ENERGY NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

October 30, 2000
GO2-00-185

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT
SR 3.6.1.3.8 EXCESS FLOW CHECK VALVES TEST FREQUENCY

References:
1. Letter BWROG-00069, dated June 14, 2000, from WG Warren, (BWR Owners Group) to Office of Nuclear Reactor Regulation, "Transmittal of Approved GE Licensing Topical Report, NEDO-32977-A, Excess Flow Check Valve Testing Relaxation," dated November 1998
2. Letter GO2-00-088, dated May 11, 2000, from AE Mouncer (Energy Northwest) to U.S. Nuclear Regulatory Commission, "Request for Amendment Technical Specification Surveillance Requirement SR 3.6.1.3.8 Excess Flow Check Valves Test Frequency"
3. Letter, dated October 3, 2000, JS Cushing (NRC) to JV Parrish (Energy Northwest) "Request for Additional Information (RAI) for WNP-2, (TAC NO. MA9063)"
4. Letter GO2-00-177, dated October 12, 2000, PJ Inserra (Energy Northwest) to U.S. Nuclear Regulatory Commission, "Request for Amendment To Technical Specification Surveillance Requirement SR 3.6.1.3.8 (Additional Information)"

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101, 50.59 and 50.90, Energy Northwest hereby submits a request for amendment to the WNP-2 Operating License. This letter is in accord with our response (Reference 4) to the NRC request for additional information (Reference 3) and revises our initial amendment request (Reference 2) in its entirety. Specifically, we are requesting a revision to Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.8. In addition, a revision to the TS Bases has been initiated pursuant to the Bases Control Program of TS 5.5.10 and 10CFR 50.36(a) and is included to assist the Staff in its review of the proposed TS change. These changes are included for information only and are not considered part of this application for license amendment. This proposed change is similar to requests for Technical Specification changes granted to Duane Arnold and Fermi-2.

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Surveillance Requirement 3.6.1.3.8 currently requires verification of the actuation capability of each Excess Flow Check Valve (EFCV) every 24-months. This proposed change is to relax the SR frequency by allowing a "representative sample" of reactor instrument line EFCVs to be tested every 24-months, such that each reactor instrument line EFCV will be tested at least once every 10-years (nominal). The proposed change is similar to existing performance-based testing programs, such as Inservice Testing for snubbers and Option B to 10CFR50 Appendix J.

The proposed change will also result in limiting the surveillance requirement to only the reactor instrument line EFCVs. The current requirement specified in the current SR 3.6.1.3.8 includes testing containment atmosphere and suppression pool instrument line EFCVs goes beyond the NRC Policy Statement Technical Specification screening criteria. The containment atmosphere and suppression pool instrument line EFCVs were conservatively included when implementing the Improved Standard Technical Specifications. The containment atmosphere and suppression pool instrument line EFCV testing is proposed to be relocated to the Final Safety Analysis Report (FSAR).

The basis for this amendment is consistent with that described in the Boiling Water Reactor Owners' Group (BWROG) GE Licensing Topical Report, NEDO-32977-A, and found acceptable by the NRC for referencing in relaxation of EFCV surveillance testing, subject to certain conditions. This amendment request is tailored to address the plant specific information and conditions required in the NRC Safety Evaluation provided in Reference 1. In accordance with 10CFR50.55a an Inservice Testing (IST) program relief request will be separately submitted.

Additional information has been attached to this letter to complete the amendment request.

- Attachment 1 describes the background of the issue and provides an evaluation of the proposed changes including a review of the BWROG responses related to generic questions raised by the NRC in Reference 1.

- Attachment 2 contains an evaluation of the change in accordance with 10CFR50.92(c) and concludes they do not result in a significant hazards consideration.

- Attachment 3 provides the Environmental Assessment Applicability Review and notes that the proposed change meets the eligibility criteria for a categorical exclusion as set forth in 10CFR51.22(c)(9). Therefore, in accordance with 10CFR51.22(b), an environmental assessment of the change is not required.

- Attachment 4 provides marked up pages of the TS and Bases revision.

- Attachment 5 consists of the typed TS pages as proposed by this amendment.

The attached proposed change to the Technical Specification has been approved by the WNP-2 Plant Operations Committee and reviewed by the Energy Northwest Corporate Nuclear Safety Review Board. In accordance with 10CFR50.91, the State of Washington has been provided a copy of this letter.

Energy Northwest requests that the NRC approves and issues the TS amendment by January 31, 2001, with a 30-day implementation time. The proposed amendment is needed to minimize personnel radiation exposure during the next WNP-2 refueling outage.

Should you have any questions or desire additional information regarding this matter, please contact PJ Inserra or me at (509) 377-4147.

Respectfully,

DK Atkinson
(Acting) Vice President, Operations Support/PIO
Mail Drop PE08

Attachments: as noted

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Senior Resident Inspector - 988C
DJ Ross - EFSEC
TC Poindexter - Winston & Strawn
DL Williams - BPA/1399

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Request for Amendment
Technical Specification SR 3.6.1.3.8
Excess Flow Check Valve Test
Frequency

I, DK Atkinson, being duly sworn, subscribe to and say that I am the (Acting) Vice President, Operations Support/PIO, for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief that the statements made in it are true.

DATE October 30, 2000

D.K. Atkinson
DK Atkinson
(Acting) Vice President, Operations Support/PIO

On this date personally appeared before me DK Atkinson, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 30 day of October 2000

Lori A. Mij
Notary Public in and for the
STATE OF WASHINGTON

Residing at Benton County

My Commission expires 3-29-01

Summary of Proposed Technical Specification Change

Background

WNP-2 Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.8 currently requires demonstration that each Excess Flow Check Valve (EFCV) is OPERABLE by verifying the valve actuates to the isolation position on an actual or simulated isolation signal every 24-months. The 24-month frequency is based on the typical performance of this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance was performed with the reactor at power. Since testing requires the reactor to be pressurized, this SR is normally performed during the reactor pressure vessel system leakage test, which is performed during refueling outages.

WNP-2 has EFCVs located in reactor instrument lines and in selected instrument lines that sample containment atmosphere and suppression pool level. Opening a downstream test drain valve from each EFCV with the reactor pressurized and verifying proper operation tests all reactor instrument line EFCVs Pressurizing through a test connection and verifying proper operation tests each containment atmosphere and suppression pool instrument line EFCV.

All instrument lines that penetrate primary containment conform to Regulatory Guide 1.11. The lines that connect to the reactor pressure boundary include a restricting orifice inside primary containment, are Seismic Category I, and terminate in instruments that are Seismic Category I. The instrument lines that sense containment atmosphere and suppression pool level are Seismic Category I and terminate at instrument racks that are Seismic Category I. The reactor, containment atmosphere and suppression pool instrument lines include manual isolation valves and EFCVs located outside containment.

Reactor Instrument Line EFCV

The proposed change is to relax the SR by allowing a representative sample of the reactor instrument line EFCVs to be tested every 24-months, such that each EFCV will be tested at least once every ten years. The proposed change is being requested to minimize radiation exposure during refueling outages, cut down on outage critical path time, and increase the availability of instrumentation during outages without significantly impacting the risk to the general public.

The Boiling Water Reactor Owners' Group (BWROG) has issued a report that provides a basis for this request. The report (NEDO-32977-A) provides justification for the relaxation of the SR frequency as described above. The report demonstrates the high degree of reactor instrument line EFCV reliability and the low consequences of a reactor instrument line EFCV failure. A similar TS change has been approved for the Duane Arnold Energy Center (Docket No. 50-331) on December 29, 1999, and for Fermi-2 (Docket No. 50-341) on March 14, 2000. The TS change request for WNP-2 also includes the relocation of the testing requirements for the containment atmosphere and suppression pool EFCVs to the Final Safety

Analysis Report (FSAR). This aspect of change was not part of the above NRC approvals and will be discussed below.

Basis for the Proposed Technical Specification Change

Reactor Instrument Line EFCV SR Frequency Change

Reliability data shown in the BWROG report documents 2 reactor instrument line EFCV failures at WNP-2 and indicates that the failures were related to configuration and testing methodology (See Testing Data Table). As stated in the "Maintenance Information" table on page 26 of the BWROG report, the test methodology and design basis were being reviewed at the time of the issuance of that report. The result of that additional investigation and testing methodology adequacy verification revealed that in one (1) case the valve actually functioned properly as designed, but the simulated test conditions did not account for the system design which prevented sustained flow to maintain the valve shut. Therefore, this case is no longer considered a failure.

WNP-2 reactor instrument line failure rate values have been updated using 15 years of plant service life, since additional testing was performed during Refueling Outage R-14 which took place after issuance of the initial BWROG report. Based on 15 years and one (1) failure the WNP-2 Best Estimate Failure Rate is revised from 1.69E-7 to 7.9E-8 per hour, which is less than the industry average of 1.01E-7 per hour. Additionally, the BWROG calculates the release frequency initiated by an instrument line break for a plant with 94 instrument lines with a 24-month surveillance interval, while WNP-2 has 96 instrument line EFCVs. We have determined that the BWROG conclusion that releases would be infrequent remains applicable to WNP-2.

The postulated break of an instrument line attached to the reactor coolant boundary is discussed and evaluated in the FSAR, Subsection 15.6.2. The evaluation assumes that the break occurs at a point where the reactor instrument line EFCV would not be able to isolate the break. The line size and the restricting orifice in the line minimize leakage from a break postulated to occur upstream of the EFCV. The integrity and functional performance of the secondary containment and standby gas treatment system are not impaired by this event, and the calculated potential offsite exposures are substantially below the requirements of 10CFR100. Therefore, a failure of a reactor instrument line EFCV, though not expected as a result of this TS change, is bounded by the previous evaluation of an instrument line break. The radiation dose consequences of such a break are not impacted by this proposed change.

At WNP-2, The EFCVs in the reactor instrument lines are identical models, with similar configurations. Operating environments are similar, except for those valves located in the steam tunnel. The temperature and radiation exposure environment in the steam tunnel is greater than that of the reactor building. However, surveillance testing has demonstrated that the steam tunnel environment has not resulted in an adverse failure history. Energy Northwest (WNP-2) has not experienced any failure of steam tunnel EFCVs during surveillance testing.

Consequently, the EFCVs in the steam tunnel are included with those at other locations within the reactor building, since the steam tunnel environment has no impact on the failure rate of those valves. As such, all of the reactor instrument line EFCVs will be considered one group.

Any future reactor instrument line EFCV failure would be evaluated in the WNP-2 Corrective Action Program. Also, the WNP-2 10CFR50.65 Maintenance Rule Program was revised to provide a means to track the performance of EFCVs under specific categories. The Maintenance Preventable Functional Failure Reliability Criterion for reactor instrument line EFCV is presently established as no more than one (1) failure per a two (2) year rolling period. The criterion is exceeded on the second failure. Maintenance Rule Functional Failures will be disseminated through the industry via the Equipment Performance Information Exchange (EPIX) system as applicable.

Containment Atmosphere and Suppression Pool EFCV Relocation Basis

Prior to implementing the Improved Standard Technical Specification on WNP-2, the EFCV surveillance requirement was under TS 4.6.3.4. The surveillance required that "Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated Operable...." During the implementation of the Improved Standard Technical Specification, NUREG 1434 (BWR 6) was used for the conversion. However, NUREG 1434 does not address EFCVs. WNP-2 provided a plant specific surveillance addition to reflect the BWR 5 design. Although NUREG 1433 (BWR 4) was consulted, it only addressed reactor instrument line EFCVs as BWR 4's do not have EFCVs in containment atmosphere instrument line applications. WNP-2 conservatively included the containment atmosphere and suppression pool EFCVs in the scope of the SR. This increase in scope to the NUREG 1433 SR was beyond that required by the criteria of 10CFR50.36(c)(3).

The containment atmosphere and suppression pool instrument line EFCV testing (to verify the valve actuates to the isolation position) is not necessary to assure facility operation will be within safety limits as these valves have no active design or safety function. Further, testing is not required to meet a limiting condition for operation. Specifically, these EFCVs are not an instrument used to detect a significant degradation of the reactor coolant boundary, nor is their actuation to the isolation position a design feature that is an initial condition of a design basis accident or transient. These EFCVs are not a component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient, and they are not a component which operating experience or probabilistic risk has shown to be significant to public health and safety. A break of the instrument line would not establish conditions that would cause the EFCV to operate as neither containment pressure nor suppression pool head would be sufficient to cause their actuation. Additionally, EFCVs are not required to close in response to a containment isolation signal and are not postulated to operate under post-LOCA conditions. An instrument line break coincident with a design basis LOCA is outside the design bases of WNP-2. Containment atmosphere and suppression pool instrument line EFCVs are not credited in any accident or transient analysis in the FSAR.

Containment atmosphere and suppression pool instrumentation line EFCVs differ in design, application and test method to those in the reactor instrument lines. Consequently, the failure rate of these valves has no applicability to the reactor instrumentation line EFCVs.

- Design Differences: The containment atmosphere and suppression pool instrument line EFCVs at WNP-2 have the same metal body and the same plunger as the reactor instrument line EFCVs. However, they are fitted with a spring that is designed to actuate under a specified airflow or pressure of approximately 35 psig.

- Application Differences: These EFCVs were installed to meet the requirements of Regulatory Guide 1.11. However, these EFCVs do not provide a design or safety function to close due to either an instrument line break or a LOCA. In the event of instrument line break, normal containment atmosphere or suppression pool water level head is not sufficient to cause these valves to actuate. During a LOCA, these valves must remain open to sense containment pressure and suppression pool level parameters. In comparison, reactor instrument line EFCVs are designed to prevent flow in case of an instrument line break during normal plant operation. However, neither the reactor instrument line nor the containment atmosphere and suppression pool instrument line EFCVs are credited in the WNP-2 instrument line break safety analysis.

- Testing Differences: Containment atmosphere and suppression pool instrument line EFCVs are tested significantly different than reactor instrument line EFCVs. Containment atmosphere EFCVs are tested with air between 32-38 psig. The suppression pool EFCVs are tested with water at pressure between 32-38 psig. During testing the valve is isolated from the containment and the line is pressurized by external means to simulate the desired condition. In comparison, reactor instrument line EFCVs are tested with RPV water at significantly higher pressure.

- Containment Atmosphere and Suppression Pool Instrument Line EFCV Test History:
  WNP-2 EFCV performance history of containment atmosphere and suppression pool instrument line EFCV application has shown high reliability. Testing methodology has been the primary problem in test history results. Seven (7) problems have been reported over the operating life of the plant. Evaluation of these problems has shown that in four (4) cases the problems were directly related to the test methodology. The testing apparatus and method had failed to create the design operating pressure or flow rate that would cause consistent valve actuation. Inadequate air flow due to using hoses too long and too small in diameter resulted in not simulating the desired pressure and flow conditions and thereby yielded faulty results. Three (3) of the problems were concluded as failures. One (1) failure occurred in each of the years 1985, 1995, and 1996. The 1996 failure was attributed to a failed air pump, which resulted in forcing debris in the instrument line and clogging the valve. Although test methodology is suspected as the cause of the 1985 and 1995 failures, sufficient cause determination was not documented at the time to conclusively confirm this. In all seven (7) cases there were no degradation mechanisms that indicate an aging problem with the valves. Likewise, there have been no reported

conditions of these valves inadvertently closing and not reopening. Using 3 EFCV failures over the life of the plant result in a conservative containment atmosphere and suppression pool instrument line EFCV failure rate of 1.2E-6 per hour [3/(15 yr. * 19 valves * 8760 hr/yr)].

The differences between the containment atmosphere and suppression pool instrument line EFCVs and those in the RPV instrumentation lines in design, application, and testing make it inappropriate to apply the failure rate of one into the other. As such, the failure rate of the containment atmosphere and suppression pool instrument line EFCV is not considered as impacting the failure rate of the reactor instrument line EFCVs.

Evaluation of Significant Hazards Considerations

Summary of Proposed Change

The proposed change will relax the frequency of SR 3.6.1.3.8 from testing each EFCV from every 24-months to testing a "representative sample" of reactor instrument line EFCVs every 24-months, such that each reactor instrument line EFCV will be tested at least once every 10 years (nominal). In general, reactor instrument line EFCVs have low failure rates. This high reliability and the low significance associated with a reactor instrument line EFCV failure are the primary bases for this change as documented in a BWROG report NEDO-32977-A. The reactor instrument lines at WNP-2 include a flow-restricting orifice upstream of the EFCV to limit reactor water leakage in the event of a rupture. Previous evaluation of such an instrument line rupture (WNP-2 FSAR 15.6.2) does not take credit for the mitigating action of the EFCV and is bounded by the Steam Line Break analysis (WNP-2 FSAR 15.6.4). This change will also relocate containment atmosphere and suppression pool instrument line EFCV testing requirements to the FSAR.

No Significant Hazards Consideration Determination

Energy Northwest has evaluated the proposed change to the TS using the criteria established in 10CFR50.92(c) and has determined that it does not represent a significant hazards consideration as described below:

- **The operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The current SR frequency requires each reactor instrument line EFCV to be tested every 24-months. The reactor instrument line EFCVs at WNP-2 are designed so that they will not close accidentally during normal operation, but will close if a rupture of the instrument line is indicated downstream of the valve, and have their status indicated in the control room. This proposed change allows a reduced number of reactor instrument line EFCVs to be tested every 24-months. There are no physical plant modifications associated with this change. Industry operating experience demonstrates a high reliability of these valves. Neither reactor instrument line EFCVs nor their failures are capable of initiating previously evaluated accidents; therefore; there can be no increase in the probability of occurrence of an accident regarding this proposed change.

Reactor instrument lines connecting to the reactor coolant pressure boundary are equipped with EFCVs and also have a flow-restricting orifice inside containment and upstream of the EFCV. The consequences of an unisolable rupture of such an instrument line has been previously evaluated in WNP-2 FSAR 15.6.2. The instrument lines that penetrate primary containment conform to Regulatory Guide 1.11 (WNP-2 FSAR 7.1.2.4). Those instrument lines are

Seismic Category I and terminate in instruments that are Seismic Category I (reference WNP-2 FSAR Table 6.2-16 note 27).

The sequence of events in WNP-2 FSAR Section 15.6.2.2 for a reactor instrument line break assumes a continuous discharge of reactor water through the instrument line until the reactor vessel is cooled and depressurized (5 hours). Although not expected to occur as a result of this change, the postulated failure of an EFCV to isolate as a result of reduced testing is bounded by this previous evaluation. Therefore, there is no increase in the previously evaluated consequences of the rupture of an instrument line and there is no potential increase in the consequences of an accident previously evaluated as a result of this change.

The containment atmosphere and suppression pool instrument line EFCVs are required to remain open to sense containment atmosphere and suppression pool level conditions during postulated accidents. They are not required to close during an instrument line break assumed during normal plant operation nor is their design capable of closing during normal plant conditions. These EFCVs do not meet the criteria for inclusion in 10CFR50.36(c)(3) as they have no active safety function and thus relocation of their testing requirements to the FSAR cannot effect the probability of an increase in the consequences of an accident previously evaluated.

- **The operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.**

This proposed change allows a reduced number of reactor instrument line EFCVs to be tested each operating cycle and that the testing requirements for containment atmosphere and suppression pool instrument line EFCVs be relocated to the FSAR. No other changes in requirements are being proposed. Industry operating experience demonstrates the high reliability of these valves. The potential failure of a reactor instrument line EFCV to isolate by the proposed change in testing is bounded by the previous evaluation of an instrument line rupture. This change will not physically alter the plant (no new or different type of equipment will be installed). This change will not alter the operation of process variables, structures, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

- **The operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.**

The consequences of an unisolable rupture of an instrument line has been evaluated in WNP-2 FSAR Section 15.6.2 in accordance with the requirements of Regulatory Guide 1.11. That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (5 hours). The only margin of safety applicable to this proposed change is considered to be that implied by this evaluation. Since a continuous discharge was assumed in this evaluation, any potential failure of a reactor instrument line EFCV to isolate as a result of reduced testing frequency is bounded by existing analysis and does not involve a significant reduction in the margin of safety.

There is no accident for which the containment atmosphere or suppression pool instrument line EFCVs are designed to actuate to the isolation position for mitigation. A postulated break of a containment atmosphere or suppression pool instrument line under normal operating conditions would not result in a condition that would create the ability for these EFCVs to operate because neither the containment pressure nor the suppression pool level head would be sufficient to result in their actuation. As these EFCVs have no active design or safety function, the relocation of testing requirements would not involve a significant reduction in the margin of safety. A postulated break of any instrument line simultaneously with a loss of coolant accident is beyond the design basis for the plant.

Based upon the above, the proposed amendment is judged to involve no significant hazards considerations.

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION SR 3.6.1.3.8
EXCESS FLOW CHECK VALVES TEST FREQUENCY
Attachment 3

ENVIRONMENTAL CONSIDERATION

Energy Northwest has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21

Energy Northwest has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10CFR Section 51.22(c)(9). Pursuant to 10CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The change meets the eligibility criteria for categorical exclusion set forth in 10CFR Section 51.22(c)(9) for the following reasons:

1) does not involve a significant hazards consideration.

2) does not involve an increase in the amounts, or a change in the types, of any effluent that may be released offsite, and

3) this request does not involve an increase in individual or cumulative occupational exposure.

Marked-Up Version of Technical Specification
Surveillance Requirement SR 3.6.1.3.8

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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 INSTRUMENT LINE EFCVs actuates to the isolation position on an actual or 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 the combined leakage rate for all secondary containment bypass leakage paths is ≤ 0.74 scfh when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

Marked-Up Version of Technical Specification
Surveillance Requirement SR 3.6.1.3.8 Bases

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

A REPRESENTATIVE SAMPLE OF REACTOR INSTRUMENT LINES' EXCESS FLOW CHECK VALVES

This SR requires a demonstration that ~~each~~ EFCVs ARE ~~is~~ OPERABLE by verifying that ~~the~~ valve actuates to the isolation position on an actual or simulated instrument line break condition. This SR provides assurance that the REACTOR ~~instrumentation lines~~ EFCVs will perform as designed. The excess flow check valves in reactor ~~pressure sensing lines~~ INSTRUMENT LINES are tested by providing an instrument line break signal with pressure at 85 psig to 110 psig. Testing within this pressure range provides a high degree of assurance that these valves will close during an instrument line break while at normal operating pressure. ~~The excess flow check valves in primary containment pressure sensing lines are tested by providing an instrument line break signal with pressure at approximately 35 psig, since this is the pressure they would expect to experience during a DBA.~~

EACH TESTED

INSERT #1

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 has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ In addition, due to operational concerns, the Surveillance should not be performed during MODES 1, 2, or 3. This restriction has been established to limit the thermal cycles at the containment penetration.

INSERT #2

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. Other administrative controls, such as those that limit the shelf life and

(continued)

Insert 1 to BASES for SR 3.6.1.3.8

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time.

Insert 2 to BASES for SR 3.6.1.3.8

The nominal 10 year interval is based on performance testing. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint. (Reference 5)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.12

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 1 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is $\leq$ 1.0 gpm times the total number of hydrostatically tested PCIVs when tested at 1.1 P_a (41.8 psig). The combined leakage rates must be tested at the Frequency required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. FSAR, Chapter 6.2.

2. FSAR, Section 15.2.4.

3. 10 CFR 50.36(c)(2)(ii).

4. Licensee Controlled Specifications Manual.

5. GE Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated November 1998

REQUEST FOR AMENDMENT
TECHNICAL SPECIFICATION SR 3.6.1.3.8
EXCESS FLOW CHECK VALVES TEST FREQUENCY
Attachment 5

Replacement Pages for Technical Specification
Surveillance Requirement SR 3.6.1.3.8

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.3	----------NOTES---------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. ---------- Verify each primary containment isolation manual valve and blind flange that is located inside primary containment and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
SR 3.6.1.3.4	Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
SR 3.6.1.3.5	Verify the isolation time of each power operated and each automatic PCIV, except MSIVs, is within limits.	In accordance with the Inservice Testing Program

(continued)