TVA

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

January 21, 1997

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 1, 2, AND 3 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI - ADOPTION OF 1989 CODE EDITION FOR INSERVICE TESTING (IST) OF VALVES

This letter provides notification that BFN is converting the code of record for IST of valves from the 1986 edition to the 1989 edition of ASME Section XI. This conversion is in accordance with 10 CFR 50.55a(f)(4)(iv) and NUREG-1482, Section 1.3, and is expected to be complete by August 1, 1997. TVA understands that it must comply with the 1989 edition of ASME Section XI, Article IWV in its entirety.

In conjunction with this conversion, BFN will use ASME/American National Standards Institute Operations and Maintenance (OM) Standards, Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants," (OM-10) for leak rate testing of Category A and AC valves. BFN will continue to perform IST of pumps in accordance with the 1986 edition of ASME Section XI, Article IWP.

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BFN is also converting to performance-based testing of containment isolation valves (CIV) in accordance with Option B of 10 CFR 50 Appendix J. Option B of 10 CFR 50 Appendix J was approved for BFN by NRC letter dated February 22, 1996, which transmitted Technical Specifications Amendment Numbers 228, 243, and 203. The CIVs are included in the IST program as Category A or AC valves, and ASME Section XI requires leak rate testing for these valves. BFN has been using the 10 CFR 50, Appendix J, test program to monitor Category A and AC valve leak rates in accordance with Generic letter 89-04, position 10. BFN will continue to utilize the 10 CFR 50, Appendix J, leak rate program (including the newly approved Option B) to test Category A and AC CIVs.

Since the Appendix J program will be performance-based, not all CIVs in the IST program will be tested every refueling outage. TVA will use OM-10 for leak rate testing of Category A and AC valves. Use of OM-10 for leak rate testing of Category A and AC valves is accepted by NUREG-1482, Section 4.4.5 provided the applicable paragraphs (4.2.2.1 and 4.2.2.3) of OM-10 are followed. For Category A and AC valves that perform a function other than containment isolation (i.e., pressure isolation valves), testing will continue every refueling outage for seat leakage in accordance with OM-10, paragraph 4.2.2.3.

TVA has previously submitted Requests for Relief PV-25 and PV-37 for the Residual Heat Removal (FCV-74-54 and -68) and Core Spray (CS) (FCV-75-26 and -54) testable check valves, respectively. These relief requests sought relief from full stroking of the valves during cold shutdowns and refueling outages. Relief Request PV-25 has been accepted by NRC, however, the staff has requested additional information for PV-37. As a result of adopting the 1989 ASME Section XI Code and OM-10, TVA is canceling PV-25 and withdrawing PV-37. In their place, TVA is submitting Refueling Outage (RO) Justification RO-1 and RO-2. The use of Refueling

Outage Justifications is described in NUREG-1482, Sections 2.4.2 and 3.5. These Refueling Outage Justifications document the BFN alternate testing methods, permitted by OM-10, for the RHR and CS testable check valves. Therefore, specific NRC approval for RO-1 and RO-2 is not required.

Enclosure 1 to this letter provides an update to the BFN IST program for valves that reflects the adoption of the ASME Section XI, 1989 edition of the code, Option B (performance-based testing) of 10 CFR 50 Appendix J, and OM-10. Enclosure 2 to this letter provides Refueling Outage Justifications RO-1 and RO-2.

There are no new commitments contained in this letter. If you have any questions, please call me at (205) 729-2636.

Sincerely,

T. E. Abney

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosures
cc: See page 4

Enclosures
cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3
AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI
ADOPTION OF 1989 EDITION OF CODE FOR
INSERVICE TESTING OF VALVES

PROGRAM REVISION

(SEE ATTACHED)

I. INTRODUCTION/POLICY STATEMENTS

A. Introduction

10 CFR 50.55a(f)(4)(ii) requires the Inservice Testing (IST) of nuclear power facility pumps and valves whose functions are required for safety. This testing is required to be conducted in 120-month intervals, beginning with initial power operation and continuing throughout the service life of the facility. The testing must comply with the latest edition and addenda of American Society of Mechanical Engineers (ASME) Section XI, incorporated by reference in 10 CFR 50.55a(b), 12 months prior to the start of the 120-month interval in question. Specific exceptions to these requirements are included here as Relief Requests. BFNP Units 1, 2, and 3 are on a concurrent IST interval. BFNP completed its initial 120-month interval on August 31, 1992 and began the second 120-month interval on September 1, 1992. The second (current) ten-year IST interval will end on August 31, 2002.

The second ten-year interval for the BFNP IST program for pumps complies with Section XI of the ASME Boiler and Pressure Vessel Code, 1986 Edition, as required by 10 CFR 50.55a(b)(2). All references to subsections IWP of Section XI in the procedure correspond to the 1986 Edition of ASME Section XI. Portions of later editions and addenda of ASME Section XI, which are incorporated by reference in 10 CFR 50.55a(b), may be used subject to limitations and modifications established by NRC, provided that all applicable requirements of the respective editions and addenda are met. BFN is converting the code of record for valve testing in the BFN IST Program from the 1986 edition of ASME Section XI to the 1989 edition of ASME Section XI. After the conversion is completed, the second ten-year interval for the BFNP IST program for valves will comply with Section XI of the ASME Boiler and Pressure Vessel Code, 1989 Edition. All references to subsection IWV of Section XI in the IST Program will then correspond to the 1989 Edition of ASME Section XI. This conversion is expected to be completed by August 1, 1997.

NRC Generic Letter 89-04 and NUREG-1482 provide alternatives to ASME Section XI IST requirements that are acceptable to NRC. In order to utilize the alternative methods, they must be complied with in their entirety and their use documented in the IST program.

TVA Nuclear Power Standard 8.6 (STD-8.6) is the TVA corporate level procedure which establishes IST program requirements for all TVA nuclear power plants. STD-8.6 requires establishment, implementation, and maintenance of IST in accordance with 10 CFR 50.55a(f)(4) and GL 89-04. Site Standard Practice (SSP) -8.6, ASME Section XI IST of Pumps and Valves, is the procedure that administratively controls the IST program at BFNP.

I. INTRODUCTION/POLICY STATEMENTS (Continued)

The BFNP IST program was prepared using the following for guidance:

- ASME Section XI, 1986 and 1989 Editions
- ASME/ANSI Operations and Maintenance (O&M) Standards, Parts 1, 6, and 10
- NRC GL 89-04
- NRC Regulatory Guide 1.26
- NUREG-0800
- NRC Temporary Instructions 2515/110 and 2515/114
- NRC Inspection Procedure (IP) 73756
- NRC GL 91-18
- NUREG-1482
- BFNP Updated Final Safety Analysis Report (UFSAR)
- BFNP Technical Specifications (TS)
- BFNP System Design Criteria

All plant components were reviewed for ASME Section XI IST applicability and those meeting the following criteria have been included in the BFNP IST program. Components that meet the criteria of IWP-1200 and IWV-1200 of ASME Section XI are excluded from the IST program.

1. Components classified as ASME Code Class 1, 2, or 3 equivalent that are required to perform a specific safety function. Classification is determined by TVA Nuclear Engineering and designated on the controlled Inservice Inspection (ISI) series of drawings.

2. Components which give nuclear safety-related overpressure protection to nuclear safety-related Code class equivalent systems, subsystems, and components.

3. Additional safety-related components that are not ASME Code Class 1, 2, or 3 equivalent but which have been determined to require augmented IST. These components are identified in the BFNP IST program listing as non-ASME Code Class.

Changes to the plant are controlled by administrative procedures (SSPs) that require a review for ASME Section XI applicability of all proposed design changes, procedure changes, temporary alterations, modifications, tests, or experiments performed under the 10 CFR 50.59 process. SSP-9.3, Plant Modifications and Design Change Control, is the SSP used to control changes to the plant configuration and design. SSP-2.3, Administration of Site Procedures, is the SSP used to control plant procedures. SSP-12.13, 10 CFR 50.59, Evaluations of Changes, Tests, and Experiments, is the SSP used to control changes in accordance with the 10 CFR 50.59 process. The BFNP ASME Section XI IST Program (and all implementing procedures) are changed as required to reflect approved changes in the plant configuration, processes, or procedures. These changes are implemented prior to return to operation of the affected systems or components.

I. INTRODUCTION/POLICY STATEMENTS (Continued)

B. Pump IST Program

The pump test program shall be conducted in accordance with Subsection IWP of Section XI of the ASME Boiler and Pressure Vessel Code except where relief has been requested under the provisions of 10 CFR 50.55a(f)(5)(iii). The pump test program may also utilize NUREG-1482 and Generic Letter 89-04 provided the applicable associated requirements are complied with and their use is documented in the IST program. Part II details the IST program for all safety-related pumps at Browns Ferry Nuclear Plant. Each parameter to be measured as well as specific notes concerning requests for relief are also listed.

C. Valve IST Program

BFN has begun converting the code of record for valve testing from the 1986 edition of ASME Section XI to the 1989 edition of ASME Section XI. This conversion is expected to be completed by August 1, 1997. The valve test program shall be conducted in accordance with Subsection IWV of Section XI of the applicable edition of the ASME Boiler and Pressure Vessel Code except where relief has been requested under the provisions of 10 CFR 50.55a(f)(5)(iii). The valve test program may also utilize NUREG-1482 and Generic Letter 89-04 provided the applicable associated requirements are complied with and their use is documented in the IST program. The valve test program is included in Part II.

1. Valves are categorized according to the following.

Category A - Valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their function. Valves for which seat leakage is important may generally be classified as pressure isolation valves (PIV), containment isolation valves (CIV), or both PIV/CIV. CIVs are tested in accordance with ASME/ANSI OM-10 requirements for category A valves. This deviation from ASME Section XI requirements is allowed by NUREG-1482 Section 4.4.5 provided the applicable requirements of OM-10 are met. These requirements include Paragraphs 4.2.2.1 through 4.2.2.3 of OM-10. The actual test methods shall be in accordance with the Browns Ferry Appendix J testing program. BFN has implemented Option B of 10 CFR 50 Appendix J for leak rate testing of CIVs. Since Option B is performance-based with regard to frequency, not all CIVs will be leak rate tested every refueling outage. CIVs that are also PIVs will continue to be leak rate tested every refueling outage since they have an additional safety function (pressure isolation).

Category B - Valves for which seat leakage in the closed position is inconsequential for fulfillment of their function.

Category C - Valves which are self-actuating in response to some system characteristic, such as pressure (safety and relief valves including vacuum relief valves) or flow direction (check valves).

Category D - Valves which are actuated by an energy source capable of only one operation, such as rupture disks or explosive-actuated valves.

Valves are tested in accordance with the requirements of their category. Valves which fall into more than one category are tested in accordance with the requirements of all applicable categories. Duplicaton or repetition of common testing requirements for valves that fall into more than one category is not performed.

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2, AND 3
AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) SECTION XI
ADOPTION OF 1989 EDITION OF CODE FOR
INSERVICE TESTING OF VALVES

REFUELING OUTAGE JUSTIFICATION RO-1 AND RO-2

(SEE ATTACHED)

REFUELING OUTAGE JUSTIFICATION RO-1

System: Core Spray (CS)(75)

Drawing: 1, 2, 3-47E814-1 (CS)

Components: Testable check valves 75-26, 75-54

Category: AC

Class: 1

Function: Valves open to allow em.ergency cooling water supply to the reactor.
Valves close to maintain primary containment isolation and prevent loss of reactor coolant.

Impractical Requirement: ASME Operations and Maintenance Standard (OM), Part 10, 1987 Edition through the OMa-1988 addenda paragraph 4.3.2.1 states "Check valves shall be exercised nominally every 3 months, except as provided by paras. 4.3.2.2, 4.3.2.3, 4.3.2.4, and 4.3.2.5". Paragraph 4.3.2.2(e) states "If exercising is not practicable during plant operation or cold shutdowns, it may be limited to full-stroke during refueling outages".

Specifically, the impractical requirement for testing of these valves is the requirement to exercise the valves quarterly and during cold shutdowns.

Basis for Deferral to Refueling Outage: These valves are located inside the drywell (primary containment) where the atmosphere is inerted with a nitrogen atmosphere during operation as required by Technical Specification 3.7.A.5.a. (and may remain inerted depending on the reason for going to CSD). The testable check valve actuators used to perform the stroke test on the valve only open the valve disc approximately 30° (full open is 75°) and cannot open the valves unless the pressure across the disc is equalized. Due to potentially inadvertent valve operation caused by non-class 1E circuitry to the valve operator, the air supply to each valve operator is normally disconnected. As a result the valves cannot be part-cycled quarterly (self-actuation is unaffected). Entry to the drywell to reconnect the air supply to these valves and equalize the pressure across the valve discs would be hazardous to personnel unless the unit is in CSD with the drywell atmosphere deinerted. It is therefore impractical to perform a partial stroke of these valves unless the unit is at cold shutdown with the drywell deinerted and plant conditions allow access to the valve operators.

NRC Generic Letter 89-04, Position 1, states that a check valve's full-stroke to the open position may be verified by passing the maximum required accident condition flow through the valve. A full flow test using Core Spray system pumps during operation would be impossible unless reactor pressure was reduced to less than 450 psig. If reactor pressure could be reduced to less than 450 psig, full flow testing would still be impractical due to the adverse effects of the additional cold water on vessel water level. The increase in water level from the flow test could result in carryover to the main steam lines, possibly damaging the turbines or causing a reactor scram. The addition of such a large quantity of cold water would also have an adverse effect on reactivity resulting in an undesirable power transient.

A full flow test using Core Spray system pumps during cold shutdowns would be impractical due to the effects on vessel water level. The reactor vessel head would still be installed and the main steam lines could be flooded without the main steam line plugs installed. This could delay return of the unit to power operation. The addition of such a large quantity of water could also affect water chemistry in the reactor vessel, delaying return to power operation until water chemistry is returned to acceptable parameters.

Basis for Deferral to Refueling Outage (continued):

ASME/ANSI OM-10 allows check valves to be disassembled every refueling outage to verify full stroke capability. Based on past work performed on these valves and radiological conditions in the area, a typical dose received by a TVA employee involved in performing disassembly of these valves would be 540 mR. The dose received by a TVA employee when performing the partial stroke test using the permanently installed valve operator would be around 45 mR. Since the partial stroke te[illegible] has to be performed after the valve disassembly and reassembly to prove operability, the 540 mR figure represents the total additional dose per person received in performing the disassembly. The total additional dose for the disassembly would therefore be 1.6 man-Rem per disassembly (based on three personnel at the valve). Because most cold shutdowns are relatively short in duration, there would usually not be sufficient time to perform a disassembly of these valves without impacting return of the unit to power operation. For these reasons, disassembly is not a practical method of verifying full-stroke capability for these valves during cold shutdowns.

Full-stroking of these valves by removing the actuator, stroking the valve disc full open using a torque wrench, reinstalling the actuator, resetting the limit switches, and performing the partial-stroke test as verification of operability would involve an additional dose received by TVA personnel performing the manual full-stroke (approximately 60 percent of the dose received during disassembly, or 324 mR total dose per valve per full stroke). Because of the amount of work and dose involved, a full stroke of these valves using a torque wrench cannot be practically performed during cold shutdowns. This partial disassembly could also ca[illegible] roblems since it would involve the position indicating and actuator electrical and pneumatic control lines for the valve. The magnetic position indicating switches are particularly difficult to set and could delay startup from cold shutdown.The only time when sufficient time would exist to perform a full-stroke of these valves would be during a refueling outage.

Alternate Testing:

These valves will be partial-stroked in accordance with cold shutdown testing guidelines described in NUREG-1482 when the drywell atmosphere is deinerted and personnel can safely perform the work without exceeding ALARA guidelines. This will be done using either the permanently installed air operators mounted on the valves or by use of a wrench or other mechanical device with the actuator still attached to the valve shaft.

As a minimum, both valves on each unit will be full-stroked open and closed during each refueling outage. This will be done by one of the three following methods:

(1) First, a Core Spray injection to the open reactor vessel may be performed if allowed by the existing plant conditions. The main steam line plugs will be installed during refueling outages and flooding of the steam lines will not be a concern, particularly if the flow test is performed during reactor vessel floodup prior to fuel movement.
(2) Second, the permanent valve operators may be removed and an adapter attached for cycling the valve disc with a torque wrench. If this option is used, the torque required to lift the disc off of the valve seat will be measured and recorded as required by ASME/ANSI OM-10. A partial stroke of the valve disc will be performed after reattachment of the valve actuator.
(3) Third, the valve disc may be manually stroked full-open by hand if the valve bonnet has been removed for maintenance activities. If this third option is used there will not be a measurement of torque required to lift the disc off the seat. (Reference NRC NUREG-1482 Page A-9). A partial stroke of the valve disc will be performed after reassembly of the valve.

Any one of these three methods will prove free movement of the valve disc to the full open position and back to full closed. All other testing and corrective action requirements of ASME/ANSI OM-10 will be followed for these valves.

REFUELING OUTAGE JUSTIFICATION RO-2

System: Residual Heat Removal (RHR)(74)

Drawing: 1, 2, 3-47E811-1 (RHR)

Components: Testable check valves 74-54, 74-68

Category: AC

Class: 1

Function: Valves open to allow emergency/shutdown cooling water supply to the reactor.
Valves close to maintain primary containment isolation and prevent loss of reactor coolant.

Impractical Requirement: ASME Operations and Maintenance Standard (OM), Part 10, 1987 Edition through the OMa-1988 addenda paragraph 4.3.2.1 states "Check valves shall be exercised nominally every 3 months, except as provided by paras. 4.3.2.2, 4.3.2.3, 4.3.2.4, and 4.3.2.5". Paragraph 4.3.2.2(e) states "If exercising is not practicable during plant operation or cold shutdowns, it may be limited to full-stroke during refueling outages".

Specifically, the impractical requirement for testing of these valves is the requirement to exercise the valves quarterly and during cold shutdowns.

Basis for Deferral to Refueling Outage: These valves are located inside the drywell (primary containment) where the atmosphere is inerted during operation as required by Technical Specification 3.7.A.5.a. (and may remain inerted during CSD depending on the reason for going to CSD). Due to potentially inadvertent valve operation caused by non-class 1E circuitry to the valve operator, the air supply to each valve operator is normally disconnected and the valves cannot be cycled quarterly (self-actuation is unaffected). These valves cannot be opened using the valve operators unless pressure across the valve disc is equalized. Entry to the drywell to reconnect the air supply to these valves and equalize the pressure across the valve disc would be hazardous to personnel unless the unit is in CSD with the drywell atmosphere de-inerted.

Full flow testing of these valves using the RHR system pumps and the LPCI flow path during operation is not possible unless reactor pressure is less than 450 psig. If reactor pressure could be dropped below 450 psig, full flow testing during operation would still be impractical due to the adverse effects on reactor water chemistry and core reactivity. The injection of the cold water during power operation would cause an undesirable power transient. Reactor vessel water level control would also be adversely affected, possibly resulting in flooding of the main steam lines. This could delay return of the unit to power operation until the lines could be drained. Full flow testing using the shutdown cooling flow path would not be possible unless reactor vessel pressure was below 105 psig. The adverse effects on reactor water chemistry and core reactivity experienced during LPCI injection would also occur when using shutdown cooling flow (during power operation).

Basis for Deferral to Refueling Outage (continued):	Full flow testing of these valves is possible during CSD using the shutdown cooling mode of RHR. However, it may not be possible to cycle both valves during the same shutdown. Most cold shutdowns are necessitated by the need to perform maintenance and are of relatively short duration. Shutdown cooling is needed to remove residual heat loads after entering cold shutdown and is usually operated continuously until the unit is ready to return to power operation. In order to swap loops of RHR to inject through the other testable check valve, plant personnel would have to flush the other loop of RHR until chemistry requirements for water are met and then realign the system. This activity would require significant use of plant resources, increase the operator workload during a period of high activity, and possibly delay returning the unit to power operation.
Alternate Testing:	A minimum of one of the testable check valves will be cycled during each CSD (provided three months has passed since the previous CSD). Also, both testable check valves will be cycled during each refueling outage. This will normally be done by verifying the valve opens using normal shutdown cooling flow (9000 GPM minimum) and closes on cessation of shutdown cooling flow. Closure of the testable check valves may also be demonstrated before shutdown cooling is initiated or after shutdown cooling is terminated by other positive means. If Technical Specifications and plant conditions allow, the opposite testable check valve will be cycled using shutdown cooling flow during the same CSD. If both valves cannot be cycled during the same CSD, then the uncycled valve will be cycled first during the next CSD (provided Technical Specifications and plant conditions allow and three months or more has passed since the previous stroke of the valve in question). If entry to the drywell is possible, the valves may be cycled by connecting temporary air to the valve operators instead of by passing flow through the shutdown cooling flow path. In this case, the valves would then be stroked full open and full closed using the control room handswitch.