, anchor bolt spacing violations on these typical supports.
HPCI Drain	4-8	Flexibility concern on 2"ø HPCI line due to differential displacement between RB and TB

4.5 Main Steam Pressure Transmitters PT 1-72, 76, 82, 86 & 93 (Pkg. 05)

The Main Steam Pressure Transmitters which tap off from the four Main Steam Lines upstream from the Main Steam Manifold (Balancing Header) and downstream from the Stop Valves are shown on Flow Diagram 1-47E801-2. The instrument lines to the pressure transmitters are routed on 1/2-inch diameter stainless steel piping from the pipe taps off the Main Steam Lines and the Manifold passing under the steel grating at Elevation 606'-3" in the Turbine Building to a penetration through the south wall of the area. The piping then runs to Instrument Racks 1-25-112 and 1-25-113C located on the south corridor of the Turbine Building at Elevation 586'.

Support configurations underneath the steel grating are generally rigid, consisting of strut channel welded to the underside of the grating beams. As such, several potential outliers were identified on these lines, mainly due to the limited flexibility in the piping and support configuration to accommodate the Main Steam header movements, as evidenced from the loose or missing clamps on the existing strut channel supports, broken overhead welds, and bent pipes noted during the walkdown. The identified potential outliers are tabulated below.

Subsystem Description	POS #	Description of Potential Outlier Condition
SS Piping to 1-PT-1-72 & -76	5-1	Overspan conditions for both lines. Also, flexibility issue exists on piping to 1-PT-1-72.
SS Piping to 1-PT-1-82	5-2	Overspan conditions due to broken weld on the support and missing clamps.
SS Piping to 1-PT-1-86	5-3	Overspan conditions due to broken weld on the support and loose clamp.
SS Piping to 1-PT-1-86	5-4	Flexibility concern on ½"ø piping.

SS Piping to PT 1-93	5-5	Overspan condition due to broken support.
SS Piping to PT 1-93	5-6	Flexibility concern on ½"ø piping along the wall.

4.6 Main Steam Sample Lines to Sampling Station (Package 06)

The Main Steam Sample Lines tap off from each of the four Main Steam Lines upstream of the Main Steam Stop Valves. The lines are shown on Flow Diagram 1-47E801-1 and Mechanical Control Diagram 1-47E610-43-1. The sample lines begin as a 1-inch diameter piping and transition to one-half-inch diameter tubing. The tubing is routed through the Turbine Building and exits the area through the south wall to Instrument Rack 1-25-149 in the Turbine Building south corridor, Elevation 586'. On the rack, the four tubes connect into a single tube with three branches. Seismic verification boundary valves 1-43-631 and 1-43-631A terminate two of the branches on the rack. One one-quarter-inch diameter tube runs from the rack through the constant temperature bath and on to the grab sample station and normally closed valve 1-43-632. The tubing is typically supported with tubing clamps attached to strut channels mounted to the wall.

Also included in this package are the Pressure Sensing lines for PT-16A and B which tap off from the 1-inch diameter piping taps for the Steam Line A sample lines. This piping transitions to tubing and runs west into the Turbine Lube Oil Pump area. The transmitters are attached to wall-mounted steel plate bracket supports and are enclosed by a mesh cage. The tubing lines are adequately supported using U-bolts and fabricated steel angle brackets mounted to the wall with expansion anchors.

Four (4) potential outliers were identified which include unanchored terminal equipment, broken support potentially due to inadequate pipe flexibility, and support hardware deficiencies. These potential outliers are tabulated below.

Subsystem Description	POS #	Description of Potential Outlier Condition
Sample Line A	6-1	Missing tubing clamp.
Sample Station	6-2	Unanchored constant temperature bath cabinet.
Sample Line A to 1-PT-1-16A/B	6-3	Flexibility concerns on sample lines to 1-PT-1-16A/B, involving broken support.
Sample Line A to 1-PT-1-16A/B	6-4	Missing nuts on U-bolt support.

4.7 Main Steam Bypass (Package 07)

The Main Steam Bypass Valve Chest is located on a loop from the Main Steam Header upstream of the Stop Valves. The Bypass Valves and piping are shown on Flow Diagram 1-47E801-2. This package includes only the portion of Main Steam piping from the Main Steam Manifold to and including the Bypass Valve chest assembly which is a seismic boundary valve (1-FCV-1-61 to -69). The bypass valve chest is supported with trapeze spring hangers on each end of the valve assembly. An outlier pertaining to the seismic adequacy of the Main Steam Bypass Valve was identified, as listed below.

Subsystem Description	POS #	Description of Potential Outlier Condition
MS Bypass Valve Assembly	7-1	MS Bypass valve chest is not covered in the valve screening criteria, thus, its seismic adequacy needs to be verified.

4.8 Main Steam Stop Valve Above Seat Drains (Package 08)

The 1-inch diameter Main Steam Stop Valve Above Seat Drains originate at each of the four stop valves and are shown on Flow Diagram 1-47E807-1. The drains run down through the stop valve platform grating at Elevation 601'-6" and continue east to the four verification boundary valves (1-FCV-6-100 through -103) under the east end of the platform. The piping is typically supported with rod hangers attached from steel angle members welded to the platform steel. The verification boundary valves have operators which are independently supported by rod hangers to the platform supporting beam.

Several potential outliers were identified for this system; one has to do with proximity interaction concern and the rest are maintenance items. The potential outliers are tabulated below.

Subsystem Description	POS #	Description of Potential Outlier Condition
1-FCV-6-101, -102 & -103	8-1	Proximity interactions of valve yokes and beam.
SV-D Above Seat Drain Piping	8-2	Broken pipe strap on support.
1-FCV-6-100	8-3	Broken flex conduit connection to operator.

4.9 Main Steam to Steam Seal Regulator (Package 09)

Steam is taken from the west side of the Manifold between the Stop and Bypass valves and routed to the Steam Seal Feed valves. The line is shown on Flow Diagrams 1-47E801-2 and 1-47E807-2. This package includes the 4-inch diameter steam line off the Manifold to the air operated pressure control/relief valve 1-PCV-1-147 (Steam Seal Regulator) and the normally closed motor operated isolation valves 1-FCV-1-145 (Steam Seal Bypass Valve) and 1-FCV-1-154 (Aux. Steam Isolation Valve) which are all considered as seismic boundary valves. Air instrumentation line for valve 1-PCV-1-147 routed to Panel 1-25-102 located along the west corridor wall at Elevation 617' of the Turbine Building is also included in this package.

Several potential outliers were identified during the walkdown of this system, including valve performance issues, piping overspan, pipe flexibility and falling interaction concerns. The potential outliers are listed below.

Subsystem Description	POS #	Description of Potential Outlier Condition
Main Steam to 1-FCV-1-146	9-1	Overspan condition on the 4"ø MS line to the supply to steam seal MOV 1-FCV-1-146.
Main Steam to 1-FCV-1-146	9-2	Loose rod hanger eye nuts and disengaged rod.
1-PCV-1-147 Air Tubing	9-3	Inadequate flexibility for air line connected to the top of valve.
1-PCV-1-147 Instrumentation	9-4	Falling interaction concern from nearby block wall at El. 617'.
1-PCV-1-147	9-5	Extended valve operator.
1-FCV-1-146, -145 & -154	9-6	Extended valve operator, and substantial operator weight.

4.10 Steam Supply to RFP Turbines (Package 10)

Steam is taken from the east side of the Manifold between the Stop and Bypass valves and routed to the turbine drives for the Reactor Feed Pumps at Elevation 617'. The lines are shown on Flow Diagrams 1-47E801-2 and 1-47E807-2. The 6-inch pipe runs below the turbine operating deck from the Main Steam line and penetrates up through the turbine deck (El. 617') where it branches to 4-inch diameter and into the RFP rooms (A, B & C). The line reduces to 3-inch diameter and continues back down through the turbine deck floor penetration to the SJAE's (Package 11, Section 4.11) and Off-Gas Preheaters

(Package 12, Section 4.12). The line is typically supported on one-way stanchion supports which are susceptible to sliding off the supports during a DBE seismic event.

The 4-inch branch lines that enter into the RFP rooms terminate at the respective RFP Turbine HP Stop Valves, 1-FCV-6-127, -135 and -143 (RFP-A, -B and -C), which are seismic boundary valves. In addition, HP Stop Valve Above Seat Drain Valves 1-FCV-6-122, -127 and -132, as well as HP Steam Isolation Valves 1-FCV-6-153, -155 & -157 are also considered as seismic boundary valves. Instrument tubing to Panels 1-25-100A and 1-25-101B located on the turbine deck are also included in this package.

As such, several supports on the line were identified as potential outliers for further evaluation of the as-installed configuration. Other potential outliers identified include piping overspan, large in-line masses, and seismic interaction concerns. Potential outliers for this system are tabulated below.

Subsystem Description	POS #	Description of Potential Outlier Condition
Steam Supply Line	10-1	Stanchion supports for the steam supply header at El. 617 ' (Total of 6). 1 support for the 6"ø line near H/T4; 2 supports for the 3"ø line near D/T4; and 1 support for each of the 4"ø branch line to RFP Turbine compartments.
Steam Supply Line	10-2	Turbine Bldg. overhead crane.
RFP Stop Valve Above Seat Drains	10-3	Valve operators with large eccentric mass on ½"ø and ¾"ø lines.
Tubing to PI 1-134	10-4	Missing or broken clamps (2 places) on tubing to PI 1-134. Also, tubing is bent and sagging.
Steam Supply Line	10-5	Missing nut on rod hanger support.
Steam Supply Line	10-6	Overspan conditions on the 6"ø steam supply line.

4.11 Steam Supply to Steam Jet Air Ejectors (Package 11)

The Steam Supply line to the Steam Jet Air Ejectors (SJAE's) is shown on Flow Diagram 1-47E801-2. The portions of piping included in this package begin with the 3-inch Steam Supply line that drops from the turbine deck floor penetration near column lines T3 and D (past RFP rooms, refer to Section 4.10 above), runs north along the Off-Gas pipe chase, branches eastward into the SJAE rooms and terminate at the seismic boundary valves 1-PCV-1-151 & -166 in SJAE Room A and 1-PCV-1-153 & -167 in SJAE Room B.

Potential outliers identified for this package, consist of mainly piping overspan on 1-inch diameter steam drains, are listed below.

Subsystem Description	POS #	Description of Potential Outlier Condition
SJAE 's 1A & 1B	11-1	Need to verify the seismic adequacy of SJAE anchorage.
MOV 1-FCV-6-114	11-2	Broken flex conduit connection to valve motor operator.
MS Supply to SJAE's	11-3	Overspan condition on the 3"ø supply line.
MS Supply Drain at SJAE 1B	11-4	Overspan condition on the 1"ø steam supply drain line downstream of 1-FCV-1-114.
Steam Trap Drain at SJAE 1A	11-5	Overspan condition on the 1"ø steam trap drain between 1-FCV-1-172 and 1-PCV-1-166.
Steam Trap Drain at SJAE 1B	11-6	Overspan condition on the 1"ø steam trap drain between 1-FCV-1-173 and 1-PCV-1-167.

4.12 Steam Supply to Off-Gas Preheaters (Package 12)

The Steam Supply line to the Off-Gas Preaheaters is also shown on Flow Diagram 1-47E801-2, and is a continuation of the same steam supply line from RFP's and SJAE's above. The portion of piping included in this package consists of the 2-inch diameter line (reduced from the 3-inch diameter past the SJAE's) along the Off-Gas pipe chase and enters eastward into the Off-Gas room through a masonry block wall. It is to be noted that new manual isolation valves 1-SHV-1-741 & -743 and check valves 1-CKV-1-742 & -744 will be installed in Unit 1 per DCN 51112, consistent with the modifications performed for Units 2 and 3 as part of their MSIV seismic ruggedness programs. These new sets of valves are now the boundary points for the Off-Gas Preheaters.

The only potential outlier for this package, concerning the seismic falling interaction between the piping and its components and the nearby masonry block wall, will then be resolved by this plant modification.

Subsystem Description	POS #	Description of Potential Outlier Condition
Steam Supply Lines to Off-Gas Preheaters	12-1	Lines penetrate into the Off-Gas Preheater room through masonry block wall.

4.13 SJAE's Drain to Condenser (Package 13)

The Steam Jet Air Ejectors (SJAE's) drain to Condenser is shown on Flow Diagrams 1-47E801-2 and 1-47E805-3. The drains originate in the SJAE rooms, at boundary valves 1-6-826 & 1-6-822 downstream of the SJAE's in Rooms A and B, respectively, and tie back into a single 1-1/2 inch diameter line which is routed through the Off-Gas pipe chase and connects into an 8-inch diameter collector pipe that is directly attached to the Condenser 1C shell. The collector pipe has several other drain lines attached to it, thus it was considered to be a potential outlier.

Subsystem Description	POS #	Description of Potential Outlier Condition
SJAE's Drain to Condenser 1C	13-1	Drain to condenser ties into a multi-system collector.

4.14 Main Steam Drain Line (Turbine Building) to Condenser (Pkg. 14)

This package includes the 4-inch diameter Main Steam Drain line downstream of normally closed motor operated 1-FCV-1-59 on Elevation 565' in the Steam Tunnel which is the primary drain path to the Condenser. Past valve 1-FCV-1-59, the MS drain line drops into a small pipe chase, turns north and enters an 8-inch diameter guard pipe that is embedded below the slab. The drain line and the guard pipe continue under the gap between the Steam Vault and the Turbine Structure and re-enter the Turbine Building in the Moisture Separator Level Control Reservoir Room. Since the drain line runs through the guard pipe at ground level, differential building displacements will be minor and, based on a review of drawings, adequate flexibility has been provided to accommodate the movements. The guard pipe ends at the room wall and the drain line runs through the room, exits through the north wall into the condenser bay and then enters Condenser 1A.

Also included in the package is a 4-inch vent line which branches off the above mentioned 4-inch diameter MS drain between valves 1-FCV-1-58 and -59 and rises up to Elevation 593' where it crosses the building separation between the Turbine Building Steam Vault and the Turbine Pedestal Structure. The vent line has adequate flexibility to accommodate anticipated differential seismic displacements between the two structures. The vent line continues on to the south end of Condenser 1A and runs down the gap between the concrete structure and the condenser before connecting back to the 4-inch MS drain prior to its entering the Condenser 1A.

The above lines are shown on Flow Diagram 1-47E-801-1. Support systems for these lines in the Turbine Building are representative of commercial design practice and were determined to be adequate during the walkdown. Potential outliers associated with this package are tabulated below.

Subsystem Description	POS #	Description of Potential Outlier Condition
Main Steam Drain Line to Condenser 1A	14-1	Spring hanger rod detached from 4"ø drain line.
1-FCV-1-59	14-2	Broken flex conduit connection at base.
1-FCV-1-59	14-3	Valve operator (motor) in close proximity to the side rail of steel ladder.

4.15 Condenser (Package 15)

Condensers are the ultimate heat sink for the MSIV alternate leakage treatment drain path. The Main Condenser anchorage was reviewed during the walkdown. Each of the three Condensers is mounted on five concrete pedestals, four at the corners and one in the center of the condenser. The concrete pedestals were observed to be in good condition. Confirmation of the condenser seismic capacity and anchorage adequacy is required to ensure its structural integrity during a DBE seismic event.

Subsystem Description	POS #	Description of Potential Outlier Condition
Condensers 1A, 1B & 1C	15-1	Need to verify the seismic adequacy of the condenser and its anchorage configurations.

4.16 Summary of MSIV Seismic Walkdown Evaluations

A total of fifty-four (54) potential outliers were identified for further evaluation and resolution from the seismic walkdown of the above 15 subsystems under BFN-1 MSIV seismic ruggedness verification program.

Detailed description of the potential outlier conditions, including as-built data, sketches and/or photos are documented in the corresponding walkdown data packages (WDP's) for the respective plant areas. A listing of all walkdown data packages generated under the BFN-1 MSIV seismic ruggedness verification program is provided in Appendix A.

5. POTENTIAL OUTLIER RESOLUTION

Potential outliers identified during the in-plant screening walkdowns, as documented in the respective Potential Outlier Sheet (POS) and tabulated in Table 4-1, were further evaluated for appropriate resolution. Further evaluations and bounding analyses of these potential outliers consisted of hand calculations using basic engineering mechanics techniques for simple configurations, and rigorous piping analyses using TPIPE computer program (Reference 7-11) for more complex piping configurations.

Outlier evaluation guidelines and acceptance criteria are discussed in Section 5.1 below. Results of the outlier evaluation are summarized in Section 5.2.

5.1 Outlier Evaluation Guidelines

Conditions which do not meet the above in-plant screening guidelines or which were judged by the Walkdown Team members to require further reviews are documented as "Potential Outliers". Technical bases and methods used for further analysis and evaluation of these potential outliers are based on industry standard engineering practices, and are consistent with those recommended in BWROG Report GE NEDC-31858P (Reference 7-1), Generic Implementation Procedure (GIP, Reference 7-3) and TVA Design Criteria BFN-50-C-7306 (Reference 7-7), as applicable. Realistic effects of non-linear behavior due to design features and phenomena such as proximity impact with other plant features, interferences and small clearances to stiff structures, geometric restoring forces, wall penetration sealants, and support ductile behavior are considered in the analysis as appropriate.

5.1.1 Seismic Demand

Turbine Building at Browns Ferry Nuclear Plant is classified as a Class II structure per BFN FSAR (Reference 7-11). As such, no dynamic response analysis was performed for the building. The building below the operating floor (El. 617 ft.) is a reinforced concrete framed structure supported on steel H-piles to bedrock. The superstructure above the operating floor consists of welded steel rigid braced frames. As a Class II structure, Turbine Building was designed to seismic zone 1 and a wind speed of 100 mph per Uniform Building Code (Reference 7-13), utilizing industry-standard design guidelines such as AISC (Reference 7-14) and ACI (Reference 7-15).

Majority of the MSIV alternate leakage treatment piping and associated components within the MSIV seismic verification boundary, including the condensers, are located in the Turbine Building. Since no in-structure response spectra were available for the Turbine Building, horizontal seismic demand for components located within about 40 feet of the Turbine Building effective grade elevation (El. 568 ft.) is taken as the BFN 5% damped design basis DBE input spectrum (0.2g) scaled by 1.6 for soil amplification per BFN FSAR (Reference 7-11), and 1.5 for building amplification per GIP (Reference 7-3). For components located above 40 feet of the Turbine Building effective grade elevation, an additional amplification factor of 1.5 is conservatively applied. In the vertical direction, seismic demand is taken as 2/3 that of the horizontal direction, with a soil amplification factor of 1.1 instead of 1.6 per BFN FSAR (Reference 7-11).

5.1.2 Equipment Anchorage Acceptance Criteria

Unanchored, unrestrained, marginally or inadequately anchored equipment components identified as potential outliers are subject to further evaluations. Anchorage capacities are based on those provided in Appendix C of the Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment (Reference 7-3) with appropriate reduction factors as applicable. Piping and tubing attached to equipment components with flexible support systems, such as vibration isolators, are evaluated for seismic anchor movement.

5.1.3 Pipe Support Acceptance Criteria

Pipe support anchorage loads are verified against capacities as provided in Appendix C of the GIP (Reference 7-3). Support components that may exhibit non-ductile behavior are accepted based the following stress allowables:

Flexural and tensile stresses:	lesser of 0.7 S_u and 1.2 S_y
Shear stresses:	lesser of 0.42 S_u and 0.72 S_y
Bolt stresses:	greater of 0.7 S_u and minimum specified S_y

Where, S_u is the material's ultimate strength
S_y is the material's yield strength

When test data are available, acceptable loads based on test data consider mean less one standard deviation capacity.

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

5.1.4 Pipe Stress Acceptance Criteria

Pipe stresses induced by the combination of normal operating loads (dead load and pressure) and seismic loads (DBE inertial loads and DBE seismic anchor movements) are limited to 2.0 S_y.

Majority of the piping and components within the MSIV seismic verification boundary are associated with the Main Steam system which is high energy (i.e., high pressure and temperature), thus the effects from these loads may be significant. Although most piping within the seismic boundary are typically supported by rod hangers, thermal effects are included in the evaluations when judged to be significant, as in the case of rigid support systems, or when addressing proximity interaction and piping flexibility issues.

5.2 Outlier Evaluation Results

Results of further evaluations and analyses to resolve the identified potential outliers are presented in Table 5-1 by the designated subsystems. A listing of the calculation packages associated with further evaluations and bounding analyses of potential outliers identified under the BFN-1 MSIV seismic ruggedness verification program is provided in Appendix B of this report.

For those outliers not meeting the evaluation acceptance criteria as described in the above Section 5.1, plant modifications are designed in accordance with TVA Design Criteria BFN-50-C-7306 (Reference 7-7) and implemented accordingly to resolve these outlier conditions. Other miscellaneous maintenance and/or housekeeping items are resolved through the issuance of work orders. Discussions of the plant modifications including general maintenance and/or housekeeping items are presented in Chapter 6 of this report.

TABLE 5-1
OUTLIER RESOLUTION SUMMARY
BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION PROGRAM

System Description	Outlier		Seismic Issue*					Recommended Resolution
	No.	Description of Condition	A	F	P	D	V	
1.0 Main Steam Drain Line – Turbine Bldg. Main Steam Vault								
Main Steam Drain Taps	1-1	Flexibility concerns associated with the MS lines and Drains.				√		As-installed configurations were evaluated and found to be acceptable per calculation.
MOV 's 1-FCV-1-57 & 58	1-2	Excessive mass and extended valve operators.					√	As-installed configurations were evaluated and found to be acceptable per calculation.
Main Steam and HPCI Drains	1-3	Flexibility concerns associated with the MS and HPCI Drains.				√		As-installed configurations were evaluated and found to be acceptable per calculation.
Main Steam Bypass Drain	1-4	Proximity interaction concerns between bypass line and adjacent 2-way support.			√			As-installed configurations were evaluated and found to be acceptable per calculation.
2.0 Main Steam Lines – Turbine Bldg.								
MS Balancing Header	2-1	Upper pipe clamp nut is missing.	√					Work order 98-001447-001 has been in place. **(WO)**
MS Turbine Stop/Control Valves	2-2	MS stop/control valves are not covered in the valve screening criteria, thus, seismic adequacy of these valves needs to be verified.					√	Seismic adequacy of these valves is presented in a calculation.
3.0 Main Steam Drain Line – Reactor Building MSIV Vault								
MSIV 's 1-FCV-1-15, 27, 38 & 52	3-1	MSIV 's are not covered in the valve screening criteria, thus, seismic adequacy of these valves needs to be verified.					√	Seismic adequacy of these valves is presented in a calculation.
MOV 1-FCV-1-56	3-2	Excessive mass and extended valve operators.					√	As-installed configurations were evaluated and found to be acceptable per calculation.
4.0 HPCI/RCIC/Aux. Boiler Drains								
RCIC Drain	4-1	Excessive pipe overspan condition.		√				Overspan condition found to be unacceptable by calculation. Additional support to be installed per calculation. **(M)**
RCIC Drain	4-2	Pipe support with questionable capacity.	√					Pipe support OK per calculation.
Aux. Steam Boiler	4-3	Missing rod hanger hardware (eye nut).	√					Issue work order to re-install the missing nut. **(WO)**
HPCI Drain	4-4	Multiple pipe overspan conditions on the 1"ø drain line.		√				As-installed configurations were evaluated and found to be acceptable per calculation.
HPCI Line	4-5	Multiple pipe overspan conditions on the 2"ø line.		√				As-installed configurations were evaluated and found to be acceptable per calculation.
HPCI Line and Valve 1-73-222	4-6	Proximity interaction of the 2"ø HPCI line and the valve.			√			As-installed configurations were evaluated and found to be acceptable per calculation.

TABLE 5-1 (CONT'D)

OUTLIER RESOLUTION SUMMARY
BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION PROGRAM

System Description	Outlier		Seismic Issue*					Recommended Resolution
	No.	Description of Condition	A	F	P	D	V	
4.0 HPCI/RCIC/Aux. Boiler Drains (cont'd)								
HPCI Drain	4-7	Several floor-mounted supports were noted to be damaged (potentially due to thermal effects). Also, anchor bolt spacing violations on these typical supports.	√					Existing support anchorage capacities are adequate per calculation. However, the damaged supports located just passed the blowout panel needs to be repaired and longer lugs are required to avoid pipe from sliding off the existing supports due to thermal movements. **(M)** *Use 3" thermal movement as a guide for the design of support mods.*
HPCI Drain	4-8	Flexibility concern on 2"ø HPCI line due to differential displacement between RB and TB				√		As-installed configurations were evaluated and found to be acceptable per calculation.
5.0 MS PT 1-72, -76, -82, -86 & -93								
SS Tubing to 1-PT-1-72 & 76	5-1	Overspan conditions for both lines. Also, flexibility issue exists on tubing to 1-PT-1-72.	√	√		√		Replace existing support at N-end with rod hanger; and add new rod hanger support at S-end. **(M)**
SS Tubing to 1-PT-1-82	5-2	Overspan conditions due to broken weld on the support and missing clamps.	√	√				Replace both existing supports at N- and S-ends with rod hangers. **(M)**
SS Tubing to 1-PT-1-86	5-3	Overspan conditions due to broken weld on the support and loose clamp.	√	√				Replace both existing supports at N- and S-ends with rod hangers. **(M)**
SS Tubing to 1-PT-1-86	5-4	Flexibility concern on ½"ø tubing.				√		PER 04-000616-000 was generated to identify the required work order to remove existing clamp on support. **(WO)**
SS Tubing to PT 1-93	5-5	Overspan condition due to broken support.	√	√				Replace existing support with rod hanger. **(M)**
SS Tubing to PT 1-93	5-6	Flexibility concern on ½"ø tubing along the wall.	√			√		PER 04-000616-000 was generated to identify the required work order to remove existing clamp on the spacer support. **(WO)**
6.0 Main Steam Sample Lines to Sampling Station								
Sample Line A	6-1	Missing tubing clamp.	√					PER 04-000616-000 was generated to identify the required work order to re-install the missing clamp. **(WO)**
Sample Station	6-2	Unanchored constant temperature bath cabinet.	√					Provide positive anchorage to the cabinet. **(M)**
Sample Line A to 1-PT-1-16A/B	6-3	Flexibility concerns on sample lines to 1-PT-1-16A/B, involving broken support.	√	√				Replace existing wall-mounted support with rod hangers. **(M)**
Sample Line A to 1-PT-1-16A/B	6-4	Missing nuts on U-bolt support.	√					PER 04-000616-000 was generated to identify the required work order to re-install the missing nuts. **(WO)**

TABLE 5-1 (CONT'D)

OUTLIER RESOLUTION SUMMARY
BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION PROGRAM

System Description	Outlier		Seismic Issue*					Recommended Resolution
	No.	Description of Condition	A	F	P	D	V	
7.0 Main Steam Bypass								
MS Bypass Valve Assembly	7-1	MS Bypass valve chest is not covered in the valve screening criteria, thus, its seismic adequacy needs to be verified.					√	Seismic adequacy of the bypass valve assembly is presented in a calculation.
8.0 MS Stop Valve Above Seat Drains								
1-FCV-6-101, -102 & -103	8-1	Proximity interactions of valve yokes and beam.			√			As-installed configurations were evaluated in a calculation. Valves 101 & 102 were found to be OK as-is, but not for valve 103. Initiate work request to cope the bottom flange of the WF beam supporting the steel grating at El. 601'-6". **(WO)** *Note that a lateral (E-W) clearance of 3" minimum should be provided to preclude potential seismic impact between the valve and nearby platform steel.*
SV-D Above Seat Drain Piping	8-2	Broken pipe strap on support.	√					PER 04-000616-000 was generated to identify the required work order to replace broken pipe strap. **(WO)**
1-FCV-6-100	8-3	Broken flex conduit connection to operator.					√	PER 04-000617-000 was generated to identify the required work order to repair broken flex conduit. **(WO)**
9.0 Steam to Steam Seal Regulator								
Main Steam to 1-FCV-1-146	9-1	Overspan condition on the 4"ø MS line to the supply to steam seal MOV 1-FCV-1-146.	√	√				As-installed configurations were evaluated and found to be acceptable per calculation.
Main Steam to 1-FCV-1-146	9-2	Loose rod hanger eye nuts and disengaged rod.	√	√				PER 04-000616-000 was generated to identify the required work orders (2) to correct rod hanger hardware deficiencies. **(WO)**
1-PCV-1-147 Air Tubing	9-3	Inadequate flexibility for air line connected to the top of valve.				√		As-installed configurations were evaluated and found to be acceptable per calculation.
1-PCV-1-147 Instrumentation	9-4	Falling interaction concern from nearby block wall at El. 617'.		√	√			Re-route tubing and protect instrumentation panel, as necessary, to preclude falling interactions. **(M)**
1-PCV-1-147	9-5	Extended valve operator.					√	As-installed configurations were evaluated and found to be acceptable per calculation.
1-FCV-1-146, 145 & 154	9-6	Extended valve operator, and substantial operator weight.					√	As-installed configurations were evaluated and found to be acceptable per calculation.

TABLE 5-1 (CONT'D)

OUTLIER RESOLUTION SUMMARY
BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION PROGRAM

System Description	Outlier		Seismic Issue*					Recommended Resolution
	No.	Description of Condition	A	F	P	D	V	
10.0 Steam Supply to RFP Turbines								
Steam Supply Line	10-1	Stanchion supports for the steam supply header at El. 617 ' (Total of 6). 1 support for the 6"ø line near H/T4; 2 supports for the 3"ø line near D/T4; and 1 support for each of the 4"ø branch line to RFP Turbine compartments.		√				Modify existing stanchion supports for the postulated DBE seismic movements. A total of 3 supports require modifications – one in the S-end (H/T4) and two in the N-end (D/T4) of the steam supply header at the Turbine deck, El. 617. Note that similar type supports inside each of the RFP Turbine rooms are OK as-is. (M)
Steam Supply Line	10-2	Turbine Bldg. overhead crane.			√			Resolved per calculation.
RFP Stop Valve Above Seat Drains	10-3	Valve operators with large eccentric mass on ½"ø and ¾"ø lines.					√	As-installed configurations were evaluated and found to be acceptable per calculation.
Tubing to PI 1-134	10-4	Missing or broken clamps (2 places) on tubing to PI 1-134. Also, tubing is bent and sagging.	√	√				PER 04-000616-000 was generated to identify the required work order to re-install tubing clamps and repair bent tubing. (WO)
Steam Supply Line	10-5	Missing nut on rod hanger support.	√					PER 04-000616-000 was generated to identify the required work order to re-install the missing nut. (WO)
Steam Supply Line	10-6	Overspan conditions on the 6"ø steam supply line.	√	√				As-installed configurations were evaluated and found to be acceptable per calculation.
11.0 Steam Supply to SJAE's								
SJAE 's 1A & 1B	11-1	Need to verify the seismic adequacy of SJAE anchorage.	√					Seismic adequacy of SJAE anchorage was verified per calculation.
MOV 1-FCV-6-114	11-2	Broken flex conduit connection to valve motor operator.					√	PER 04-000617-000 was generated to identify the required work order to repair the broken flex conduit connection to valve motor operator. (WO)
MS Supply to SJAE's	11-3	Overspan condition on the 3"ø supply line.	√	√				As-installed configurations were evaluated and found to be acceptable per calculation.
MS Supply Drain at SJAE 1B	11-4	Overspan condition on the 1"ø steam supply drain line downstream of 1-FCV-1-114.	√	√				As-installed configurations were evaluated and found to be acceptable per calculation.
Steam Trap Drain at SJAE 1A	11-5	Overspan condition on the 1"ø steam trap drain between 1-FCV-1-172 and 1-PCV-1-166.	√	√				Overspan condition was found to be unacceptable per calculation. Add a new support near T4 wall (rod hanger). (M)
Steam Trap Drain at SJAE 1B	11-6	Overspan condition on the 1"ø steam trap drain between 1-FCV-1-173 and 1-PCV-1-167.	√	√				Overspan condition was found to be unacceptable per calculation. Add a new support near T4 wall (rod hanger). (M)

TABLE 5-1 (CONT'D)

OUTLIER RESOLUTION SUMMARY
BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION PROGRAM

System Description	Outlier		Seismic Issue*					Recommended Resolution
	No.	Description of Condition	A	F	P	D	V	
12.0 Steam Supply to Off-Gas Preheaters								
Steam Supply Lines to Off-Gas Preheaters	12-1	Lines penetrate into the Off-Gas Preheater room through masonry block wall.		√	√			Re-route pipes to preclude falling interactions. Note that pipe re-routing will be done in conjunction with the installation of new isolation valves (1-SHV-1-741 & 743 and 1-CKV-1-742 & 744) under DCN 51112. **(M)**
13.0 SJAE's Drain to Condenser								
SJAE's Drain to Condenser 1C	13-1	Drain to condenser ties into a multi-system collector.						Re-route drain line to achieve a direct path to condenser. **(M)**
14.0 Main Steam Drain Line to Condenser								
Main Steam Drain Line to Condenser 1A	14-1	Spring hanger rod detached from 4"ø drain line.	√					PER 04-000616-000 was generated to identify the required maintenance work order to re-connect the detached rod to pipe. **(WO)**
1-FCV-1-59	14-2	Broken flex conduit connection at base.					√	PER 04-000617-000 was generated to identify the required work order to repair broken flex conduit. **(WO)**
1-FCV-1-59	14-3	Valve operator (motor) in close proximity to the side rail of steel ladder.			√			As-installed configurations were evaluated and found to be unacceptable per calculation. Valve should be re-oriented with about 6" clearance to the nearest fixed structure to the north. **(M)** *Note that the 4"ø MS drain and vent pipes are to be replaced under DCN 51112. As such, provide 6" clearance when installing the new pipes.*
15.0 Condensers								
Condensers 1A, 1B & 1C	15-1	Need to verify the seismic adequacy of the condenser and its anchorage configurations.	√					Condenser and its anchorage configurations were evaluated and found to be acceptable per calculation.

* <u>Seismic Issue Notations:</u>
A = Anchorage/Support Capacity; **F** = Failure/Falling Interaction; **P** = Proximity & Impact; **D** = Differential Displacement; **V** = Valve Screening & Performance

Note: Items that require engineering design modifications are listed as (M), and are to be resolved by various DCN's.
Items that require resolution by work order requests as listed as (WO).

6. SUMMARY AND RECOMMENDATIONS

A total of fifty-four (54) potential outliers were identified in the MSIV seismic ruggedness verification walkdown for BFN-1. Majority of these potential outliers were resolved by performing further analyses and evaluations to the acceptance criteria of TVA Design Criteria BFN-50-C-7306 (Reference 7-7). For the remaining outliers that did not meet the above evaluation criteria, plant design modifications are developed for appropriate resolution. In addition, work requests are initiated for those outliers that fall into the general category of maintenance and housekeeping items, such as missing support hardware, broken parts, proximity interaction and flexibility concerns and others. Table 6-1 presents a summary of the BFN-1 MSIV seismic ruggedness verification program.

6.1 Plant Modifications

Plant design modifications are developed for outliers not meeting the acceptance requirements of TVA Design Criteria BFN-50-C-7306 (Reference 7-7). Design modifications ranged from simple support hardware modifications to addition of new pipe supports, rerouting of piping and others. A total of fifteen (15) plant design modifications were implemented for the resolution of these MSIV seismic ruggedness outliers. A brief description of each of the plant modifications is provided in Table 6-2.

Engineering designs of the plant modifications are documented in various calculation packages listed in Appendix C of this report. These plant modifications will be implemented under several Design Change Notices (DCN's 51112, 51669, and 51126).

6.2 Maintenance and Housekeeping Items

In addition to the plant modifications discussed in Section 6.1 above, a total of fifteen (15) miscellaneous maintenance and housekeeping items were identified for appropriate actions and resolution. These maintenance items, along with a brief description, are tabulated in Table 6-3. These items will be disposed through maintenance work requests.

TABLE 6-1

Summary of BFN-1 MSIV Seismic Ruggedness Verification Program

Sub-system No.	Subsystem Description	No. of Potential Outliers	No. of Design Mods	No. of Maint. Items
1	MS Drain Line – Turbine Bldg. Main Steam Tunnel	4	0	0
2	MS Lines – Turbine Bldg.	2	0	1
3	MS Drain Line – Reactor Bldg. Main Steam Vault	2	0	0
4	HPCI/RCIC/Aux. Boiler Drains	8	2	1
5	MS Pressure Transmitters PT 1-72, 76, 82, 86 & 93	6	4	2
6	MS Sample Lines to Sampling Station	4	2	2
7	MS Bypass	1	0	0
8	MS Stop Valve Above Seat Drains	3	0	3
9	MS to Steam Seal Regulator	6	1	1
10	Steam Supply to RFP Turbines	6	1	2
11	Steam Supply to Steam Jet Air Ejectors (SJAE's)	6	2	1
12	Steam Supply to Off-Gas Preheaters	1	1	0
13	SJAE's Drain to Condenser	1	1	0
14	MS Drain Line (Turbine Bldg.) to Condenser	3	1	2
15	Condenser	1	0	0
	Total	**54**	**15**	**15**

TABLE 6-2

Summary of Plant Modifications
BFN-1 MSIV Seismic Ruggedness Verification Program

Item	Ref. (1)	Description of Plant Modification	Reference DCA Drawing No. (2)
1	POS 4-1	Add one (1) new support.	1-47B456-2100 see note (3)
2	POS 4-7	Add four (4) new supports.	1-47B400-2030 to -2033
3	POS 5-1	Add two (2) new supports.	1-47B400-2026 & -2027
4	POS 5-2	Add two (2) new supports.	1-47B400-2022 & -2023
5	POS 5-3	Add two (2) new supports.	1-47B400-2024 & -2025
6	POS 5-5	Add one (1) new support.	1-47B400-2021
7	POS 6-2	Provide anchorage to the Constant Temperature Bath Cabinet.	1-48B879-1 see note (4)
8	POS 6-3	Add one (1) new support.	1-47B400-2020
9	POS 9-4	Reroute tubing to preclude seismic falling interaction.	see note (2)
10	POS 10-1	Add three (3) new supports.	1-47B400-2034-1 & -2, and 1-47B400-2035 & -2036

TABLE 6-2 (cont'd)

Summary of Plant Modifications
BFN-1 MSIV Seismic Ruggedness Verification Program

Item	Ref. (1)	Description of Plant Modification	Reference DCA Drawing No. (2)
11	POS 11-5	Add one (1) new support.	1-47B400-2028
12	POS 11-6	Add one (1) new support.	1-47B400-2029
13	POS 12-1	Reroute piping to preclude seismic falling interaction.	see note (2)
14	POS 13-1	Reroute drain line directly to the condenser.	see note (2)
15	POS 14-3	Relocate valve to preclude seismic proximity interaction.	see note (2)

(1) Detailed description and as-built information of these items, including photos and/or sketches, can be found in the Potential Outlier Sheet (POS) contained in the respective MSIV seismic ruggedness Walkdown Data Packages (WDP's).

(2) Refer to DCN 51112, unless noted otherwise.

(3) Refer to DCN 51669.

(4) Refer to DCN 51126.

TABLE 6-3

Summary of Misc. Maintenance & Housekeeping Items
BFN-1 MSIV Seismic Ruggedness Verification Program

Item	Ref. *	Description	Recommended Action
1	POS 2-1	Replace missing upper pipe clamp nut.	Work Order WO # 98-001447-001 was issued.
2	POS 4-3	Replace missing eye nut.	Work Order WO # 04-712923-000 was issued.
3	POS 5-4	Remove existing clamp on pipe support.	Work Order WO # 04-712923-000 was issued. (See also DCA 51112-131)
4	POS 5-6	Remove existing clamp on pipe support.	Work Order WO # 04-712923-000 was issued. (See also DCA 51112-132)
5	POS 6-1	Replace missing tubing clamp.	Work Order WO # 04-712923-000 was issued.
6	POS 6-4	Replace missing nut on U-bolt support.	Work Order WO # 04-712923-000 was issued.
7	POS 8-1	Cope the bottom flange of the WF beam to provide 3" clearance (E-W) to preclude seismic proximity interaction.	PIC 61400 was added to DCN 51112 to initiate the work.
8	POS 8-2	Replace broken pipe strap on support.	Work Order WO # 04-712923-000 was issued.
9	POS 8-3	Repair broken flex conduit connection.	Work Order WO # 04-712926-000 was issued.
10	POS 9-2	Replace loose rod hanger eye nuts and disengaged rod.	Work Order WO # 04-712923-000 was issued.

TABLE 6-3 (cont'd)

Summary of Misc. Maintenance & Housekeeping Items
BFN-1 MSIV Seismic Ruggedness Verification Program

Item	Ref. *	Description	Recommended Action
11	POS 10-4	Replace missing or broken tubing clamps and repair bent tubing.	Work Order WO # 04-712923-000 was issued.
12	POS 10-5	Replace missing nut on rod hanger support.	Work Order WO # 04-712923-000 was issued.
13	POS 11-2	Repair broken flex conduit connection.	Work Order WO # 04-712926-000 was issued.
14	POS 14-1	Reinstall detached rod to pipe support.	Work Order WO # 04-712923-000 was issued.
15	POS 14-2	Repair broken flex conduit connection.	Work Order WO # 04-712926-000 was issued.

* Detailed description and as-built information of these items, including photos and/or sketches, can be found in the Potential Outlier Sheet (POS) contained in the respective MSIV seismic ruggedness Walkdown Data Packages (WDP's).

7. REFERENCES

7-1 BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems, GE NEDC-31858P, Revision 2. September 1993.

7-2 TVA calculation, "MSIV Leakage Containment System Boundaries, Physical Properties, System 001." Rev. 3.

7-3 "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment", Revision 2A, March 1993. Seismic Qualification Utility Group (SQUG).

7-4 EPRI Report NP-7149. March 1991. "Summary of the Seismic Adequacy of Twenty Classes of Equipment Required for Safe Shutdown of Nuclear Plants." Electric Power Research Institute, Palo Alto, California.

7-5 EPRI Report RP-2635-1. February 1987. "Piping Seismic Adequacy Criteria Recommendation Based on Performance During and After Earthquakes." Volumes 1 and 2. Electric Power Research Institute, Palo Alto, California.

7-6 TVA Walkdown Instruction WI-BFN-0-CEB-07, "Engineering Walkdown Instruction for MSIV Seismic Ruggedness Verification." Revision 0.

7-7 TVA Detailed Design Criteria BFN-50-C-7306, "Qualification Criteria for Seismic Class II Piping, Pipe Supports, and Components." Revision 1.

7-8 TVA calculation, "In-Plant Screening Tools for Seismic II/I Spray Walkdown Evaluations." Revision 1.

7-9 TVA calculation, "Main Steam Seismic Ruggedness Verification." Revision 5.

7-10 TVA calculation, "Main Steam Seismic Ruggedness Evaluation." Revision 1.

7-11 "*TPIPE* Program User Manual." Version 16, July 1994.

7-12 TVA Browns Ferry Nuclear Plant Final Safety Analysis Report (FSAR).

7. REFERENCES (CONT'D)

7-13 Uniform Building Code.

7-14 American Institute of Steel Construction (AISC). "Manual of Steel Construction." 6th Edition.

7-15 American Concrete Institute (ACI). "Building Code Requirements for Reinforced Concrete." ACI 318-1963 Edition.

7-16 USAS B31.1.0, "Power Piping." 1967 Edition.

APPENDIX A:

BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION

WALKDOWN DATA PACKAGES

BFN-1 MSIV Seismic Ruggedness Verification
Walkdown Data Packages

Walkdown Data Package (WDP)	Title	Rev.
BFN1-CEB-MSIV-01	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 1	0
BFN1-CEB-MSIV-02	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 2	0
BFN1-CEB-MSIV-03	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 3	0
BFN1-CEB-MSIV-04	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 4	0
BFN1-CEB-MSIV-05	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 5	0
BFN1-CEB-MSIV-06	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 6	0
BFN1-CEB-MSIV-07	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 7	0
BFN1-CEB-MSIV-08	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 8	0
BFN1-CEB-MSIV-09	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 9	0
BFN1-CEB-MSIV-10	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 10	0
BFN1-CEB-MSIV-11	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 11	0
BFN1-CEB-MSIV-12	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 12	0
BFN1-CEB-MSIV-13	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 13	0
BFN1-CEB-MSIV-14	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 14	0
BFN1-CEB-MSIV-15	MSIV Seismic Ruggedness Walkdown Screening Evaluation Documentation for BFN Unit 1 – Package 15	0

APPENDIX B:

BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION

OUTLIER RESOLUTION CALCULATION PACKAGES

BFN-1 MSIV Seismic Ruggedness Verification
Outlier Resolution Calculation Packages

Title	Rev.
Main Steam Drain Line, MSIV Ruggedness Seismic Analysis – Resolution of POS 1-1, 1-3 & 1-4	0
HPCI Drain Line, MSIV Ruggedness Seismic Analysis – Resolution of POS 4-6, 4-7 & 4-8	0
Misc. Main Steam Valve Performance Issues, MSIV Ruggedness Seismic Analysis – Resolution of POS 2-2, 3-1 & 7-1	1
Misc. Extended Valve Operator Issues, MSIV Ruggedness Seismic Analysis – Resolution of POS 1-2, 3-2, 9-5, 9-6 & 10-3	0
Misc. Piping Overspan Issues, MSIV Ruggedness Seismic Analysis – Resolution of POS 4-4, 4-5, 9-1, 10-6, 11-3, 11-4, 11-5 & 11-6	0
Misc. Seismic Interaction Issues, MSIV Ruggedness Seismic Analysis – Resolution of POS 10-1, 10-2 & 11-1	0
Misc. Valve Proximity Interaction Issues, MSIV Ruggedness Seismic Analysis – Resolution of POS 8-1, 9-3 & 14-3	0
Seismic Verification of Condenser and its Anchorage, MSIV Ruggedness Seismic Analysis – Resolution of POS 15-1	0
RCIC Line, MSIV Ruggedness Seismic Analysis – Resolution of POS 4-1	1
RCIC Line, MSIV Ruggedness Seismic Analysis – Resolution of POS 4-2	0

APPENDIX C:

BFN-1 MSIV SEISMIC RUGGEDNESS VERIFICATION

PLANT MODIFICATION DESIGN CALCULATION PACKAGES AND MAINTENANCE WORK ORDERS

BFN-1 MSIV Seismic Ruggedness Verification
Plant Modification Design Calculation Packages and Work Orders

Potential Outlier No.*	Calculation or Work Order No.	Calculation Title or Description of Work Order
2-1 (W)	WO 98-001447-001	Replace missing upper pipe clamp nut
4-1 (M)	Calculation	*RCIC Line, MSIV Ruggedness Seismic Analysis – Resolution of POS 4-1, Rev. 1*
4-3 (W)	WO 04-712923-000	Replace missing eye nut
4-7 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
5-1 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
5-2 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
5-3 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
5-4 (W)	WO 04-712923-000 (see also DCA 51112-131)	Remove pipe clamp on existing support
5-5 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
5-6 (W)	WO 04-712923-000 (see also DCA 51112-132)	Remove pipe clamp on existing support
6-1 (W)	WO 04-712923-000	Replace missing tubing clamp
6-2 (M)	Calculation	*Design of Anchorage for Bath and Circulator Cabinet for Unit 1, Rev. 0*
6-3 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
6-4 (W)	WO 04-712923-000	Replace missing nut on U-bolt support

* (M) – Plant Modifications (W) – Work

BFN-1 MSIV Seismic Ruggedness Verification
Plant Modification Design Calculation Packages and Work Orders

Potential Outlier No.*	Calculation or Work Order No.	Calculation Title or Description of Work Order
8-1 (W)	Refer to DCN 51112 PIC 61400	Cope bottom flange of the WF beam to provide minimum of 3" clearance in the E-W direction
8-2 (W)	WO 04-712923-000	Replace broken pipe strap on the support
8-3 (W)	WO 04-712926-000	Replace broken flex conduit connection
9-2 (W)	WO 04-712923-000	Replace loose rod hanger eye nuts and the disengaged rod
9-4 (M)	Refer to DCN 51112	Reroute instrument tubing
10-1 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
	Calculation	*Main Steam Seismic Ruggedness Verification, Units 2 & 3, Rev. 5*
10-4 (W)	WO 04-712923-000	Replace missing or broken tubing clamps and repair bent tubing
10-5 (W)	WO 04-712923-000	Replace missing nut on rod hanger support
11-2 (W)	WO 04-712926-000	Replace broken flex conduit connection
11-5 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
11-6 (M)	Calculation	*Main Steam Seismic Ruggedness Verification, Unit 1, Rev. 0*
12-1 (M)	Refer to DCN 51112	Reroute piping
13-1 (M)	Refer to DCN 51112	Reroute drain line to condenser
14-1 (W)	WO 04-712923-000	Reinstall detached rod to pipe support
14-2 (W)	WO 04-712926-000	Replace broken flex conduit connection
14-3 (M)	Refer to DCN 51112	Relocate valve

* (M) – Plant Modifications (W) – Work Orders

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