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

August 11, 2000

TVA-BFN-TS-400 10 CFR 50.90

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

Gentlemen:

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

BROWNS FERRY NUCLEAR PLANT (BFN) - UNITS 2 AND 3 - TECHNICAL SPECIFICATIONS CHANGE (TS) NO. 400 - REQUEST FOR LICENSE AMENDMENT - RELAXATION OF EXCESS FLOW CHECK VALVE (EFCV) SURVEILLANCE TESTING FREQUENCY

In accordance with the provisions of 50.90, TVA is submitting a request for an amendment (TS-400) to licenses DPR-52 and DPR-68 to change the TS for Units 2 and 3. The proposed change will relax the Surveillance Requirement (SR) frequency by allowing a representative sample (approximately 20 percent) of EFCV to be tested every 24 months, such that each EFCV is tested once every 120 months.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate marked-up TS pages from Units 2 and 3 showing the proposed changes.

The basis for this request is consistent with the General Electric NEDO-32977-A (Boiling Water Reactor Owner's Group (BWROG) Topical Report, B21-00658-01), Excess Flow Check Valve Testing Relaxation, dated June 2000. This report was

D030

Printed on recycled paper

initially submitted to the NRC as part of the Duane Arnold Energy Center proposed TS amendment on April 12, 1999. The BWROG report was supplemented by the January 6, 2000, letter, BWR Owners Group Generic Response to NRC Request For Additional Information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements. The report was approved for use by NRC SER dated March 14, 2000. Additionally, issues raised by NRC in the March 14, 2000, SER were addressed in the June 2000, issue of the topical report. Technical Specification Task Force (TSTF) Generic Traveler Number 34 was previously submitted to the NRC for approval.

TVA issued a plant specific radiological analysis to assess the effects of this proposed change in terms of off-site and main control room doses. The analysis shows the dose contribution from the proposed change is acceptable when compared to dose limits prescribed in 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19. Also, a plant specific analysis was performed to estimate steam release frequency into the secondary containment (Reactor Building area). TVA's evaluation has determined that the proposed change in test frequency has a minimal effect on EFCV reliability. The results of the dose calculation and estimated release frequency are provided in Enclosure 1.

To assist the staff in the review of the change, TVA has included a proposed revision to the TS Bases. These bases pages are for information only and are not considered part of this application for a license amendment.

Precedent exists for the requested TS change, Duane Arnold and Fermi 2 have previously provided justifications for relaxation of the SR frequency for testing EFCVs utilizing BWROG Report, B21-00658-01 as a basis for the change. These applications were approved by NRC.

The BFN Plant Operations Review Committee and the BFN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of BFN Units 2 and 3 in accordance with the proposed change will not endanger the health and safety of the public. TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt 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.

TVA plans to implement this change on both Units 2 and 3 by each respective unit's next scheduled refueling outage. Therefore, TVA requests that NRC approve the proposed change by February 1, 2001, to allow implementation of the change prior to the Unit 2 Cycle 11 refueling outage scheduled for Spring of 2001.

There are no commitments contained in this letter. If you have any questions about this change, please telephone me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

cc: See Page 5

Subscribed and sworn to before me
on this 11th day of August 2000.

Barbara A. Slanton
Notary Public

My commission Expires 09/22/2002

Enclosures
cc (Enclosures):

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

Chairman
Limestone County Commission
310 West Washington Street
Athens, Alabama 35611

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

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

State Health Officer
Alabama Department of Public Health
434 Monroe Street
Montgomery, Alabama 36130-1701

ENCLOSURE 1
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 2 AND 3

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGE TS-400
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I.	DESCRIPTION OF THE PROPOSED CHANGE	E1-2
II.	REASON FOR THE PROPOSED CHANGE	E1-3
III.	SAFETY ANALYSIS A. Radiological Dose Assessment B. Excess Flow Check Valve Failure Rate C. Release Frequency D. Conclusions	E1-3
IV.	NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION	E1-8
V.	ENVIRONMENTAL IMPACT CONSIDERATION	E1-9

I. DESCRIPTION OF THE PROPOSED CHANGE

TVA is requesting a change to the Units 2 and 3 TS that will relax the testing frequency of the excess flow check valves (EFCV) by allowing a representative sample to be tested every 24 months such that each EFCV is tested at least once every 120 months. The current specification requires that each reactor instrumentation line EFCV be actuated to the isolation position on a simulated instrument line break every 24 months.

The proposed change will verify a representative sample of the reactor instrumentation line EFCVs actuate to the isolation position during a simulated instrument line break signal every 24 months. TVA proposes that a representative sample of the EFCVs (approximately 20%) are tested each refueling outage; thus, each valve is tested every 120 months (nominal).

The proposed change to the TS is listed below. Enclosure 2 contains copies of the appropriate marked-up TS pages for Units 2 and 3 showing the changes. The change is the same for both Units 2 and 3. In addition, Enclosure 2 provides the associated markup TS Basis pages. These are provided to the NRC for information only.

Units 2 and 3, page 3.3-16, Primary Containment Isolation Valves, Surveillance Requirements, SR 3.6.1.3.8 will be revised by adding requirements that will allow testing of a representative sample of the EFCVs on a 24 month interval. The deleted text is shown with ~~strikethrough~~, and changed or added text is shown in ***bold italics***. The revised SR will read as follows:

SURVEILLANCE REQUIREMENTS continued

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.8	Verify ~~each~~ ***a representative sample of*** reactor instrumentation line EFCV***s*** actuates to the isolation position on a simulated instrument line break signal.	24 months

II. REASON FOR THE PROPOSED CHANGE

TVA is requesting a change to the Unit 2 and 3 TS that will reduce the number of EFCVs tested as part of refueling outage activities. Instrument line EFCVs which connect to the reactor coolant pressure boundary are normally tested during the reactor pressure vessel system leakage test, which is performed as part of refueling outage activities. The EFCVs are designed to close upon a failure of a instrument line downstream of the valve. Testing of the EFCVS typically requires that the reactor be pressurized to normal operating pressure, opening an instrument drain valve and observing valve closure. Testing has historically required approximately 140 man-hours to complete and can be a critical path item during refueling outages. This places an undue burden on the TVA staff without a commensurate increase in plant safety.

Reducing the number of valves tested as with any reduction in maintenance, inherently reduces the risk of industrial and occupational hazards, including inadvertent exposure to radioactive fluids. Furthermore, there is a consequential reduction in radioactive waste generated during testing activities.

III. SAFETY ANALYSIS

The safety objective of the Primary Containment Isolation System is to provide the capability, in the event of the postulated loss-of-coolant accident, to limit the release of fission products to the plant environs so that offsite doses would be within the limits of 10 CFR 100 and 10 CFR 50, Appendix A, GDC 19. Isolation of all pipes or ducts which penetrate the primary containment is required to maintain leakage within permissible limits.

An analysis was performed to determine the feasibility of relaxing the current TS Surveillance Requirement to test each reactor instrumentation line EFCV every 24 months. The proposed change would allow a representative sample of EFCVs to be tested each 24 months. In that way, each EFCV would be tested at least once every 120 months (nominal). This test program is consistent with that described in General Electric NEDO-32977-A (Boiling Water Reactor Owners Group (BWROG) Topical Report B21-00658-01), Excess Flow Check Valve Testing Relaxation, dated November, 1998, (revised through June, 2000). This report was approved by the staff on March 14, 2000.

Periodic testing of a representative sample of EFCVs selected on a performance basis will continue to ensure the reliability of these valves. This, along with the plant design, assures that the assumed release rate in the plant's safety analysis remains conservative.

BWROG Topical Report B21-00658-01 was reviewed, along with the licensing requirements, operational experience, and consequences associated with the testing requirements for EFCVs in instrument lines connecting to the reactor coolant pressure boundary. The report concluded that the change in the test frequency had insignificant impact on valve reliability. The BWROG report also concluded that the demonstrated reliability of EFCVs coupled with low consequences of EFCV failure provided adequate justification for extending the test interval up to once every 120 months.

Failure of an EFCV to close does not involve a significant increase in the probability or consequences of an accident previously evaluated. Instrumentation piping connected to the reactor primary system which leaves the primary containment is dead ended at instrument racks located in the Reactor Building. These instrument lines are provided with a manual block valve and an EFCV both of which are located outside primary containment. Except for the jet pump sensing lines, a one fourth(¼)-inch orifice has been installed in instrument lines that penetrate the primary containment boundary into the secondary containment (Reactor Building area). Sense lines for jet pump flow within the reactor vessel, including the reactor vessel penetration, are constructed of one fourth(¼)-inch sensing pipe. The one fourth(¼)-inch sense line, effectively provides the same flow area as the one fourth(¼)-inch orifice in the other instrument lines. This design limits the release of reactor coolant in the event of an instrument line break outside primary containment.

The leakage from a postulated broken instrument line outside containment is reduced by design to the maximum extent practical, consistent with instrument response requirements. The rate and extent of coolant loss is well within the capability of reactor coolant make-up systems. The integrity and functional performance of secondary containment and the Standby Gas Treatment System will be maintained. The potential exposure will be substantially below the limits of 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19. A break in the portion of an instrument line between the containment and EFCV located outside primary containment and the direct blowdown to the reactor building

was considered. It was concluded in the June 26, 1972, Safety Evaluation For The Tennessee valley Authority Browns Ferry Nuclear Plant Units 1, 2, and 3, that the isolation provisions for instrument lines penetrating primary containment were adequately designed and meets the intent of NRC Safety Guide 11, Instrument Lines Penetrating Primary Containment Backfitting Considerations (Regulatory Guide 1.11, dated February 17, 1972).

The effect of extending the EFCV test interval is a corresponding increase in the potential for a release. However, even with a 120 month test interval, the release frequency from an individual line continues to remain very low. Also, since the EFCVs are located in secondary containment (Reactor Building), any release from a failed EFCV would be treated by the Standby Gas Treatment System providing additional mitigation of any postulated offsite release from a broken instrument line.

A. RADIOLOGICAL DOSE ASSESSMENT

TVA performed a plant specific dose assessment to determine the off-site and control room dose due to an instrument line break outside of the drywell. The reactor was assumed to be operating at normal pressure and temperature. A manual reactor scram is assumed to occur ten minutes following the line break. The reactor coolant was at equilibrium Technical Specification limits for operation of 3.2 μCi/gm I-131 equivalent. As a consequence of the reactor scram and subsequent depressurization an iodine spike of 500 times the equilibrium release rate was incorporated in the model.

The computer code STP was used to determine the time dependent releases. The computer code COROD was used to determine the control room operator doses and computer code FENCDOSE was used to determine the offsite doses . These computer codes are part of the BFN licensing basis as described in Chapter 14 of the FSAR. The release rate and pressure of the water were derived from GE NEDO-21143-1, Radiological Accident Evaluation - The CONAC03 Code, dated December 1981. GE NEDO-21143-1 assumes Operator action occurs at ten minutes. The Standby Gas Treatment system is initiated followed by a plant shutdown. The results of the assessment are shown on the next page.

The doses are in rem.

Dose Assessment Resulting From EFCV Failure			
	Control Room	2-Hour Exclusion Boundary	30-Day Low Population Zone
Thyroid	9.050E-02	2.255E-01	1.642E-01
Gamma	5.924E-06	3.078E-04	6.059E-04
Beta	4.409E-05	1.067E-04	2.604E-04

As shown in the table above, the control room doses due to an instrument line break will be less than the 10 CFR 50 Appendix A, GDC 19 limits. The off-site doses will be less than 10 percent of the 10 CFR 100 limits.

B. EFCV FAILURE RATE

The reliability of EFCVs were evaluated based on testing experience provided by 12 different BWR plants. The composite data indicated that EFCVs are very reliable. It can be noted that the data shown in the BWROG report documents that BFN experienced 21 EFCV failures during the Unit 2 restart effort and 5 EFCV failures during the Unit 3 restart effort. The testing followed a 6 and 10 year extended outage respectfully. TVA attributed this high failure rate to crud build up and valve sticking, test methodology, lack of experience of the test personnel, and one broken valve spring.

Since initial restart of Units 2 and 3 in 1991 and 1995 respectfully, TVA has experienced only 3 EFCV failures. TVA reviewed the surveillance test results from the last seven refueling outages (four outages on Unit 2 and three outages on Unit 3. The results of this review are provided in the tables below:

BFN EFCV Failures				
	Cycle 7	Cycle 8	Cycle 9	Cycle 10
Unit 2	1	1	0	0
Unit 3	1	0	0	NA

Additionally, TVA calculated a plant specific failure rate from the last seven refueling outages and a 24 month fuel cycle. This is shown below on the next page:

BFN EFCV Failures Rate				
Operating time (years)	Operating Time (hours)[1]	Number of Failures	Estimate Failure Rate/hour	Upper Limit Failure Rate/Hour
703.5	6.16E+06	3	4.87E-07	1.26E-06
703.5	6.16E+06	15	2.43E-06	3.70E-06[2]

As shown in the tables, BFN has experienced 3 EFCV failures in the previous 6.16E+06 hours of reactor operation. The calculated upper limit value for BFN for a normal 18 month surveillance interval considering three failures is 1.26E-06 failures per hour which is approximately two times the generic value of 6.30E-07 failures per hour. For a ten year interval, TVA calculated an upper limit value of 3.70E-06 EFCV failures per hour approximately six (6) times the generic value. Although the failure rates are greater than the generic value, TVA considers this a low value.

Any future EFCV failure would be evaluated per the BFN corrective action program. Additionally, as part of implementation of this TS amendment, the 10 CFR 50.65 Maintenance Rule Program, will be revised to include a specific EFCV performance acceptance criteria.

C. Release Frequency

The release frequency is the product of the instrument line break frequency and EFCV failure rates. An instrument line break frequency of 3.52E-05 per year has been calculated for BFN. The EFCV failure rate was derived based on the assumption that there will be a five fold increase in failures at BFN. TVA found that for a two year surveillance interval (24 month fuel cycle) the release frequency would be approximately 7.64E-05 events per year and for a ten year surveillance interval the release frequency would be 3.82E-04 events per year.

The specific values for the two and ten year surveillance interval are sufficiently low that it can be concluded that a change in surveillance test frequency has minimal impact on EFCV reliability.

---

[1] Operating time is calculated as follows: 67 valves * 7 cycles * 1.5 years per cycle = 703.5 years. 703.5 years * 8760 hours per year = 6.16E+06 hours.

[2] Failure rate for the ten year interval is obtained using a chi-square distribution considering 5 times the normal failure rate in the ten year period, 15 failures.

The release frequency, considering a ten year surveillance frequency BFN, is equivalent to approximately one event every 2,600 years.

D. **Conclusions**

Implementation of this change represents a increase in release frequency of approximately 3.06E-04 events per year from the current release frequency estimate of 7.64E-05 events per year for two year surveillance interval. This increase is not significant, especially since any postulated coolant leakage is within the capability of the reactor coolant makeup systems, and the consequences of such an accident are not expected to lead to a core damage event.

The radiological consequences from an instrument line break were found to be a very small portion of the 10 CFR 100 and 10 CFR 50 Appendix A, GDC 19 limits. In the unlikely event that an instrument line breaks and the EFCV fails to close, core damage would not be expected to occur and the doses would be less than regulatory limits.

Therefore, it has been concluded, considering the low consequences of a release, the extension of the test interval does not significantly affect the risk to the public associated with the failure of an instrument line and the failure of an EFCV to perform its intended function.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA is requesting a change to the Units 2 and 3 TS that will relax the testing frequency of the excess flow check valves (EFCV) by allowing a representative sample to be tested every 24 months such that each EFCV is tested at least once every 120 months. The current specification requires that each reactor instrumentation line EFCV be actuated to the isolation position on a simulated instrument line break every 24 months.

TVA has concluded that operation of BFN Units 2 and 3 in accordance with the proposed change to the TS does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The current excess flow check valve (EFCV) frequency requires that each reactor instrument line EFCV be tested every 24 months. The EFCVs are designed to automatically close upon excessive differential pressure including failure of the down stream piping or instrument and will reopen when appropriate. This proposed change will allow a reduction in the number of EFCVs that are verified tested every 24 months, to approximately 20 percent of the valves each cycle. BFN and industry operating experience demonstrates high reliability of these valves. Neither the EFCVs or their failure is capable of initiating a previously evaluated accident. Therefore, there is no increase in the probability of occurrence of an accident previously evaluated.

The instrument lines going to the Reactor Coolant Pressure boundary with EFCVs installed have flow restricting devices upstream of the EFCV. The consequences of a unisolable failure of an instrument line has been previously evaluated and meets the intent of NRC Safety Guide 11. The offsite exposure has been calculated to be substantially below the limits of 10 CFR 100. Additionally, coolant lost from such a break is inconsequential compared to the makeup capabilities of normal and emergency makeup systems. Although not expected to occur as a result of this change, the affects of a postulated failure of an EFCV to isolate and instrument line break as a result of reduced testing are bounded by TVA analysis.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

This proposed change reduces the number of EFCVs tested each operating cycle. No other changes to the TS are being proposed. BFN and industry operating experience demonstrates that these valves are highly reliable, a proposed reduction in test frequency is bounded by previous evaluation of a line rupture. The change will

not alter the operation of process variables, structures, systems or components described in the BFN Updated Final Safety Analysis Report. Therefore, reduction in the number of EFCVS tested each cycle does not result in the possibility of a new or different kind of accident.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

The consequences of an unisolable rupture of an instrument line has been previously evaluated and meets the intent NRC Safety Guide 11. The proposed amendment does not involve a significant reduction in a margin of safety.

Therefore, the proposed revised surveillance frequency does not adversely affect the public health and safety, and does not involve any significant safety hazards.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

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

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-400
MARKED PAGES

I. AFFECTED PAGE LIST

Technical Specifications	
Unit 2	Unit 3
3.6-16	3.6-16
Bases	
3.3-34	3.3-34
3.3-36	3.3-36

II. MARKED PAGES

See attached.

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.	24 months
SR 3.6.1.3.8	Verify ~~each~~ ***a representative sample of*** reactor instrumentation line EFCV***s*** actuate~~s~~ to the isolation position on a simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 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 maximum pathway 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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience with these components supports performance of the Surveillance at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that ~~each~~ ***a representative sample of*** reactor instrumentation line excess flow check valve***s*** (EFCV***s***) ~~is~~ ***are*** OPERABLE by verifying that the valve***s*** actuate~~s~~ to the isolation position on an actual or simulated instrument line break signal. This SR provides assurance that the instrumentation line EFCVs will perform so that the radiological consequences will not exceed the predicted radiological consequences during events evaluated in Reference 5. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. ~~Operating experience with these components supports performance of the Surveillance at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ **Insert**

(continued)

Insert

The representative sample consist of an approximately equal number of EFCVs tested each 24 months, such that each EFCV is tested at least once every 120 months (nominal). The nominal 120 month interval is based on other performance-based testing programs such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure the overall reliability is maintained. Operating experience has demonstrated that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 8).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

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.
8. ***General Electric NEDO-32977-A (Boiling Water Reactor Owners Group Topical Report, B21-00658-01),"Excess Flow Check Valve Testing Relaxation", dated June 2000.***

(continued)

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.	24 months
SR 3.6.1.3.8	Verify ~~each~~ ***a representative sample of*** reactor instrumentation line EFCV***s*** actuate~~s~~ to the isolation position on a simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 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 maximum pathway 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

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.7

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.1 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience with these components supports performance of the Surveillance at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

This SR requires a demonstration that ~~each~~ ***a representative sample of*** reactor instrumentation line excess flow check valve***s*** (EFCV***s***) ~~is~~ ***are*** OPERABLE by verifying that the valve***s*** actuate~~s~~ to the isolation position on an actual or simulated instrument line break signal. This SR provides assurance that the instrumentation line EFCVs will perform so that the radiological consequences will not exceed the predicted radiological consequences during events evaluated in Reference 5. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. ~~Operating experience with these components supports performance of the Surveillance at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint~~. **Insert**

(continued)

Insert

The representative sample consist of an approximately equal number of EFCVs tested each 24 months, such that each EFCV is tested at least once every 120 months (nominal). The nominal 120 month interval is based on other performance-based testing programs such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure the overall reliability is maintained. Operating experience has demonstrated that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 8).

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

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.
8. ***General Electric NEDO-32977-A (Boiling Water Reactor Owners Group Topical report, B21-00658-01), "Excess Flow Check Valve Testing Relaxation", dated June 2000.***

(continued)