Robert G. Byram
Senior Vice President and
Chief Nuclear Officer

PPL Susquehanna, LLC
Two North Ninth Street
Allentown, PA 18101-1179
Tel. 610.774.7502 Fax 610.774.5019
rgbyram@pplweb.com

OCT 0 4 2000

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 233 TO LICENSE
NPF-14: AND PROPOSED AMENDMENT NO. 198 TO LICENSE
NPF-22: RELAXATION OF SURVEILLANCE TESTING REQUIREMENTS
FOR EXCESS FLOW CHECK VALVES AND SUBMITTAL OF
PERTINENT IST PROGRAM RELIEF REQUESTS **Docket Nos. 50-387**
PLA-5227 **and 50-388**

Pursuant to 10CFR50.59 and 10CFR50.55a(a)(3)(i), PPL Susquehanna LLC proposes to amend the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Technical Specifications (TS) and requests the approval of the complementary Inservice Testing (IST) relief requests. The proposed changes will modify TS Surveillance Requirement SR 3.6.1.3.9 to relax the SR frequency by allowing a representative sample of Excess Flow Check Valves (EFCVs) to be tested every 24 months, such that each EFCV will be tested at least once every ten years. The SR reflected in the current SSES Technical Specifications requires testing all EFCVs every 24 months. The IST relief request is being submitted to modify the IST program to be consistent with the proposed TS change.

The reduced testing associated with this proposed change will result in an increase in the availability of the associated instrumentation, will save outage critical path time and cost, and will result in dose savings without significantly impacting the health and safety of the public.

Attachment 1 presents the Safety Assessment for the proposed change.

Attachment 2 contains the "No Significant Hazards Consideration" and "Environmental Considerations" assessments. The "No Significant Hazards Considerations" assessment concludes that the proposed Technical Specification revisions do not involve a significant

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increase in the probability or consequence of an accident previously evaluated; do not create the possibility of a new or different kind of accident from any accident previously evaluated; and do not involve a significant reduction in the margin of safety. The "Environmental Considerations" assessment concludes that the revisions conform to the criteria for actions eligible for categorical exclusion as specified in 10CFR51.22(c)(9), and will not impact the environment.

Attachment 3 contains marked-up pages of the Unit 1 and Unit 2 Technical Specifications.

Attachment 4 contains "camera ready" versions of the revised Unit 1 and Unit 2 Technical Specification pages.

Attachment 5 contains, for your information, markups of the associated TS bases.

Attachment 6 contains the revised Unit 1 and Unit 2 IST program Relief Request numbers RJ20 and RR23.

The proposed changes have been approved by the SSES Plant Operations Review Committee and reviewed by the Susquehanna Review Committee.

PPL plans to implement the proposed changes during the Unit 2 Refueling and Inspection Outage scheduled Spring 2001. Therefore, we request NRC complete the review of this change request by February 1, 2001, to support our scheduled implementation dates.

Please contact Mr. M. H. Crowthers at (610) 774-7766 if there are any questions concerning this submittal.

Sincerely,

R. G. Byram

Attachments

cc: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Sr. Project Manager
Mr. D. J. Allard, Dept. of Environmental Protection

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-387

PROPOSED AMENDMENT NO. 233 TO LICENSE NPF-14:
RELAXATION OF SURVEILLANCE TESTING REQUIREMENTS
FOR EXCESS FLOW CHECK VALVES AND SUBMITTAL OF
PERTINENT IST PROGRAM RELIEF REQUESTS
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment contains a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
BY:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 4th day of October, 2000.

Nancy J. Lannen
Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

PROPOSED AMENDMENT NO. 198 TO LICENSE NPF-22:
RELAXATION OF SURVEILLANCE TESTING REQUIREMENTS FOR EXCESS FLOW CHECK VALVES AND SUBMITTAL OF PERTINENT IST PROGRAM RELIEF REQUESTS
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment contains a revision to the Susquehanna SES Unit 2 Technical Specifications.

PPL Susquehanna, LLC
BY:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 4th day of October, 2000.

Nancy J. Lannen
Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

ATTACHMENT 1 TO PLA-5227

SAFETY IMPACT ASSESSMENT

Safety Impact Assessment

Section 1: INTRODUCTION

SSES Technical Specifications (TS) require, during each refueling outage, performance of surveillance tests (SR) on each Excess Flow Check Valve (EFCV). The BWR Owners' Group (BWROG) has developed a basis for relaxing the requirement to test each EFCV during each refueling outage (reference 1). Duane Arnold and Fermi have received approval to implement this change. The change proposed herein to the SSES Technical Specification is consistent with these submittals.

The reduced testing associated with the proposed change will result in an increase in the availability of the instrumentation during outages, a savings in outage critical-path time, outage cost, and personnel dose, without significantly impacting the health and safety of the public.

Section 2: CHANGE DESCRIPTION

TS SR 3.6.1.3.9 currently requires verification of the actuation capability of each reactor instrumentation line EFCV every 24 months. This SR is currently implemented for a set of EFCV's preceding a planned Refueling Outage and for the balance of the EFCV's during the reactor system leak test that is performed at the end of each planned Refueling Outage.

The proposed change is to relax the SR by allowing a representative sample of EFCVs to be tested every 24 months, such that each EFCV will be tested at least once every 10 years.

Section 3: SAFETY ASSESSMENT

This SR demonstrates that each EFCV is OPERABLE by verifying that the valve actuates to check flow on a simulated instrument line break downstream of the valve. 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 were performed with the reactor at power. Performance of the SR requires the reactor to be pressurized to near normal operating pressure. The SR is implemented by opening the instrument side of the valve while the process side is exposed to reactor pressure.

The SSES Final Safety Analysis Report (FSAR) Section 6.2.4.3.5 identifies that instrument lines penetrating the containment at SSES from the reactor coolant pressure boundary conform to Regulatory Guide 1.11. They are equipped with a restricting orifice located inside the drywell and as close as practicable to the connection on the process pipe with an excess flow check valve located outside as close as practicable to the containment. A manual isolation valve exists between each penetration and its associated EFCV. These manual isolation valves serve no containment isolation function. Should an instrument line which forms part of the RCPB develop a leak outside containment, a flow rate which results in a differential pressure across the EFCV of 3 to 10 psi will cause the check valve to close automatically. EFCVs are designed to have no more than a maximum of 2.0 cc/hr leakage rate per inch seat diameter in the closed position. Should an excess flow check valve fail to close when required, the main flow path through the valve has a resistance to flow at least the equivalent of a sharp-edged orifice of 0.375-inch diameter. Valve position indication and excess flow alarm are provided in the control room. Excess flow check valves in instrument lines penetrating reactor containment undergo periodic inservice testing in accordance with the SSES Inservice Testing program.

Instrument lines that do not connect to the reactor coolant pressure boundary conform to Regulatory Guide 1.11 through their qualification and installation in accordance with ASME Section III, Class 2 requirements. They are designated as "extensions of containment" as discussed in FSAR Subsection 3.13.1 and FSAR Tables 6.2-12a and 6.2-22. They are equipped with isolation and excess flow check valves whose status will be indicated in the control room.

The Boiling Water Reactor Owners' Group (BWROG) has issued a report that provides a basis for this request. This report (reference 1) provides justification for the relaxation in the SR requirement to test each EFCV every 24 months. The report demonstrates the high degree of EFCV reliability and the low consequences of an EFCV failure.

Industry experience as documented in the BWROG report indicates that EFCVs have a very low failure rate. At SSES, the SR failure rate has been approximately 1%. Only half of these SR failures have resulted in replacement of the EFCV. The SSES test history shows no evidence of common mode failure. This SSES test experience is consistent with the findings of the BWROG report. The BWROG report indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCVs at SSES, consistent with the industry, have exhibited a high degree of reliability. Any future EFCV failure will be evaluated in accordance with the SSES Corrective Action Program. Additionally, and as part of the implementation of this proposed amendment, the SSES 10CFR50.65

Maintenance Rule Program will be revised to include a specific EFCV performance acceptance criteria of no more than one failure per 24 months. This criteria recognizes that the SR frequency is 24 months.

The FSAR Section 15.6.2 demonstrates (consistent with the BWROG report) that the failure of an EFCV has very low consequence. SSES FSAR Section 15.6.2 evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation assumes the EFCV fails to isolate the break. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10CFR100 limits. Thus the failure of an EFCV, though not expected as a result of this proposed change, does not affect the dose consequences of an instrument line break.

Based on the above, the SSES FSAR and the BWROG report demonstrate a high degree of EFCV reliability and the low consequences of an EFCV failure.

Since this proposed change only affects the population of EFCV's tested each planned refueling outage, it has no impact on operator performance or operator procedures.

This change has no impact on the FSAR. The FSAR, as discussed above, contains bounding evaluations of an instrument line break that is not affected by the proposed change.

Section 4: REFERENCES

1. NEDO-32977-A "Excess Flow Check Valve Testing Relaxation", June 2000

BWR OWNERS' GROUP

W. Glenn Warren, Chairman
Tel: (205) 992-5940
Fax: (205) 992-0391
wgwarren@southernco.com

c/o Southern Nuclear • 40 Inverness Center Parkway • PO Box 1295 • Birmingham, AL 35242

BWROG-00069
June 14, 2000

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: **TRANSMITTAL OF APPROVED GE LICENSING TOPICAL REPORT, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" dated November 1998.**

Dear Sir(s):

The purpose of this letter is to transmit the approved BWROG Licensing Topical Report NEDO 32977-A, "Excess Flow Check Valve Testing Relaxation" dated November, 1998.

This approved Licensing Topical Report provides justification for relaxed testing requirements for Excess Flow Check Valves in instrument lines connected to the Reactor Coolant Pressure Boundary (RCPB). As indicated in the summary of attachments, this Approved Licensing Topical Report includes the NRC's Request for Additional Information, BWROG generic responses to RAIs, and the NRC's Safety Evaluation Report.

Very truly yours,

W.G. Warren, Chairman
BWR Owners' Group

cc: JM Kenny, BWROG Vice Chairman
TG Hurst, GE
SA Bump, GE

Attachments: (1) Nuclear Regulatory Commission Safety Evaluation of GE Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" (TAC NOS MA7884 and M84809) March 14, 2000

(2) GE Nuclear Energy Licensing Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" dated November 1998

(3) BWROG-99086, Letter from W. G. Warren, Chairman (BWROG) to NRC dated December 17, 1999, "Generic Response to Request for Additional Information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements."

(4) NG-99-0308, Letter from J. Franz (IES Utilities) to NRC, dated April 12, 1999, "Technical Specification Change Request (TSCR-010): Relaxation of Excess Flow Check Valve Surveillance Testing."

(5) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 27, 1999, "Request For Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)."

(6) NG-99-1358, Letter from K. Peveler (IES Utilities) to NRC, dated October 5, 1999, "DAEC Response to Request For Additional Information on Technical Specification Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements"

(7) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 30, 1999, "Request for Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)"

(8) NG-99-1383, Letter from Ken Peveler (IES Utilities) to NRC, dated October 8, 1999 "DAEC Response to Request for Additional Information On Technical Specification Change Request (TSCR) Regarding Excess Flow Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements.

Overview

Comments on Topical Report Nomenclature

The subject Topical Report on Excess Flow Check Valve Testing Relaxation was included in a lead plant submittal prior to receiving generic approval from the NRC. The report did not have a "NEDO" number at the time and was identified with the GE Nuclear Energy Design Record File Number **B21-00658-01** in previous correspondence. This approved report has been assigned as **NEDO-32977-A**, and is identified as such in this publication. This note calls attention to the two numbers used to refer to this report in an attempt to avoid confusion. Both numbers refer to the same Licensing Topical Report. The only modification to this report is the correction of the typographical error identified in NRC Lead Plant Question 4.

Background

Topical Report B21-00658-01 "Excess Flow Check Valve Testing Relaxation" was completed by GE Nuclear Energy for the Boiling Water Reactor Owner's Group (BWROG) in November of 1998. IES Utilities - Duane Arnold Energy Center (DAEC) acting as the lead-plant, included the Topical Report in its 1999 submittal to the NRC for relaxed testing of Excess Flow Check Valves. While DAEC was responding to the NRC's Request for Additional Information (RAI) regarding its submittal, the NRC requested that the BWROG submit a generic response to the same RAI's. The NRC, after reviewing the generic response, issued a Safety Evaluation of the GE Nuclear Energy Topical Report.

The NRC's Safety Evaluation, issued in March of 2000, requested that the BWROG either develop an industry-wide performance criteria or indicate that each licensee would develop their own EFCV performance criteria. This criteria should be based upon sound reliability modeling which is consistent with the generally expected performance of the EFCVs.

The BWROG Committee addressing the EFCV issue elected not to attempt to develop an industry-wide performance criteria. Individual licensees will be required to develop their own EFCV performance criteria and basis.

The considerations that resulted in the above decision are as follows:

- Lack of current consensus on handling the EFCV performance criteria issue.
- Potential to develop a criteria overly restrictive for some and not restrictive enough for others.
- Relative ease with which DAEC managed performance criteria issue by including EFCVs as a subset within the Maintenance Rule.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 14, 2000

Mr. W. Glenn Warren
BWR Owners Group Chairman
Southern Nuclear Company
42 Inverness Parkway
PO Box 1295
Birmingham, AL 35201

SUBJECT: SAFETY EVALUATION OF GENERAL ELECTRIC NUCLEAR ENERGY TOPICAL REPORT B21-00658-01, "EXCESS FLOW CHECK VALVE TESTING RELAXATION" (TAC NOS. MA7884 AND M84809)

Dear Mr. Warren:

The Nuclear Regulatory Commission (NRC) staff has completed its review and evaluation of General Electric Nuclear Energy (GE) Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998. This topical report was written by GE for the Boiling Water Reactor (BWR) Owners Group. It was submitted on April 12, 1999, by IES Utilities, Inc., the licensee for the Duane Arnold Energy Center (DAEC), as part of a plant-specific license amendment request. We approved the plant-specific license amendment on December 29, 1999. The topical report was reviewed for its generic applicability and the applicable safety evaluation (SE) is enclosed.

Based on the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with extremely low likelihood that this release could impact core damage frequency and negligible consequence of a release in the reactor building, we conclude that the increase in risk associated with DAEC's request for relaxation of excess flow check valve (EFCV) surveillance testing is sufficiently low and acceptable. With respect to other BWR's, we anticipate that similar conclusions can be drawn. Therefore, we consider the risk analysis portion of the topical report to be acceptable.

We also agree with the topical report that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The topical report, however, lacks guidance for an individual plant to establish EFCV performance criteria and the basis which, we believe, is essential to ensure that a corrective action program provide meaningful feedback for appropriate corrective action. Therefore, we expect the BWR Owners Group to revise the topical report to either require each licensee to develop their EFCV performance criteria and the basis or develop an industry-wide performance criteria and the basis. As noted in the enclosed SE, the EFCV performance criteria should be based on sound reliability modeling that is consistent with generally expected performance of the EFCVs. We also note that such performance criteria and the basis, once developed, will be subject to staff review.

In conclusion, we find the topical report acceptable for referencing in relaxation of EFCV surveillance testing, subject to the conditions stated above. We also find the STS changes proposed by TSTF-334 to be acceptable, pending our acceptance of industry's development of EFCV performance criteria and the basis.

In accordance with procedures established in NUREG-0390, it is requested that the BWR Owners Group publish the accepted version of this report within three months of receipt of this letter. The accepted version should incorporate this letter and the appropriate evaluation between the title page and the abstract. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Sincerely,

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: Safety Evaluation

cc w/encl: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BOILING WATER REACTOR OWNERS GROUP

GENERAL ELECTRIC NUCLEAR ENERGY TOPICAL REPORT B21-00658-01

"EXCESS FLOW CHECK VALVE TESTING RELAXATION"

1.0 INTRODUCTION

By letter dated April 12, 1999, as supplemented by letters dated October 5 and 8, 1999, Alliant Energy, the licensee for Duane Arnold Energy Center (DAEC), requested a license amendment which would allow relaxation of the frequency of surveillance testing of excess flow check valves (EFCVs) in reactor instrumentation lines. The proposed change was to relax the surveillance frequency by allowing a "representative sample" of EFCVs to be tested every 24 months, rather than having each EFCV tested every 24 months, as was previously required. The intent was to test approximately 20 percent of the EFCVs each 24 months such that each EFCV would be tested at least once every 10 years (nominal). The stated basis for the request was a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. The analysis to support this conclusion was based on the General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" (prepared for Boiling Water Reactor (BWR) Owners Group). The topical report was submitted as part of the Alliant Energy request for license amendment for DAEC.

We granted Alliant Energy's request with the issuance of Amendment No. 230 to Facility Operating License No. DPR-49 on December 29, 1999. As part of that action, we accepted the topical report insofar as it was applied to the DAEC case.

In this safety evaluation, we are reviewing the topical report for its generic applicability. By letter dated January 6, 2000, the BWR Owners Group provided additional information on the topical report, in the form of answers to our requests for additional information which we had made earlier during the DAEC review.

Furthermore, we are reviewing Standard Technical Specification Change Traveler TSTF-334, Revision 0, dated June 2, 1999, which was proposed by the industry Technical Specification Task Force. TSTF-334 implements the EFCV testing relaxation proposed by the topical report.

2.0 BACKGROUND

EFCVs in reactor instrumentation lines are used in BWR containments to limit the release of fluid from the reactor coolant system in the event of an instrument line break. Examples of EFCVs include reactor pressure vessel level/pressure instrument, main steam line flow instrument, recirculation pump suction pressure instrument, and reactor core isolation cooling steam line flow instrument. EFCVs are not required to close in response to a containment isolation signal and are not postulated to operate under post-LOCA conditions. The topical report states that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with a design basis LOCA would be of a sufficiently low probability to be outside of the design basis.

BWR/4 Standard Technical Specification (STS) Surveillance Requirement (SR) 3.6.1.3.10 currently requires verification of the actuation (closing) capability of each reactor instrumentation line EFCV every [18] months. This is typical for the technical specifications (TS) at most operating BWR/4 plants. The proposed change is to relax the SR frequency by allowing a "representative sample" of EFCVs to be tested every [18] months. The intent is to test approximately 20 percent of the EFCVs every [18] months such that each EFCV will be tested at least once every 10 years (nominal). The proposed change is similar in principle to existing performance-based testing programs, such as inservice testing of snubbers and Option B of Appendix J to 10 CFR Part 50.

TSTF-334 includes a revised Basis for TS 3.6.1.3.10. The revised Basis states, in part:

> 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.... The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. 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.

3.0 EVALUATION

This evaluation has been divided into two parts, a systems review (3.1) and a risk and radiological review (3.2).

3.1 Systems Review

The topical report provides detailed information about EFCV surveillance testing at 12 BWR plants. Testing history indicates that there is a low failure rate in EFCV surveillance testing (see Section 3.2.1 below). Thus, EFCVs have generally been very reliable performers.

The dose consequences would be low if an EFCV failed to close if an instrument line broke during normal operation (see Section 3.2.2 below).

cc:

Mr. James M. Kenny
BWR Owners' Group Vice Chairman
PP&L, Inc.
Mail Code GENA6-1
Allentown, PA 18101-1179

Mr. Thomas J. Rausch
RRG Chairman
Commonwealth Edison Company
Nuclear Fuel Services
1400 Opus Place, 4th Floor
Downers Grove, IL 60515-5701

Mr. Drew B. Fetters
PECO Energy
Nuclear Group Headquarters
MC 61A-3
965 Chesterbrook Blvd.
Wayne, PA 19087-5691

Mr. H. Lewis Sumner
Southern Nuclear Company
40 Inverness Parkway
PO Box 1295
Birmingham, GA 35201

Mr. Carl D. Terry
Vice President, Nuclear Engineering
Niagara Mohawk Power Corporation
Nine Mile Point - Station
OPS Bldg/2nd Floor
PO Box 63
Lycoming, NY 13093

Mr. George T. Jones
PP& L, Inc.
MC GENA6-1
Two North Ninth Street
Allentown, PA 18101

Mr. John Kelly
New York Power Authority
14th Floor Mail Stop 14K
Centroplex Building
123 Main Street
White Plains, NY 10601

Mr. Thomas G. Hurst
GE Nuclear Energy
M/C 182
175 Curtner Avenue
San Jose, CA 95125

Mr. Thomas A. Green
GE Nuclear Energy
M/C 182
175 Curtner Avenue
San Jose, CA 95125

3.1.1 Request For Additional Information

The staff requested additional information from the BWR Owners Group regarding certain system aspects of their request, and they responded on January 6, 2000. The following three sections discuss these issues.

3.1.1.1 Test Interval Increase

The topical report compares this situation to Option B of Appendix J to 10 CFR Part 50. We revised Appendix J in 1995 by adding Option B, which provides a risk-informed, performance-based approach to leakage rate testing of containment isolation valves. We also developed Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, as a method acceptable to us for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to us for complying with Option B, with four exceptions which are described in the Regulatory Guide.

According to the NEI document, containment isolation valve test intervals may be increased to 5 years or 3 refueling outages if a valve has shown good performance (i.e., two consecutive successful tests), and, if certain other conditions are met, to as much as 10 years. However, the Regulatory Guide took exception to those provisions of the NEI document, stating that test intervals should not exceed 5 years. The Regulatory Guide explained that this was because of uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical containment isolation valve performance data, and because of the indeterminate time period of three refueling cycles and insufficient precision of programmatic controls described in the NEI document to address these uncertainties.

Topical Report B21-00658-01 states that the NEI document allows a 10-year test interval, and that Regulatory Guide 1.163 endorsed the NEI document, without mentioning our exception to 10 years. We asked the BWR Owners Group to justify their proposal for a 10-year testing interval.

Their response indicated that the topical report established its own basis for the testing relaxation, that being high reliability, low risk, and low radiological consequences. They have proposed a cyclic nominal interval for testing a representative sample of the EFCVs. The valves are of similar design, similar application, and similar service environment. Performance of the representative sample provides a strong indicator of the performance of the total population. The 10-year nominal interval solely limits the time between tests for any specific valve and provides additional assurance that all valves remain capable of performing their intended function. The BWR Owners Group considers the valve failure rate data listed in Table 4-1 of the topical report to be the primary basis for the performance-based interval. In addition, they assume that the off-site dose consequences of a failure to isolate have been evaluated and found to be acceptable, although each site adopting this change will need to confirm the applicability of this assumption.

Further, they did specifically address the reasons Regulatory Guide 1.163 had for not accepting a 10-year interval by providing the following information:

- Unquantified leakage rates for test failures are not applicable because the maximum leakage through an unisolated instrument line is quantified. The dose consequences of the failure to isolate are acceptable (see Section 3.2.2 below).
- Repetitive/common-mode failures are not applicable as evidenced by the low industry failure rate and more specifically by the Topical Report, Table 4-2, "EFCV Failure Rates by Manufacturer."
- Aging effects are not a concern. The industry data already provided does not indicate any increase in failure rate with time in service.
- Historical performance data associated with EFCVs has been provided and is considered adequate to justify the proposed interval.
- There is no indeterminate time period involved with this proposed change. Every fuel cycle, a representative sample will be tested.

We generally agree with the BWR Owners Group's assessment, except for the aging question which is addressed in Section 3.2.1 below. Regulatory Guide 1.163 had to consider all varieties of containment isolation valves, from a fraction of an inch to several feet in diameter, carrying liquid or gas in a wide range of temperatures and pressures. Different types of valves (gate, globe, check) made of various materials, by different manufacturers, and with varying safety significance, had to be accounted for. On the other hand, EFCVs in reactor instrumentation lines are a very specific, narrow class of valves. Their history and performance are well-documented. Based on their historically high reliability and their low risk significance and radiological consequences should they fail (as discussed in Section 3.2, below), we accept the proposal that the magnitude of a test interval extension may be as great as 10 years.

3.1.1.2 Failure Feedback Mechanism

In our second question, we pointed out that, under Appendix J, Option B, testing programs, a valve that fails a test after having been put on an extended test interval must return to its original interval until it once again shows good performance (i.e., passes two consecutive tests). Risk-informed inservice testing Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," also specifies the need for a failure feedback mechanism. Topical Report B21-00658-01 has no specific failure feedback mechanism, although it does state that each plant's corrective action programs must evaluate equipment failures and establish appropriate corrective actions.

The BWR Owners Group replied that each licensee who adopts the relaxed surveillance intervals recommended by the topical report should ensure that an appropriate feedback mechanism to respond to failure trends is in place, and that TSTF-334 includes this commitment (see the proposed Basis for TSTF-334, quoted in Section 2.0, above).

Considering the historically high reliability of the EFCVs and their low risk significance and radiological consequences should they fail, we find that it is not necessary for the topical report to provide a specific failure feedback mechanism. However, see Section 3.2.1 below for an additional discussion of the corrective action program and EFCV performance criteria.

3.1.1.3 Technical Specification Level of Detail

In TSTF-334, the proposed TS says "a representative sample" of EFCVs will be tested every [18] months. The "representative sample" is not defined in the TS itself. The proposed Basis says that a licensee will test 20 percent of the valves each refueling outage and thus test all of them in a 10-year period. We asked the BWR Owners Group to justify having the specific requirements in the Bases, rather than in the proposed TS.

They replied that the term "representative sample," with an accompanying explanation in the TS Bases, is identical to current usage in the STS, NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term "representative" in TS SR 3.8.6.3, in reference to battery cell testing, and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. Therefore, the application of a "representative sample" for the EFCV testing SR, with its accompanying definition in the Bases, is consistent with the STS usage.

They also provided additional examples and explanations that supported their proposed TS as being consistent with current STS practices.

Therefore, we find the proposed TS and Bases wording to be acceptable.

3.2 Risk and Radiological Review

Below is our review of the risk and radiological analysis associated with this request. Specifically, we evaluated: (1) the estimate of the steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close and (2) the assessment of the radiological consequence of such release.

The instrument lines at DAEC include a 1/4 inch flow restriction orifice upstream of the EFCVs to limit reactor water leakage in the event of a rupture. As discussed below, in Section 3.2.2, previous evaluation of such an instrument line rupture in DAEC Updated Final Safety Analysis Report (UFSAR) 1.8.11, for which the EFCVs are designed to mitigate, do not credit the isolation of the line by the EFCVs. Thus, a failure of an EFCV is bounded by the previous evaluation of an instrument line rupture. This analysis also showed that the resulting offsite doses would be well below regulatory limits. Further discussion on radiological impact is provided in Section 3.2.2 of this safety evaluation.

The operational impact of an EFCV that is connected to the reactor pressure vessel (RPV) boundary failing to close is based on the environmental effects of a steam release in the vicinity of the instrument racks. The environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the released steam into the reactor building. However, the topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems and that the integrity and functional performance of secondary containment following instrument line break would be met.

The separation of equipment in the reactor building is also expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. Nevertheless, the presence of an unisolated steam leak into the reactor building requires the licensee to shut down the reactor and depressurize to allow access to manually isolate the line.

The BWR Owners Group estimation of the steam release frequency caused by an instrument line break concurrent with an EFCV failing to close is reviewed in Section 3.2.1 of this report. The assessment of the radiological consequences of such release is reviewed in Section 3.2.2.

3.2.1 Estimation of Release Frequency

In estimating the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency and (2) the probability of EFCV failing to close. The BWR Owners Group assumed a single instrument line break frequency of 3.52E-05/year. This estimate was based on the EPRI Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants", dated July 1992. This frequency corresponded to pipe sizes between ½ inch to 2 inches in diameter and the BWR Owners Group considered these pipe sizes to represent the subject instrument line piping. Thus, for DAEC, the product of this single instrument line break frequency and the total number of instrument lines at DAEC, 94, resulted in the total plant instrument line break frequency estimate of 3.31E-3/year.

Since the above single instrument line break frequency represents recent data, we consider its application to estimate the plant instrument line break frequency to be acceptable. We note, however, that the total plant instrument line break frequency for a plant depends on the total number of instrument lines present at the plant.

The probability of EFCV failing to close (or EFCV unavailability) was estimated using the formula:

$$\bar{A} = \lambda * \theta / 2$$

Where:
- $\bar{A}$ is the EFCV unavailability
- λ is the EFCV failure rate per year
- θ is the EFCV surveillance test interval in years

The EFCV failure frequency, λ, was estimated using the formula:

$$\lambda_u = \chi^2_{\alpha;2r+2} / 2T$$

Where :
- λ_u is the upper limit failure rate per year
- T is the operating time in years
- r is the number of failures
- $\chi^2_{\alpha:2r+2}$ is the value taken from the chi-square distribution tables which corresponds to 2r+2 degrees of freedom at $\alpha = 0.05$ (0.95 confidence level)

The topical report determined an upper limit EFCV failure rate based on 11 observed failures in about 12424.5 years of service for 12 BWR plants in the U.S. (Note: 12424.5 years was determined by multiplying the number of tested EFCVs with the time period during which the number of occurring failures was reported). For eleven observed EFCV failures, the EFCV upper limit failure rate, λ_u, was estimated to be about 1.5E-3/year.

It is noted that the formula for estimating the EFCV failure rate, λ_u, assumes that this failure rate is constant over time. Therefore, to account for the possibility that the failure rate for EFCV may change over time, potentially due to age-related factors, the EFCV failure rate was assumed to change by five fold in the report's analysis. We consider the use of this method to be acceptable.

In addition to accounting for a potential age-related degradation in the EFCV failure rate estimate, the topical report stated that each plant's corrective action programs must evaluate equipment failures and establish appropriate corrective actions. We consider this requirement to be prudent and necessary. However, to ensure that such a program can provide meaningful feedback for appropriate corrective actions, we believe that the topical report should require each licensee to develop their EFCV performance criteria and the basis.

For 55 EFCV failures (5 times the actual number of EFCV failures observed for 12 BWR plants), degrees of freedom (2r +2, where r is the number of failures) is 112. Chi-squared values, $\chi^2_{\alpha:2r+2}$, are not typically provided for degree of freedom values above 30 because a chi-squared distribution with degrees of freedom over 30 approximates the standard normal distribution. In such case χ^2 is approximated by:

$$\chi^2 = \tfrac{1}{2} (Z + (2n-1)^{1/2})^2$$

Where: - Z is the corresponding standard deviation (or a z-score) for α-point of the standard normal distribution
- n is the degrees of freedom

Thus, for a 0.95 confidence level ($\alpha = 0.05$), Z is 1.645. And,

$$\chi^2 = \tfrac{1}{2} (1.645 + (2*112-1)^{1/2})^2 = 137.42$$

Therefore, EFCV upper limit failure frequency was then calculated to be:

$$\lambda_u = \chi^2 / 2T = 137.42 / (2 * 12424.5 \text{ years}) = 5.53\text{E-3 failures per year}$$

The release frequency was then calculated by the formula:

$$\begin{aligned} RF &= I * \bar{A} \\ &= I * \lambda_u * \theta / 2 \end{aligned}$$

Where:
- RF is the release frequency
- I is the instrument line failure frequency (per year)
- $\bar{A}$ is the EFCV unavailability (calculated by $\lambda * \theta / 2$)
- θ is the surveillance interval in years

Using the surveillance interval for 2 years (current practice), the instrument line break frequency of 3.31E-3/year at DAEC, and total plant EFCV failure frequency of 5.53E-3/year, the release frequency was estimated to be 1.8E-5/year. For a surveillance interval of 10 years, the release frequency was estimated to be about 9.1E-5/year, which depicts an increase of about 7.3E-5/year from that of the 2-year surveillance test interval. It represents the increase in the total plant release frequency for a random break of any of the 94 instrument lines in DAEC and a concurrent failure of the line's EFCV to close to isolate the break.

We do not consider this estimated increase in release frequency, 7.3E-5/year, to be significant. This conclusion is based on the point that this frequency is lower than the DAEC large-break loss of coolant accident (LOCA) frequency of 3E-4/year which has the potential to lead to a core damage accident whereas the instrument line break concurrent with EFCV failing to close does not.

In addition, we consider the above method for assessing the impact of EFCV surveillance test interval increase to 10 years (along with an assumed five-fold increase in the EFCV failure rate) to be acceptable. We note that the use of observed industry data for instrument line break and EFCV failures is sound for DAEC's case. However, for a plant whose instrument line break frequency and/or EFCV failure rate exceed that of the industry average, the plant-specific data should be applied in the estimation of the release frequency. We also recognize that the method of estimating the EFCV unavailability is consistent with industry practice and that accounting for a potentially unknown change in the valve's failure rate is prudent.

3.2.2 Radiological Consequences

The lead plant, DAEC, noted that they previously evaluated the radiological consequences of an unisolable rupture of such an instrument line in response to Regulatory Guide 1.11, as documented in DAEC UFSAR 1.8.11. This evaluation assumed a continuous discharge of reactor water through an instrument line with a 1/4-inch orifice for the duration of the detection and cooldown sequence. The assumptions for the accident evaluation do not change as a result of the proposed TS change, and the evaluation in DAEC UFSAR 1.8.11 remains acceptable. Therefore, we find acceptable the licensee's determination that the proposed amendment will not involve a significant increase in the consequences of an accident previously evaluated.

The topical report also maintained that radiological consequences from reactor coolant pressure boundary instrument line breaks have been evaluated at most plants to show compliance with Regulatory Guide 1.11 and are documented in some UFSARs. A typical GE radiological evaluation of the instrument line break with and without a 1/4 inch orifice installed has been conducted using a GE methodology which has been accepted by us in BWR FSAR submittals. The results of the evaluation indicated that even without a 1/4 inch orifice installed, the resulting thyroid dose at the site boundary is about 5 percent of the regulatory limit. The report concluded, therefore, that the radiological consequence of EFCVs failing to function upon demand is sufficiently low to be considered insignificant. The report further stated that specific analyses are needed to confirm this conclusion at each plant, but that similar results would be expected.

4.0 CONCLUSION

As demonstrated in the topical report, the impact of an increase in EFCV surveillance test intervals to 10 years along with an assumed five-fold increase in the EFCV failure rate on the likelihood of a release inside the reactor building was shown to result in a release frequency of about 9.1E-5/year for DAEC. This represents an increase of about 7.3E-5/year from the current release frequency estimate (for 2-year surveillance test interval) of about 1.8E-5/year. We consider this estimate for DAEC to be sufficiently low, especially since the consequence of such an accident is not expected to lead to a core damage. For some BWR plants, the estimated release frequency may be higher than DAEC's estimate when plant-specific instrument line break frequency and/or EFCV failure rate (that are higher than the industry average) are used in the calculation. However, based on the reported individual plant EFCV failure data, we do not anticipate significant deviation from the estimate derived for DAEC.

We also agree with the topical report that the consequences of steam release from the depicted events is not significant, as it was supported by a previous analysis. Based on the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with unlikely impact on core damage and negligible consequence of a release in the reactor building, we conclude that the increase in risk associated with DAEC's request for relaxation of EFCV surveillance testing is sufficiently low and acceptable. With respect to other BWRs, we anticipate that similar conclusions can be drawn. Therefore, we consider the risk analysis portion of the topical report to be acceptable.

We also agree with the topical report that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The topical report, however, lacks guidance for an individual plant to establish EFCV performance criteria and the basis which, we believe, is essential to ensure that a corrective action program can provide meaningful feedback for appropriate corrective action. Therefore, we expect the BWR Owners Group to revise the topical report to either require each licensee to develop their EFCV performance criteria and the basis or develop an industry-wide performance criteria and the basis. The EFCV performance criteria should be based on sound reliability modeling that is consistent with generally expected performance of the EFCVs. We also note that such performance criteria and the basis, once developed, will be subject to staff review.

In conclusion, we find the topical report acceptable for referencing in relaxation of EFCV surveillance testing, subject to the conditions stated above. We also find the STS changes proposed by TSTF-334 to be acceptable, pending our acceptance of industry's development of EFCV performance criteria and the basis. We also note, as stated in TSTF-334, that some plants may require an inservice testing program relief request pursuant to 10 CFR 50.55a in order to implement these TS changes.

Principal Contributor: S. Lee

Date: March 14, 2000

PROPOSED CHANGE TSCR-010 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend the Technical Specifications by deleting the referenced page and replacing it with the enclosed new page.

SUMMARY OF CHANGES:

Page	Description of Changes
3.6-14	Revises the description of the SURVEILLANCE for SR 3.6.1.3.7 to state, "Verify a representative sample of reactor instrumentation line EFCVs actuate on a simulated instrument line break to restrict flow."

Insert 1 to BASES for SR 3.6.1.3.7

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). The nominal 10 year 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 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 7).

SAFETY ASSESSMENT

1. Introduction:

By letter dated April 12, 1999, Alliant-IES Utilities Inc. submitted a request for revision of the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). The proposed amendment would revise TS SR 3.6.1.3.7.

2. Evaluation:

The amendment will relax the frequency of SR 3.6.1.3.7 from testing each EFCV every cycle (24 months) to testing a representative sample of EFCVs every cycle. In general, EFCVs have low failure rates, with zero failures to isolate at the DAEC. This high reliability and the low significance associated with an EFCV failure are the primary bases for this change as documented in a BWROG report included as Attachment 5 to this submittal. The instrument lines at DAEC include a flow restricting orifice upstream of the EFCVs to limit reactor water leakage in the event of a rupture. Previous evaluation of such an instrument line rupture (DAEC UFSAR 1.8.11), which the EFCVs are designed to mitigate, do not credit the isolation of the line by the EFCVs. Thus, a failure of an EFCV, though not expected as a result of this change, is bounded by the previous evaluation of an instrument line rupture. The radiation dose consequences of such a break are not impacted by this change.

The reduced testing associated with this change will result in potential dose savings during the outages in which the testing is performed. An increase in the availability of instrumentation during the outage and cost savings are also considered potential benefits from this change.

Therefore, we conclude that the proposed revision to the DAEC TS is acceptable.

ENVIRONMENTAL CONSIDERATION

10 CFR Section 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in a significant increase in individual or cumulative occupational radiation exposure. IES Utilities Inc. has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR Section 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1 to this letter, the proposed amendment does not involve a significant hazards consideration.

2. The proposed change reduces the number of Excess Flow Check Valves (EFCVs) that are tested on a cyclic basis. A representative sample (instead of every valve) of EFCVs will be tested each cycle pursuant to TS SR 3.6.1.3.7. Failures that occur within the sample population will be evaluated to determine if additional testing is warranted. Operating experience demonstrates a high reliability and a very low failure rate for these EFCVs. Therefore, there will be no significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. The proposed change will not change the way the EFCVs or the systems they are part of are operated. The EFCVs installed at the DAEC connect to the Reactor Coolant Pressure Boundary (RCPB). There are no EFCVs at DAEC that connect to primary containment atmosphere. The function provided by isolating an RCPB instrument line in the event of an instrument line break is not impacted by this change. This change does not impact the radiation dose results of a previous evaluation of an instrument line rupture. Therefore, there will be no significant increase in either individual or cumulative occupational radiation exposure.

ATTACHMENT 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 27, 1999

Mr. Eliot Protsch
President
IES Utilities Inc.
200 First Street, SE
P.O. Box 351
Cedar Rapids, IA 52406-0351

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON TECHNICAL SPECIFICATION CHANGE REQUEST REGARDING EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS AT DUANE ARNOLD ENERGY CENTER (TAC NO. MA05421)

Dear Mr. Protsch:

In a letter dated April 12, 1999, IES Utilities (the licensee) submitted a request to revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.7 to allow a representative sample of reactor instrumentation line excess flow control valves (EFCV) to be tested every 24 months, instead of testing each EFCV every 24 months.

The NRC staff has reviewed the licensee's submittal regarding the EFCV surveillance requirement for DAEC, and has determined that additional information is necessary to complete our review.

Your timely response to the enclosed request for additional information (RAI) will assist us in meeting your schedule. This RAI and the schedule have been discussed with Kenneth Putnam of your staff. If you have any questions regarding this issue, please contact me at your earliest convenience at 301-415-2020.

Sincerely,

Brenda Mozafari

Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure: As stated

cc w/encl.: See next page

REQUEST FOR ADDITIONAL INFORMATION

REGARDING EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS

FOR THE DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1. You have proposed a 10-year test interval for Excess Flow Check Valves (EFCVs), and have primarily referred to Option B of Appendix J to 10 CFR Part 50, as the model for doing this. However, you have neglected to address the fact that the NRC staff, through Regulatory Guide (RG) 1.163, limits containment isolation valve testing intervals to a maximum of 5 years. By licensees' requests, the RG has been incorporated by reference into the Technical Specifications (TS) of every plant that is using Option B of Appendix J. Thus, the 5-year interval is a requirement for every plant using Option B.

   Insofar as your justification for a 10-year test interval is, for the most part, that it is like Option B of Appendix J, provide additional justification for your proposed interval that is longer than the 5-year interval used for Option B of Appendix J.

2. Under the Appendix J, Option B, program, if a component on an extended test interval fails a test, it must be returned to the nominal test interval until subsequent testing re-establishes its reliable performance. In other words, if it doesn't continue to perform well, it gets tested more often. Your proposal has no similar well-defined feedback mechanism for EFCVs. There is only the following:

   > EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. (From the proposed DAEC Bases)

   The risk-informed IST Regulatory Guide, RG 1.175, also specifies the need for a feedback mechanism.

   Justify the absence from your proposal of an explicit, well-defined performance feedback mechanism that requires increased testing when valves fail their tests, or add such a mechanism to your proposal.

3. The proposed Duane Arnold TS says "a representative sample" of EFCVs will be tested every 2 years. The "representative sample" is not defined. Your proposed Bases, which, you are careful to point out, are not part of your proposed license amendment and are included for information only, say you will test 20% of the valves each refueling outage and thus test all of them in a 10-year period. In fact, the proposed TS would allow you to test less than 20% each time, and the concept of "representative" could change with time to exclude certain valves that were problems (e.g., repeat leakers, hard to access). The point is not that these things will actually happen, but that the proposed TS contain virtually no actual requirements.

   Justify the absence of more specific requirements in the proposed TS, or add specific requirements to the proposed TS.

October 5, 1999
NG-99-1358

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center (DAEC)
Docket No: 50-331
Op. License No: DPR-49
DAEC Response to Request For Additional Information on Technical Specification Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements

References: (1) NG-99-0308, Letter from J. Franz (IES Utilities) to NRC, dated April 12, 1999, "Technical Specification Change Request (TSCR-010): Relaxation of Excess Flow Check Valve Surveillance Testing."

(2) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 27, 1999, "Request For Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)."

File: A-107a, A-117

By reference (2), the NRC requested additional information regarding the Technical Specification Change Request (TSCR-010) we submitted to you in reference (1). Enclosed is the DAEC plant-specific response to your requested information. This response is not intended to modify the Boiling Water Reactor Owners' Group report, B21-00658-01, which was submitted to you in reference (1).

There are no new commitments made in this letter.

If you should have any further questions in this matter, please contact Ken Putnam at 319-851-7238.

Sincerely,

Kenneth E. Peveler
Manager, Regulatory Performance

Attachments:

(1) DAEC Response to Request for Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements.

(2) Industry/TSTF Standard Technical Specification Change Traveler, TSTF-334.

cc: J. Karrick
E. Protsch (w/o)
D. Wilson (w/o)
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
NRC Resident Office
Docu

DAEC Response to Request for Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements

NRC Question 1:

You have proposed a 10-year test interval for Excess Flow Check Valves (EFCVs), and have primarily referred to Option B of Appendix J to 10 CFR Part 50, as the model for doing this. However, you have neglected to address the fact that the NRC staff, through Regulatory Guide (RG) 1.163, limits containment isolation valve testing intervals to a maximum of 5 years. By licensees' requests, the RG has been incorporated by reference into the Technical Specifications (TS) of every plant that is using Option B of Appendix J. Thus, the 5-year interval is a requirement for every plant using Option B.

Insofar as your justification for a 10-year interval is, for the most part, that it is like Option B of Appendix J, provide additional justification for your proposed interval that is longer than the 5-year interval used for Option B of Appendix J.

DAEC Response to Question 1:

A 10-year test interval is not proposed in this amendment request. Rather, a 24-month nominal interval, testing a representative sample is proposed. The valves in question are of similar design, similar application, and similar service environment. Performance of the representative sample provides a strong indicator of the performance of the total population. The 10-year nominal interval solely limits the time between tests for any specific valve and provides additional assurance that all valves remain capable of performing their intended function.

It was not intended that the similarity to performance-based testing programs, such as Option B of Appendix J, form the primary basis of the change request. Rather, as stated in Reference (1), the basis for the amendment request is consistent with that described in a Boiling Water Reactor Owners' Group (BWROG) report, B21-00658-01. The failure rate data listed in Table 4-1 of that report, which includes zero failures in 25 years of operations at DAEC, is considered the primary basis for the performance-based interval. In addition, the consequences of a failure to isolate have been evaluated and found to be acceptable with respect to off-site doses.

RG 1.163 is essentially an NRC staff endorsement, with exceptions, of a Nuclear Energy Institute (NEI) document, 94-01, concerning the performance-based option of 10 CFR Part 50, Appendix J. Per RG 1.163, "Because of uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical Type C component performance data, and because of the indeterminate time period of three refueling cycles and insufficient precision of programmatic controls described in Section 11.3.2 [of NEI 94-01] to address these uncertainties, the guidance provided in section 11.3.2 for selecting extended test intervals greater than 60 months for Type C tested components is not presently endorsed by the NRC staff."

We believe the data provided in the BWROG report shows that bases for limiting intervals to 60 months, as stated in RG 1.163, are not applicable to EFCVs. Specifically:

- Unquantified leakage rates for test failures are not applicable because the maximum leakage through an unisolated instrument line is quantified as discussed in UFSAR 1.8.11. The dose consequences of the failure to isolate, as discussed in UFSAR 1.8.11, are acceptable.
- Repetitive/common-mode failures are not applicable as evidenced by the low industry failure rate and more specifically by the BWROG report, Table 4-2, "EFCV Failure Rates by Manufacturer."
- Aging effects are not a concern. The industry data already provided does not indicate any increase in failure rate with time in service.
- Historical performance data associated with EFCVs has been provided and is considered adequate to justify the proposed interval.
- There is no indeterminate time period involved with this proposed change. Every 24 months, approximately 20% of the total population (e.g. about 19 valves at DAEC) will be tested.

Therefore, we believe RG 1.163 and the 60-month limit for test intervals are not applicable to EFCV test intervals. The reference to Option B of Appendix J in the amendment was a general reference to performance-based testing. EFCVs are not subject to Type C leak rate testing at DAEC. It was not intended to adopt or imply adherence to the details of the Option B program. Rather, the reference to Option B was made from a conceptual viewpoint.

NRC Question 2:

Under the Appendix J, Option B, program, if a component on an extended test interval fails a test, it must be returned to the nominal test interval until subsequent testing re-establishes its reliable performance. In other words, if it doesn't continue to perform well, it gets tested more often. Your proposal has no similar well-defined feedback mechanism for EFCVs. There is only the following:

> EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. (From the proposed DAEC Bases)

The risk-informed IST Regulatory Guide, RG 1.175, also specifies the need for a feedback mechanism.

Justify the absence from your proposal of an explicit, well-defined performance feedback mechanism that requires increased testing when valves fail their tests, or add such a mechanism to your proposal.

DAEC Response to Question 2:

Any EFCV failure would be documented in the DAEC Corrective Action Program as a Surveillance Test failure. The failure would be evaluated and corrected. The Corrective Action Program is capable of trending EFCV test failures and determining whether additional testing is warranted.

Additionally, we have revised our 10 CFR 50.65 Maintenance Rule Performance Criteria to ensure EFCV performance remains consistent with the extended test interval. The new performance criterion is less than or equal to 1 failure per year on a 3 year rolling average.

When Performance Criteria are exceeded, the structures, systems or components (SSCs) in question are placed in Maintenance Rule 50.65(a)(1) status pending a problem review and the completion of correction actions. The problem review is undertaken via a root cause analysis performed in accordance with the plant's Corrective Action Program. Per the DAEC Maintenance Rule Program, this 50.65(a)(1) review must encompass the past three years of the SSC's performance history (at a minimum) and include discussion of other applicable problem history. Industry Operating Experience (OE) must also be considered. The 50.65(a)(1) review must also include a discussion of the cumulative and instantaneous effects upon plant safety of the problem(s) as determined using the Plant Safety Analysis, and an examination of current SSC monitoring, trending, preventive and predictive maintenance activities. Corrective actions are then established and Goals are set and monitored in accordance with the NEI guidance for implementation of the Maintenance Rule. The Performance Criteria Basis document containing the criteria for EFCVs also specifically notes that significant failures of equipment monitored by the document will be evaluated under the OE Program for dissemination to the industry.

NRC Question 3:

The proposed Duane Arnold TS says "a representative sample" of EFCVs will be tested every 2 years. The "representative sample" is not defined. Your proposed Bases, which, you are careful to point out, are not part of your proposed license amendment and are included for information only, say you will test 20% of the valves each refueling outage

and thus test all of them in a 10-year period. In fact, the proposed TS would allow you to test less than 20% each time, and the concept of "representative" could change with time to exclude certain valves that were problems (e.g. repeat leakers, hard to access). The point is not that these things will actually happen, but that that proposed TS contain virtually no actual requirements.

Justify the absence of more specific requirements in the proposed TS, or add specific requirements to the proposed TS.

DAEC Response to Question 3:

The term "representative sample," with an accompanying explanation in the TS BASES, is identical to current usage in the Standard TS (STS), NUREG-1433, Revision 1. Specifically, NUREG 1433 uses the term "representative" in TS Surveillance Requirement (SR) 3.8.6.3, in reference to battery cell testing and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. Therefore, the application of a "representative sample" for the EFCV testing SR, with its accompanying definition in the BASES is consistent with the STS usage.

In addition, as required by the Technical Specification Task Force (TSTF) process for changing the STS, a generic traveler (TSTF-334), with similar SR wording as that submitted in the DAEC plant specific submittal, has been submitted to the NRC for review (Attachment 2). One of the primary reviews conducted by the TSTF committee is conformance to the Writer's Guide for TS. There were no concerns raised over the content, format or proposed use of the BASES. This traveler was approved by the TSTF on May 6, 1999 and forwarded to the NRC for review on June 23, 1999. The only difference between the DAEC wording and that in the TSTF is our deletion of the clarifying details that the "representative sample" should be composed of various configurations, model types, sizes and operating environments. At the DAEC, EFCVs are all similar models, with similar configurations and operating environments. Therefore, the clarification in the generic traveler is not necessary to ensure that an appropriate "representative sample" is chosen at each SR interval.

The proposed TS BASES change was provided "for information only" so that it would not otherwise imply Staff approval of the BASES change upon issuance of the approved amendment. This is in conformance with 10 CFR 50.36 and Staff Policy. As stated in reference (1), the TS BASES are subject to the requirements of the Bases Control Program of TS 5.5.10. Therefore, the requirements of 10 CFR 50.59 apply to changes to the content of the subject TS Bases section and NRC approval would be required if such a change resulted in an Unreviewed Safety Question or required a TS change.

The BASES are routinely used to capture commitments imposed by the Staff as terms or conditions for approval of specific TS changes in their Safety Evaluation Reports (SERs). As noted in the generic traveler, the BASES change to evaluate any failures for possible expansion of the tested population is specifically characterized as a "commitment." We, therefore, do not agree that there are "no actual requirements" in the proposed amendment. As written, the proposed change is consistent with how other, similar testing programs that utilize a sampling approach are constructed in the STS. Thus, additional requirements within the TS proper are not needed.

Attachment 2 to NG-99-1358

(copy of TSTF-334)

BLIND CARBON COPY LIST FOR NG-99-1358
October 5, 1999

Rich Anderson	M. McDermott
G. Van Middlesworth	K. Peveler
D. Jantosik	D. Lausar
CTS Project	D. Curtland
R. McGee	

SUBJECT: Please find attached one copy of NRC correspondence responding to a request for additional information for our TS amendment request associated with relaxation of EFCV surveillance testing.

REFERENCE: NG-99-0308

FILE: A-107a, A-117

Industry/TSTF Standard Technical Specification Change Traveler

Relaxed Surveillance Frequency for Escess Flow Check Valve Testing

Classification: 3) Improve Specifications

NUREGs Affected: ☐ 1430 ☐ 1431 ☐ 1432 ☑ 1433 ☐ 1434

Description:

Surveillance Requirement 3.6.1.3.10 (NUREG-1433) requires verification of the actuation capability of each reactor instrumentation line Excess Flow Check Valve (EFCV) every [18] months. This proposed change is to relax the requirement to test every EFCV, by allowing a representative sample of EFCVs to be tested every [18] months, such that all EFCVs 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 (snubbers) and Option B to 10 CFR 50 Appendix J. As added assurance of detecting any potential common failure modes, the representative sample will be comprised of the various configurations, model types, sizes and operating environments of EFCVs in the plant.

Justification:

A review of industry operating experience demonstrates that EFCVs are highly reliable and that the incidence of test failures is extremely low. Given the large number of EFCVs that are currently required to be tested each Refuel Outage (typically 100), a significant cost and dose savings can be achieved by the proposed relaxation of the testing frequency without any reduction in overall safety or reliability. The Bases change includes a commitment to evaluate any failure to isolate for the need to expand the tested population in that test interval.

(Note: Some plants may require an Inservice Testing Program Relief Request pursuant to 10 CFR 50.55a in order to implement this proposed change.)

Industry Contact:	Ford, Bryan	(601) 437-6559	bford@entergy.com
NRC Contact:	Giardina, Bob	301-314-3152	lbb1@nrc.gov

Revision History

OG Revision 0 **Revision Status: Active** **Next Action:**

Revision Proposed by: Duane Arnold

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 09-Feb-99

Owners Group Comments
(No Comments)

Owners Group Resolution: Approved Date: 09-Feb-99

TSTF Review Information

TSTF Received Date: 16-Mar-99 Date Distributed for Review

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 06-May-99

OG Revision 0 **Revision Status: Active** **Next Action:**

Incorporation Into the NUREGs

File to BBS/LAN Date: TSTF Informed Date: TSTF Approved Date:

NUREG Rev Incorporated:

Affected Technical Specifications

SR 3.6.1.3.10	PCIVs
SR 3.6.1.3.10 Bases	PCIVs

TSTF-334

Insert 1 to BASES for SR 3.6.1.3.10

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 3.6.1.3.10

The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. 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.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.7	-----------------NOTE----------------- Only required to be met in MODES 1, 2 and 3. -------------------------------------- Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days **AND** Once within 92 days after opening the valve
SR 3.6.1.3.8	Verify the isolation time of each MSIV is ≥ [2] seconds and ≤ [8] seconds.	In accordance with the Inservice Testing Program or 18 months
SR 3.6.1.3.9	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	[18] months
SR 3.6.1.3.10	Verify ~~each~~ a representative sample of reactor instrumentation line EFCVs actuates [on a simulated instrument line break to restrict flow to ≤ 1 gph].	[18] months
SR 3.6.1.3.11	Remove and test the explosive squib from each shear isolation valve of the TIP System.	[18] months on a STAGGERED TEST BASIS

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9

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 SR 3.3.6.3.7 overlaps this SR to provide complete testing of the safety function. The [18] 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 has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.10

a representative sample of

This SR requires a demonstration that ~~each~~ reactor instrumentation line excess flow check valves (EFCV) is OPERABLE by verifying that the valve [reduces flow to ≤ 1 gph on a simulated instrument line break]. INSERT 1 This SR provides assurance that the instrumentation line EFCVs will perform so that predicted radiological consequences will not be exceeded during the postulated instrument line break event evaluated in Reference 6. The [18] 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. INSERT 2 ~~Operating experience has shown that these components usually pass this Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

(S)

SR 3.6.1.3.11

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

(continued)

ATTACHMENT 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 30, 1999

Mr. Eliot Protsch
President
IES Utilities Inc.
200 First Street, SE
P.O. Box 351
Cedar Rapids, IA 52406-0351

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON TECHNICAL SPECIFICATION CHANGE REQUEST REGARDING EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS AT DUANE ARNOLD ENERGY CENTER (TAC NO. MA05421)

Dear Mr. Protsch:

In a letter dated April 12, 1999, IES Utilities Inc. submitted a request to revise Duane Arnold Energy Center (DAEC) Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.7 to allow a representative sample of reactor instrumentation line excess flow control valves (EFCV) to be tested every 24 months, instead of testing each EFCV every 24 months.

The NRC staff has reviewed your submittal regarding the EFCV surveillance requirement for DAEC, and has determined that additional information is necessary to complete our review.

Your timely response to the enclosed request for additional information (RAI) will assist us in meeting your schedule. This RAI and the schedule have been discussed with Kenneth Putnam of your staff. If you have any questions regarding this issue, please contact me at your earliest convenience at 301-415-2020.

Sincerely,

Brenda Mozafari

Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure: As stated

cc w/encl: See next page

RECEIVED
OCT 1 1
DAEC

NUCLEAR LICENSING
Resp. Due Yes*
Post. Req'd NO
File: ____

VP-Nuc. Div.
Asst. VP Nuc.
Plant Mgr.
Mgr.-Reg. Perf.
Mgr.-Eng.
Mgr.-EP
Mgr.-Training
Mgr.-Business
Mgr.-Out. Supp.
Mgr.-QA
CIPCO
Combelt
GDS
A. Gutterman

AR Coord.
Security
Rad. Pro.
Maint.
Operations
Proj. Engr.
Syst. Engr.
Prog. Engr.
Licensing
Safety Comm
CTS
TMAR Coord.
Mat. Mgmt.
OPS SUPPO.
[illegible]

* Response Completed 99-1383

REQUEST FOR ADDITIONAL INFORMATION
REGARDING EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS
FOR THE DUANE ARNOLD ENERGY CENTER
DOCKET NO. 50-331

1. Explain the discrepancy between page 11, Section 4.2, top paragraph that states "...a total of nine failures over 10,000 valve years of operation" and Table 4.1 on page 14 that indicates 11 failures.

2. Refer to page 12, Section 4.3, top paragraph. The single instrumentation line break frequency of 5.34E-6/year assumed was based on WASH-1400 data. Explain why a more updated value was not used. Individual Plant Examination data indicate that such frequency could be higher.

   EFCV unavailability used the lambda T over two formula. Provide the basis for assuming a constant failure rate for 10 years. Explain how the nature of "stickiness" might change over such a long period (10 years) with potentially new failure mechanisms becoming dominant.

   Describe the impact/change on the release frequency estimate if

   (1) a more updated instrumentation line break frequency and
   (2) a constant failure rate is not assumed.

3. Verify if there are valves in the plant that are similar to EFCVs whose failure data may be available. If such data exist, provide the data as well as the impact of applying such data on the release frequency estimate.

   In addition, ensure that you have considered in your analysis any information available on degradation mechanism(s) and root cause(s) of the failed EFCVs (or similar valves) observed at other plants. Similarly, provide assurance that this type of information (including failure rates) will be shared among the plants for future data as they become updated and available.

   Provide performance criteria for EFCVs. Describe how a cause determination will be performed and determine what specific corrective action would be taken if EFCVs do not meet their performance criteria.

Enclosure

October 8, 1999
NG-99-1383

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center (DAEC)
Docket No: 50-331
Op. License No: DPR-49
DAEC Response to Request For Additional Information on Technical Specification Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements

References: (1) NG-99-0308, Letter from J. Franz (IES Utilities) to NRC, dated April 12, 1999, "Technical Specification Change Request (TSCR-010): Relaxation of Excess Flow Check Valve Surveillance Testing."

(2) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 30, 1999, "Request For Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)."

File: A-107a, A-117

By Reference (2), the NRC requested additional information regarding the Technical Specification Change Request (TSCR-010) we submitted to you in Reference (1). Enclosed is the DAEC response to your request. This response is not intended to modify the Boiling Water Reactor Owners' Group report, B21-000658-01, which was submitted to you in reference (1).

There are no new commitments made in this letter.

If you should have any further questions in this matter, please contact Ken Putnam at 319-851-7238.

Sincerely,

Kenneth E. Peveler
Manager, Regulatory Performance

Attachment:

DAEC Response to Request for Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements.

cc: J. Karrick
E. Protsch (w/o)
D. Wilson (w/o)
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
NRC Resident Office
Docu

DAEC Response to Request for Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements

NRC Question 1:

Explain the discrepancy between page 11, Section 4.2, top paragraph that states "...a total of nine failures over 10,000 valve years of operation" and Table 4.1 on page 14 that indicates 11 failures.

DAEC Response to Question 1:

It is believed this discrepancy is a typographical error in that Table 4.1 was updated with data that was not subsequently incorporated back into the text in Section 4.2. Eleven is the correct number for the Table and the text, and is the number used in the composite failure rates in Table 4.1.

NRC Question 2:

Refer to page 12, Section 4.3, top paragraph. The single instrumentation line break frequency of 5.34E-6/year assumed was based on WASH-1400 data. Explain why a more updated value was not used. Individual Plant Examination data indicate that such frequency could be higher.

EFCV unavailability used the lambda T over two formula. Provide the basis for assuming a constant failure rate for 10 years. Explain how the nature of "stickiness" might change over such a long period (10 years) with potentially new failure mechanisms becoming dominant.

Describe the impact/change on the release frequency estimate if

(1) a more updated instrumentation line break frequency and
(2) a constant failure rate is not assumed.

DAEC Response to Question 2:

The line break frequency calculated in the GE topical report for a single instrument line is based on a break failure rate of 6.1E-12 per hour per foot of line, and a conservatively assumed average pipe length of 100 feet (6.1E-12/hr-ft * 8760 hrs/yr * 100 ft = 5.34E-6 breaks/yr). The value of 6.1E-12 per hour per foot is from WASH-1400 and is applicable to small pipe. WASH-1400 "Reactor Safety Study: An Assessment of Accident Risks in

U.S. Commercial Nuclear Power Plants," was published in 1974 and therefore had a limited amount of nuclear power plant operating experience from which to base its component failure rate data. In fact, much of its data was drawn from non-nuclear facilities. More recent pipe failure rate data is published in EPRI Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants", July 1992. This report compiles failure data from approximately 1000 years of nuclear plant operating experience.

The smallest pipe size considered in the EPRI report is 1/2 inch to 2 inch diameter pipe. Failure rate data for this class of piping will be considered representative of the subject instrument line piping. Also, failure rate data is calculated and reported on a "per section" basis rather than a "per foot" basis. (This unit of measure was chosen because the influence of welds, and their adjacent heat-affected zones on the failure rates, is by far greater than the influence of length.) A pipe section is defined to be a segment of piping between major discontinuities such as valves, pumps, reducers, tees, etc.

Table 4.4-2 of EPRI TR-100380 contains recommended pipe rupture failure rates based on reactor type (Westinghouse, Babcock & Wilcox, Combustion Engineering, and General Electric) and system. The rate for reactor coolant piping in General Electric BWRs is judged to be most representative of the subject instrumentation lines. The recommended average value representing all pipe sizes in this category is 6.7E-10 failures per section per hour. A multiplier of 1.2 (derived in Section 4.4.10.2) is applied to this value to obtain the failure rate for small pipe.

1.2 * 6.7E-10/hr-section = 8.04E-10 failures per hr per section

If a typical instrument line is assumed to contain five sections (ref. UFSAR Figure 3.2-2), its rupture failure rate is:

5 sections * 8.04E-10/hr-section * 8760 hrs/yr = 35.2E-06 failures per year

This value is 6.6 times greater than the value of 5.34E-06/yr calculated in the GE Topical Report using data from WASH-1400.

The GE Topical Report determines an upper limit EFCV failure rate based upon eleven observed failures in 1.09E+08 hours of service. It can be postulated that the failure rate for EFCVs is not constant over time, but may in fact increase over time due to age related factors.

Alliant Energy is not currently aware of any study that explores causes of EFCV failures, or that characterizes change in EFCV failure rate over time. However, even if the number of observed failures is conservatively assumed to be five times that of the actual observed number, the resulting calculated upper limit EFCV failure rate would still be acceptably small.

The formula for upper limit failure rate used in the GE Topical Report is:

$$\lambda_U = \frac{1}{2T}\chi^2_{\alpha:2r+2}$$

Where:

T is the operating time in hours

r is the number of failures

$\chi^2_{\alpha:2r+2}$ is the value taken from chi-square distribution tables which corresponds to 2r+2 degrees of freedom and 0.95 confidence level (a = 1-0.95 = 0.05)

For eleven observed valve failures, degrees of freedom is 24. The value of χ^2 for 24 degrees of freedom and a 95% confidence level is 36.415. Therefore,

$$\lambda_U = \left[\frac{1}{2*1.09E+8}\right]*36.415 = 1.67E\text{-}07 \text{ failures per hour}$$

For fifty-five observed valve failures (five times normal), degrees of freedom is 112. Chi-squared values are not typically provided for degree of freedom values above thirty because for large values, the chi-squared distribution is close to that of the standard normal distribution. In this case, χ^2 is approximated by:

$$\chi^2 = \frac{1}{2}\left[x_\alpha + \sqrt{2n-1}\right]^2$$

Where: x_α is the α-point of the standard normal distribution

n is the degrees of freedom

(Ref. CRC Standard Mathematical Tables, 18th Edition)

For a 0.95 confidence level (a = 0.05), x_α is 1.645.

$$\chi^2 = \frac{1}{2}\left[1.645 + \sqrt{(2*112)-1}\right]^2 = 137.42$$

Therefore, if the number of observed valve failures is assumed to be fifty-five, the upper limit valve failure rate would be

$$\lambda_U = \left[\frac{1}{2 * 1.09\text{E} + 8}\right] * 137.42 = 6.30\text{E-}07 \text{ failures per hour}$$

New release frequency values can be calculated from these higher values for instrument line break frequency and EFCV failure rate. Using the GE notation,

$$RF = I * A$$

and,

$$A = \lambda_U \frac{\theta}{2}$$

Where,

RF	is release frequency per year
I	is instrument line break frequency per year
A	is EFCV unavailability (failure to close probability)
λ_U	is EFCV failure rate per hour
θ	is EFCV surveillance test interval in hours

Using a surveillance interval of two years, an instrument line break frequency of 35.2E-06 per year, and an EFCV failure rate of 6.30E-07 failures per hour,

$$RF = I * \lambda_U * \left(\frac{\theta}{2}\right) = \left(\frac{35.2\text{E}-06}{\text{yr}}\right)\left(\frac{6.30\text{E}-07}{\text{hr}}\right)\left(\frac{2 \text{ yrs} * 8760 \text{ hrs/yr}}{2}\right) = 19.4\text{E}-08 \text{ events per year}$$

Inserting a surveillance interval of ten years,

$$RF = \left(\frac{35.2\text{E}-06}{\text{yr}}\right)\left(\frac{6.30\text{E}-07}{\text{hr}}\right)\left(\frac{10 \text{ yrs} * 8760 \text{ hrs/yr}}{2}\right) = 97.1\text{E}-08 \text{ events per year}$$

Corresponding release frequencies reported in the GE Topical Report are 0.78E-08 events per year for two year surveillance intervals and 3.91E-08 events per year for ten year surveillance intervals (Table 4-3).

The DAEC has 94 instrument lines affected by this technical specification change request. For two year surveillance intervals, the total release frequency of instrument line breaks with failure to isolate is,

$$RF_{plant} = 94 * 19.4E\text{-}08/yr = 1.82E\text{-}05 \text{ events per year}$$

For ten year surveillance intervals,

$$RF_{plant} = 94 * 97.1E\text{-}08/yr = 9.13E\text{-}05 \text{ events per year}$$

These values are sufficiently low that it can be concluded that a change in surveillance test frequency has minimal impact on the valve reliability.

The impact of an increased estimation of instrument line rupture frequency and a five-fold increase in assumed number of EFCV failures on the likelihood of a release to the reactor building environs has been calculated. The total plant release frequency for a rupture of any of the DAEC's 94 instrument lines and a coincident failure of the line's EFCV to isolate the break flow is 9.13E-05 events per year, which is equivalent to approximately one event in ten thousand years. These results confirm that release frequencies will remain extremely low for the DAEC and that the conclusions of the GE report are applicable. The conclusion that releases would be infrequent remains valid even with significantly different assumptions on break frequency and valve failure rates.

The request concerning the nature of valve "stickiness" is addressed in the following response to Question 3.

NRC Question 3:

Verify if there are valves in the plant that are similar to EFCVs whose failure data may be available. If such data exist, provide the data as well as the impact of applying such data on the release frequency estimate.

In addition, ensure that you have considered in your analysis any information available on degradation mechanism(s) and root cause(s) of the failed EFCVs (or similar valves) observed at other plants. Similarly, provide assurance that this type of information (including failure rates) will be shared among the plants for future data as they become updated and available.

Provide performance criteria for EFCVs. Describe how a cause determination will be performed and determine what specific corrective action would be taken if EFCVs do not meet their performance criteria.

DAEC Response to Question 3:

The design of the EFCVs to which the TS and the proposed amendment apply are unique to the rest of the plant. There are no other applications of these valves at DAEC. The design of the EFCVs are quite different when compared to typical swing-check or disc-type check valves. As a result of these design differences, EFCVs are not generally susceptible to the failure mechanisms of a standard check valve, such as corrosion, flow-induced wear, or age-related component degradation.

In response to your request for other valve data, and in addition to the nuclear industry data already provided in the BWROG report, the vendor for DAEC, Marotta Scientific Controls, Inc., was contacted to provide any available failure data for the Flow Fuse model number FVL16F valves that are installed at DAEC. The vendor representative provided applicable conclusions from a Naval Coastal System Laboratory(NCSL) study (NCSL 174-73) entitled, "Test and Evaluation of Flow Fuses for Use in Manned Pressure Chambers." The following abstract is provided from that report:

> "*Manned pressure chambers used in sea or shore diving operations are subject to catastrophic depressurization from external piping failures. Recently, a device designed to permit bidirectional fluid flow while protecting "flow out" lines has become available. This device, called a Flow Fuse, is essentially a flow sensitive check valve with the poppet spring-loaded open.*
>
> *Offering promise of rapid reliable flow shut-off but lacking substantiating use or test data, fuses were procured and performance evaluated. A test piping system capable of simulating line ruptures and sudden large increasing leaks was constructed and each of four fuse sizes were tested (1/4", ½", 1" and 2" line sizes). Data on closure flows, differential pressures, and closure speeds were collected and analyzed. Tests indicated rapid closure (10-100 milliseconds) when flow attempted to exceed the trip point settings. There were no failures during more than 10,000 total actuations on the four fuses. All fuses have adjustable trip points, easy serviceability, and should provide excellent resistance to corrosive environments.*
>
> *It is concluded that the Flow Fuse type device can provide an increase in safety for manned pressure chambers and that sufficient data has been collected to demonstrate adequacy of design and performance for certification of material adequacy.*"

Additionally, under the heading of "Wear Effects," the NCSL report further states:

> "*A disassembly and visual inspection also failed to reveal any significant wear. The Flow Fuses tested were constructed of high durability, low corrosion susceptibility materials and should be extremely long lasting...*"

Lastly from the NCSL report, and perhaps more importantly, under the heading of Life-Cycle Tests:

"These tests were designed to determine the ability of the fuse to operate within its performance envelope after repeated actuations, and to a limited extent, the operating
life expectancy of the fuse....The life-cycle tests were made immediately following the single-cycle performance envelope tests. Because the single-cycle tests require a few hundred cycles (300-400), the fuse was well "broken in" prior to commencement of the life-cycle tests. By scheduling the tests in this order, effects or trends due to "wear in" should have been avoided.

The life-cycle tests involved 2000 total acutations (cycles) and began with base-line data being collected on the fuse. The fuse was next subjected to 500 nonstop cycles and then rechecked for operation. This procedure continued until 2000 cycles were completed. The results are shown in Table B5....No trends toward degraded operations are detectable and it seems most reasonable to assume that the "cycles to failure" of this fuse is considerably greater than 2000."

These excerpts from the NCSL report are applicable to and support the BWROG report declaration of the very high reliability of the EFCVs and is provided to further support the requested TS change in test frequency. Because this additional data does not indicate higher failure rates than previously assumed and because our response to Question 2 included a five-fold increase in failure rate, additional impact on release frequency estimate was not done.

In regards to any potential concern (Question 2, "Stickiness") with EFCV performance with time as a result of the proposed test interval, no such data was available. However, the EFCV vendor manual, "Technical Manual For Excess Flow Check Valve Model No. FVL16F Part No. 280837 Revision G," states "Under normal operating conditions, the valve does not require maintenance of any kind." Conversations with the vendor representative likewise did not reveal a need to exercise the valve or any concern with performance at a reduced test frequency.

Information on degradation mechanisms and root cause(s) at other plants was included in the BWROG report, on page 25, in the Table entitled "Testing Data." A large portion of the test failures were caused by test methodology and not the actual valves, as stated in the notes to the table.

Sharing of data from any future failures would be through the Equipment Performance and Information Exchange (EPIX) system, as applicable. Also, failures of EFCVs to close when required during an event would most likely be reported under 10 CFR50.73, the Licensee Event Report system.

Any EFCV failure would be documented in the DAEC Corrective Action Program as a Surveillance Test failure. The failure would be evaluated and corrected. The Corrective

Action Program is capable of trending EFCV test failures and determining whether additional testing is warranted.

Additionally, we have revised our 10 CFR 50.65 Maintenance Rule Performance Criteria to ensure EFCV performance remains consistent with the extended test interval. The new performance criterion is less than or equal to 1 failure per year on a 3 year rolling average.

When Performance Criteria are exceeded, the structures, systems or components (SSCs) in question are placed in Maintenance Rule 50.65(a)(1) status pending a problem review and the completion of correction actions. The problem review is undertaken via a root cause analysis performed in accordance with the plant's Corrective Action Program. Per the DAEC Maintenance Rule Program, this 50.65(a)(1) review must encompass the past three years of the SSC's performance history (at a minimum) and include discussion of other applicable problem history. Industry Operating Experience (OE) must also be considered. The 50.65(a)(1) review must also include a discussion of the cumulative and instantaneous effects upon plant safety of the problem(s) as determined using the Plant Safety Analysis, and an examination of current SSC monitoring, trending, preventive and predictive maintenance activities. Corrective actions are then established and Goals are set and monitored in accordance with the NEI guidance for implementation of the Maintenance Rule. The Performance Criteria Basis document containing the criteria for EFCVs also specifically notes that significant failures of equipment monitored by the document will be evaluated under the OE Program for dissemination to the industry.

BLIND CARBON COPY LIST FOR NG-99-1383
October 8, 1999

Rich Anderson	M. McDermott
G. Van Middlesworth	K. Peveler
D. Jantosik	D. Lausar
CTS Project	D. Curtland
R. McGee	

Subject: Duane Arnold Energy Center (DAEC)
Docket No: 50-331
Op. License No: DPR-49
DAEC Response to Request For Additional Information on Technical Specification Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements

References: (1) NG-99-0308, Letter from J. Franz (IES Utilities) to NRC, dated April 12, 1999, "Technical Specification Change Request (TSCR-010): Relaxation of Excess Flow Check Valve Surveillance Testing."

(2) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 30, 1999, "Request For Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)."

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GE Nuclear Energy

175 Curtner Avenue
San Jose, CA 95125

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Excess Flow Check Valve Testing Relaxation

Prepared for:

Excess Flow Check Valve Committee
BWR Owners' Group

Prepared by: P.D. Knecht
C.C. Eftimie
R.M. Fairfield

Project Manager: original signed by Rick Hill
R.A. Hill

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Excess Flow Check Valve Testing Relaxation

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Summary of Results

This report reviews the licensing requirements, operational experience and consequences associated with testing requirements for Excess Flow Check Valves (EFCV) in instrument lines connected to the Reactor Coolant Pressure Boundary (RCPB). Testing of EFCVs in instrument lines connected to the containment atmosphere is not required. The purpose of the report is to explore strategies for eliminating or extending the testing interval for EFCVs. This would provide a benefit to all BWRs in the form of reduced outage time, occupational exposure and associated costs.

The review concludes that the safety significance of EFCVs is extremely small and that they need not be addressed by plant Technical Specifications. The review also concludes that demonstrated experience of valve reliability, coupled with low consequences of excess flow check valve failure, provide justification for extending the test interval up to once in ten years. However, elimination of testing is not recommended, since periodic testing provides assurance of reliable functioning of the valves to provide an effective means of protection of the working environment if an instrument line break were to occur.

Excess Flow Check Valve Testing Relaxation

1. Background

EFCVs are utilized in BWR containments to limit the release of fluid in the event of an instrument line break. Table 1-1 lists typical instrument functions for lines containing EFCVs. Mark III plants in general do not contain as many EFCVs because many of the instrument racks are located within the accessible containment building.

Table 1-1 Typical RCPB EFCV Applications

Instrument	Typical # of EFCVs	Environment Sensed
RPV Level/Pressure	16	RPV
RPV Head Pressure	1	RPV
Main Steam Line Flow	16	RPV
Core Plate dP	3	RPV
Recirculation Pump Seal Pressure	4	RPV
Recirculation Pump Suction Pressure	2	RPV
Recirculation Flow	24	RPV
Recirculation Discharge	8	RPV
Recirculation Pump Differential Pressure	4	RPV
RCIC Steam Line Flow	4	RPV
HPCI Steam Line Flow	4	RPV
Total	86	

EFCVs in instrument lines which connect to the RCPB are normally tested during refueling outages to meet Technical Specification requirements. Instrument lines that connect to the containment atmosphere, such as those which measure drywell pressure, or monitor the containment atmosphere or suppression pool water level, are considered extensions of primary containment. A failure of one of these instrument lines during normal operation would not result in the closure of the associated EFCV, since normal operating containment pressure is not sufficient to operate the valve. Such EFCVs will only close with a downstream line break concurrent with a LOCA. Since these conditions are beyond the plant design basis, EFCV closure is not needed and containment atmospheric instrument line EFCVs need not be tested.

Testing can require several hundreds of manhours to complete and can be a critical path activity during some outages. Testing typically requires the reactor to be pressurized to normal operating pressure. EFCVs are generally tested by opening an instrument drain valve, and observing valve closure by either direct indication (valve position indication), or by a combination of indirect indications (audible sounds, pressure, temperature, level, or flowrate). Attachment A provides the results of a survey of BWR plant-specific design and operating practices associated with EFCVs.

EFCVs are made by several manufacturers. Some are simple self-actuating ball check valves, with allowable leakage rates of up to 3 gpm when closed. Others are more sophisticated devices, with remote position indication and a solenoid-operated reset. EFCVs are not required to be leak tight.

The purpose of this report is to address issues associated with extending the EFCV testing interval.

Current Impact of Testing

The benefits of EFCV test relaxation lie in reduced cost of labor during outages and reduction in outage lengths without significantly impacting the risk to the general public. An estimate of the impacts of EFCV testing made at one BWR is shown in Table 1-2. Note that at this BWR some of the EFCVs were bench tested during the outage rather than in place (in-situ). Bench testing increases the labor and decreases the exposures associated with EFCV testing.

Table 1-2 EFCV Impact

Activity	Labor (Man-hours)	Exposure (Man-Rem)	Critical Path Time (hours)
Valving in/out	38	0.75	
In-situ Testing	92	0.6	
Bench Testing	224	0	
Instrument Line Refill	38	0.75	
Total	**392**	**2.1**	**16**

The utility estimated that EFCV testing was costing them about $125,000/year (averaged over a 24 month cycle). In addition, a certain personnel risk occurs due to the need to test the valves.

Planned Approach

The planned approach to reduce the exposures, costs and outage impact resulting from EFCV testing is to justify a less frequent EFCV testing interval than is currently specified in Technical Specifications. The planned approach provides a justification (Section 2) for relocation of the EFCV testing requirements from Technical Specifications to administrative documents such as the Technical Requirements Manual (TRM). This conclusion is supported by offsite dose and reliability evaluations contained in Sections 3 and 4. Relaxation of the testing interval relies upon plant specific submittals based on Option B of 10CFR50, Appendix J.

2. Tech Specs and Licensing Basis

2.1. Technical Specifications

Each BWR Technical Specification is unique, but has certain similarities to other Technical Specifications. BWR/4 Standard Specifications (Reference 1) Section 3.6.3 requires that EFCVs be operable during operating conditions with the reactor pressure above atmospheric. Surveillance requirement 4.6.3.4, originally based on the fuel cycle length, specifies that operability be demonstrated once per 18 months.

Improved BWR/4 Technical Specifications (Reference 2) provide relaxed restoration times (from 4 hours to 12 hours) if an EFCV is found to be inoperable, but still require testing at an 18-month interval. SR 3.6.1.3.10 states:

> *"Verify each reactor instrumentation line EFCV actuates [on a simulated instrument line break to restrict flow to < 1 gph]."* [1]

The specified interval is *"[18] months"*. The NRC provides the basis for the recommended interval in Reference 2 as follows:

> *"The [18] 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 [18] month frequency. Therefore, the frequency was concluded to be acceptable from a reliability standpoint."*

Reference 3 provides the criteria in Table 2-1 for establishing a Technical Specification Limiting Condition for Operation (LCO). If an LCO is not needed, any surveillance testing associated with that LCO similarly would not be needed.

[1] Information included in brackets [ ] represent plant-specific information

Table 2-1 LCO Criteria

	Requirement	Comment
Criterion 1	*Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.*	EFCVs do not satisfy this criterion.
Criterion 2	*A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*	EFCVs do not satisfy this criterion.
Criterion 3	*A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*	EFCV closure is not needed for accident mitigation (Section 3).
Criterion 4	*A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.*	EFCV closure is not significant to public health and safety (Section 4).

From Table 2-1, and as discussed in later sections of this report, it is evident the EFCVs do not meet any of the criteria for having a LCO included in Technical Specifications. Consequently, justification exists for relocating any testing requirements from the Technical Specifications. It can be noted that the improved BWR/6 Standard Technical Specifications (Reference 4) do not currently contain reference to EFCVs.

2.2. Licensing Bases

2.2.1. General Design Criteria

General Design Criteria (GDC) 55 and 56 contained in 10CFR50, Appendix A (Reference 5), provide design requirements for isolation of lines that penetrate the primary containment. Instrument lines which monitor the RPV or containment internal conditions are subject to isolation requirements, but as noted in the GDCs, *"unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis"*. An alternate licensing basis acceptable to the NRC for isolation of instrument lines connected to the RCPB is described in Regulatory Guide 1.11 (Reference 6). This is described in Section 2.2.2.

2.2.2. Regulatory Guide 1.11 (Reference 6)

Instrument lines constitute closed, extended containment boundary system piping outside containment. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," accepts instrument lines as extensions of primary containment, and allows that their configuration satisfies the "other defined basis" requirements of GDC 55 and 56. Automatic isolation of instrument lines during a LOCA is not prudent, since these instrument lines provide safety functions for reactor protection and containment isolation which need to be operable during a LOCA.

Regulatory Guide 1.11 design requirements are summarized in Table 2-2.

Table 2-2 Regulatory Guide 1.11 Requirements

Line Type	Requirement	Comment with regards to EFCVs
Lines that are part of the protection system	*a) Should satisfy the requirements for redundancy, independence, and testability of the protection system*	Instrument lines generally comply with this requirement. This requirement is not applicable to EFCV testing.
Lines that are part of the protection system and Lines that are not part of the protection system	*b) Should be sized or orificed to assure that in the event of a postulated failure of the piping or of any component (including the postulated rupture of any valve body) in the line outside primary reactor containment during normal reactor operation,* *(1) the leakage is reduced to the maximum extent practical consistent with other safety requirements,* *(2) the rate and extent of coolant loss are within the capability of the reactor coolant makeup system,* *(3) the integrity and functional performance of secondary containment, if provided, and associated safety systems (e.g., filters, standby gas treatment system) will be maintained, and* *(4) the potential offsite exposure will be substantially below the guidelines of 10CFR100.*	This requirement is not applicable to EFCV testing. Item b) (3) may not be satisfied if the EFCVs fail to function.(Section 3.2). Section 3 and Attachment B demonstrate that the designs generally meet the intent of this requirement.

Lines that are part of the protection system and Lines that are not part of the protection system	*c) Should be provided with an isolation valve capable of automatic operation or remote operation from the control room or from another appropriate location, and located in the line outside the containment as close to the containment as practical. <u>There should be a high degree of assurance that this valve:</u>* *(1) will not close accidentally during normal reactor operation,* *(2) will close or be closed if the instrument line integrity outside containment is lost during normal reactor operation or under accident conditions, and* *(3) will reopen or can be reopened under the conditions that would prevail when valve reopening is appropriate. Power operated valves should remain as-is upon loss of power. The status (opened or closed) of all such isolation valves should be indicated in the control room. If a remotely operable valve is provided, sufficient information should be available in the control room or other appropriate location to assure timely and proper actions by the operator.*	EFCVs generally comply with this requirement. Reliability of EFCVs is discussed further in Section 4.
Lines that are part of the protection system and Lines that are not part of the protection system	*d) Should be conservatively designed up to and including the isolation valve and of a quality at least equivalent to the containment. These portions of the lines should be located and protected so as to minimize likelihood of their being damaged accidentally. They should be protected or separated to prevent failure of one line from inducing failure of any other line. Provisions should be included to permit periodic visual inservice inspection, particularly of those portions of the lines outside containment up to and including the isolation valve.*	Instrument line separation generally addresses this requirement. This requirement is not applicable to EFCV testing.
Lines that are part of the protection system and Lines that are not part of the protection system	*e) Should not be so restricted by components in the lines, such as valves and orifices, that the response time of the connected instrumentation will be increased to an unacceptable degree.*	Instrument line orificing generally does not impact the performance of the instrumentation. This requirement is not applicable to EFCV testing.

As shown in Table 2-2, the design criteria of RG 1.11 are generally met by the current BWR designs. It should be noted that with the exception of testing *implied* by RG 1.11 item c, no specific testing requirements are defined by the regulatory guide. The discussion portion of the regulatory guide states:

> *"Sufficient experience with valves of a similar type should be available to assure a high probability that the valve will not close when the instrument line is intact and its safety function is required, but that it will close if the instrument line is ruptured downstream".*

The performance of EFCVs discussed in Section 4 provide this high degree of assurance.

2.2.3. 10CFR50 Appendix J (Reference 7)

A Containment Isolation Valve (CIV) is defined in Appendix J as *"any valve which is relied upon to perform a containment isolation function."* Appendix J prescribes air-testing requirements for containment isolation valves, and provides for exemptions for valves which are water sealed. Most EFCVs are connected to water-filled systems, and are tested for operability with water.

10CFR50 Appendix J testing is only applicable to EFCVs if they perform a containment isolation function. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post-LOCA conditions. As discussed in Section 3.1, the functioning of EFCVs is not necessary to remain within 10CFR100 limits.

Consequently, for purpose of 10CFR50, Appendix J, CIV testing, EFCVs do not provide a containment isolation function and are exempt from consideration under Appendix J.

2.2.4. ASME OM-10, Subsection ISTC (Reference 8)

ASME OM-10, Subsection ISTC, *"Inservice Testing of Valves in Light-Water Reactor Power Plants"* (Reference 8), establishes the requirements for inservice testing of valves in light-water reactor nuclear power plants. Testing is required for valves *"required to perform a specific function in shutting down a reactor to the cold shutdown condition, in maintaining the cold shutdown condition, or in mitigating the consequences of an accident".*

EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with a design basis LOCA would be of sufficiently low probability to be outside of the design basis. Furthermore, following a design basis LOCA, isolation is not necessary to achieve acceptable consequences of an accident (Section 3.1). Therefore, because EFCVs do not perform the functions for which the ISTC applies (i.e., they are not needed to mitigate an accident), EFCV testing is not required by the ASME code. Consequently, OM-10 is not considered to be part of the licensing basis for EFCV testing.

2.2.5. Other Licensing Bases

10CFR50 Appendix A, General Design Criterion (GDC) 55, provides the isolation requirements for lines penetrating containment that are connected to the reactor coolant pressure boundary. The NRC has allowed exceptions to the GDC in previous evaluations in a similar manner to that provided through Regulatory Guide 1.11. Another similar evaluation has been established for the CRD withdrawal lines.

NUREG-0803, "*Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping*", was issued in response to draft NUREG-0785, "*Safety Concerns Associated With Pipe Breaks in the BWR Scram System*". In response to NUREG-0785, GE Nuclear Energy prepared NEDO-24342, "*GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks*". In NEDO-24342, GE Nuclear Energy contended that the CRD withdrawal lines are small diameter (3/4") lines and perform important safety functions, and, therefore, automatic isolation valves should not be used.

The staff concluded that a departure from the explicit requirements of GDC 55, such as that represented by the CRD hydraulic design, is justified without isolation valves. This assessment was based on the fact that the CRD withdraw lines penetrating the containment and routed to the HCUs are small in diameter (3/4") and are conservatively designed and of high quality. Even when the staff postulated a break in one of these lines during reactor operation (including scram), they found that:

- The restricted flow area of the CRD limits the reactor coolant leakage to a very small value (within the capabilities of the reactor coolant makeup capabilities).
- The reactor can be shut down and cooled down in an orderly manner.

The similarity between the CRD line discussion and instrument lines provides a precedence for acceptability of the instrument lines without credit for EFCVs. Both cases represent small lines which connect to the RPV pressure boundary and contain restricted flow areas. Based on the low consequence of instrument line breaks, a case can be made that EFCVs are not required. Testing of the EFCVs is not justifiable if the valves themselves are not required.

3. Consequence Evaluations

3.1. Radiological Consequences

Radiological consequences from RCPB instrument line breaks have been evaluated at most plants to show compliance with Regulatory Guide 1.11 and are documented in some UFSARs. A typical GE radiological evaluation of the instrument line break with and without a ¼" orifice installed has been conducted (Attachment B) using a GE methodology which has been accepted by the NRC in GE FSAR submittals. No credit is taken in these evaluations for the operation of the Standby Gas Treatment System (SGTS).

The results of the evaluation (Attachment B) indicate that even without a ¼" orifice installed, the resulting thyroid dose at the site boundary is 16 Rem, which is about 5% of the 10CFR100 limit and may be considered insignificant. Similarly, whole body exposures are shown to be about 1% of the 10CFR100 limits without credit for an orifice. With an orifice, the doses are reduced by a factor of more than 5, which provides added conservatism.

The radiological consequence of EFCVs failing to function upon demand is sufficiently low to be considered insignificant. Specific analyses are needed to confirm this conclusion at each plant, but similar results would be expected. Because of the insignificant consequence with EFCV failure, it can be concluded that the EFCVs are not needed to assure a containment isolation function.

3.2. Operational Consequences

The operational impact of an EFCV which is connected to the RPV pressure boundary failing to close is based on the environmental effects of a steam release in the vicinity of the instrument racks. With the exception of potential jet impingement impacts, the environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the average release of about 6,000 lb/hr of steam into the reactor building (assuming a ¼" restriction in the line). A release of this magnitude, with or without an orifice, is within the pressure control capacity of reactor building ventilation systems (typically greater than 100,000 cfm). Due to its large volume, the bulk reactor building temperature would not be expected to be significantly affected except in the vicinity of the break.

Without successful functioning of the EFCV, the steam release to the reactor building of this magnitude or greater may exceed the capability of the SGTS to remove moisture and control humidity to the charcoal beds. However, a detailed evaluation would be necessary to confirm SGTS performance. No credit for SGTS operation is taken in the offsite dose calculations (Appendix B) and the results are acceptable. Therefore, the intent to maintain the integrity and functional performance of secondary containment following instrument line breaks is met independent of SGTS operability.

Separation of equipment in the reactor building minimizes the operational impact of an instrument line break on other equipment due to jet impingement. Nevertheless, the presence of an unisolated steam leak into the reactor building would require a reactor shutdown and depressurization to allow access to manually isolate the line.

4. Risk Evaluation

4.1. Failure Modes

The basis for the evaluation of public risk relies on an estimate of the failure of the EFCV to close when required, the instrument line break frequency, and the offsite consequence of the event. It can be speculated that most EFCVs fail to close due to sticking rather than some other mechanical problem. Sticking is assumed to be a time-dependent phenomenon which controls the EFCV unavailability as a function of the surveillance testing interval.

4.2. Failure Rates

The reliability of EFCVs was evaluated based on testing experience provided by different BWR utilities in response to the BWR Owners' Group survey (Attachment A). The data selected from the survey for evaluation, as well as the result of the evaluation, are presented in Table 4-1. The composite data show a very high reliability for the EFCVs with a total of 11 failures in over 10,000 valve years of operation.

The values shown in Table 4-1 for the "Upper Limit Failure Rate" were calculated using the following equation:

$$\lambda_U = \frac{1}{2T}\chi^2_{\alpha:2r+2} \qquad (1)$$

where:

λ_U = the upper limit failure rate per hour
T = the operating time in hours
r = the number of failures
$\chi^2_{\alpha:2r+2}$ = the value taken from the chi-square distribution tables which corresponds to 2r+2 degrees of freedom and $\alpha = 0.05$ ($1-\alpha = 0.95$ is the specified confidence level).

The last row of Table 4-1 shows the composite failure rate of EFCVs based on the data from the plants listed in the table. The best estimate and the confidence limit were calculated using the sum of the operating times and failures shown in the table. Table 4-2 shows failure rates calculated for each EFCV manufacturer, based on the data presented in Table 4-1. The results of the failure rate analysis presented in Tables 4-1 and 4-2 show relatively consistent values if calculated for each plant or for each valve manufacturer.

Based on this data, the composite failure rate value of 1.67E-7/hr between testing associated with a 95% confidence level for EFCVs is considered a best estimate of the reliability of EFCVs based on the current testing experience.

4.3. Instrument Line Break

The instrument line break frequency was calculated based on the WASH-1400 small pipe break failure rate of 6.1E-12 per hour per foot of line, and a conservatively assumed average 100 feet of line from the EFCV to the instrument. It has been assumed that this failure rate applies equally to all small pipe sizes (½" to 2", per WASH-1400). It is also assumed that this value is independent of whether the line is pressurized or not. Therefore, for a single instrument line the resulting frequency is 5.34E-6 breaks per year (6.1E-12 * 8760 hrs/year * 100 ft. = 5.34E-6).

4.4. Release Frequency

The risk impact on the public health and safety from EFCVs can be evaluated as the product of a release frequency (due to a break in an instrument line concurrent with an EFCV failure to close) and the consequence of the release (Section 3.1). The release frequency can be calculated based on the instrument line break frequency (Section 4.3) and EFCV failure to close probability.

Table 4-3 shows the release frequency from a single instrument line, assuming different test intervals for the EFCV. The release frequency was calculated using the following equations:

$$RF = I * \overline{A} \qquad (2)$$

and

$$\overline{A} = \lambda \frac{\theta}{2} \qquad (3)$$

where:

RF = release frequency per year
I = instrument line break frequency per year (Section 4.3)
$\overline{A}$ = EFCV unavailability (failure to close probability)
λ = EFCV failure rate per hour (Section 4.2)
θ = EFCV surveillance test interval in hours

Based on the release frequency shown in Table 4-3 for one instrument line, and assuming 86 instrument lines with testable EFCVs in a plant, the release frequency from any broken instrument line is:

- For 18-month surveillance test interval 86 * 5.86E-9 = 5.04E-7 events/year

- For 10-year surveillance test interval 86 * 3.91E-8 = 3.36E-6 events/year

These release frequencies are sufficiently low that it can be concluded that a change in surveillance test frequency has minimal impact on the valve reliability.

Table 4-1 EFCV Failure Rates

Plant [Note 1]	Make of EFCV	Operating Time [years] [Note 2]	Operating Time [hours] [Note 2]	Number of Failures	Best Estimate Failure Rate [/h] [Note 3]	Upper Limit Failure Rate [/h]	Notes
Browns Ferry	Marotta	100.5	8.80E+05	3	3.41E-06	8.81E-06	4, 6
Brunswick	Valcor	267	2.34E+06	0	0	1.28E-06	5
Clinton	Dragon	220	1.93E+06	0	0	1.55E-06	
DAEC	Marotta	1974	1.73E+07	0	0	1.73E-07	
Dresden	Chemquip	922	8.07E+06	0	0	3.71E-07	
Fermi 2	Dragon	930	8.15E+06	0	0	3.68E-07	
Fitzpatrick	Marotta	2019	1.77E+07	0	0	1.69E-07	
Monticello	Chemquip	2314	2.03E+07	1	4.93E-08	2.34E-07	
Oyster Creek	Chemquip	465	4.07E+06	0	0	7.36E-07	
Susquehanna	Marotta and Valcor	144	1.26E+06	4	3.17E-06	7.26E-06	6, 7
VY	Chemquip	1725	1.51E+07	1	6.62E-08	3.14E-07	
WNP2	Dragon	1344	1.18E+07	2	1.69E-07	5.34E-07	
Composite		**12424.5**	**1.09E+08**	**11**	**1.01E-07**	**1.67E-07**	

Notes to Table 4-1:

1. LaSalle data was not included in the table due to inconclusive reported data.
2. Determined by multiplying the number of tested EFCVs with the time period during which the number of occurring failures was reported.
3. These failure rates are obtained by dividing the number of failures by the operating time.
4. Utility reports 67 valves tested during last outage, of which 3 failed the test. The EFCV operating time for 3 failures is: 67valves * 1.5years = 100.5years
5. Utility reports 89 valves per unit tested at an interval of 18 months (1.5 years). There were no failures during the last test of each unit. Therefore, the EFCV operating time for 0 failures is 2units * 89valves * 1.5years = 267years.
6. Conservative results. The valves reported "failed" passed a bench test after being removed.
7. Unit 2 reports 96 EFCVs tested per unit every outage, presently 18 month cycle and 4 failures during the last outage of Unit 2. Therefore, the EFCV operating time is: 96 valves*1.5 years = 144 years Only Unit 2 data shown.

Table 4-2 EFCV Failure Rates by Manufacturer

Make of EFCV	Plant*	Operating Time [years]	Operating Time [hours]	Number of Failures	Best Estimate Failure Rate [/h]	Upper Limit Failure Rate [/h]
Chemquip	Monticello, VY, Oyster Creek, Dresden	5426	4.75E+07	2	4.21E-08	1.33E-07
Dragon	Clinton, Fermi 2, WNP2	2494	2.18E+07	2	9.2E-08	2.89E-07
Marotta	Browns Ferry, DAEC, Fitzpatrick, Susquehanna*	4093.5	3.59E+07	3	8.37E-08	2.16E-07
Valcor	Brunswick, Susquehanna*	67.5	5.91E+05	0	0	5.07E-06

** Susquehanna has a combination of valves; No breakdown of numbers of failures by valve type was available.*

Table 4-3 Release Frequency from a Single Instrument line
(based on 5.34E-6 instrument line break frequency, and 1.67E-7 EFCV failure rate)

EFCV Test Interval [years]	EFCV Test Interval [hours] θ	EFCV Unavailability $\overline{A}$	Release Frequency [/year] RF
1.5	13140	1.10E-03	5.86E-09
2	17520	1.46E-03	7.81E-09
6	52560	4.39E-03	2.34E-08
10	87600	7.31E-03	3.91E-08

5. Discussion

Risk of EFCV failure

Failure of an EFCV to close will not involve a significant increase in the probability or consequences of an accident previously evaluated. If an EFCV fails to close, a low leak rate will exist due to the ¼" orifice, effective valve restriction, or instrument tubing size. FSAR analyses for a ruptured instrument line have shown offsite doses well below 10CFR100 limits without EFCVs.

The effect of extending the EFCV testing intervals is a corresponding increase in the potential frequency for a release. However, even if the test interval were increased to once in 10 years, the release frequency from an individual line remains very low at about 4E-8/year. Therefore, considering the low consequence of release, the extension of the surveillance interval does not affect the risk to the public associated with a failure an instrument line and the failure of an EFCV to perform its intended function.

The risk to the public can be shown by combining the release frequencies in Section 4 with a consequence of release from Section 3.1. The corresponding public risk with the current testing basis can be shown to be ~3E-5 mRem/yr. (Whole Body) [5.04E-7 events/yr x .05 Rem/event = 2.5E-5 mRem/yr]. With an extended testing interval, this value changes to ~2E-4 mRem/yr. (Whole Body). This is five to six orders of magnitude below 10CFR20.105 annual exposure limits to the general public of 500 mRem/yr (Whole Body).

Clearly, the risk to public health and safety (based on plant experience) is extremely low and not impacted by the testing interval. This further justifies that the EFCVs do not meet the criteria for being included in the Technical Specifications.

As discussed in Section 3.2, instrument lines in Mark I and Mark II plants are located outside of the Primary Containment in the Reactor Building, in an area served by the Standby Gas Treatment System following an accident. This provides additional mitigation of any postulated offsite release from a broken instrument line.

Personnel Hazard Reduction

Changing the testing interval, as with any reduction in required maintenance, inherently reduces the risk of industrial accidents including inadvertent exposure to radioactive liquid and occupational exposure. Furthermore, there is a consequential reduction in the amount of liquid radwaste that requires processing. Both these issues provide favorable benefits from an increased testing interval.

Licensing Basis

Section 2 provided justification that Technical Specifications form the sole licensing basis for testing of RCPB EFCVs. The review also demonstrated that a basis for establishing an LCO in Technical Specifications for EFCVs could not be justified. The basis for the surveillance interval is that testing during outages shows reliable performance. The evaluation in Section 4 shows that even with a 10-year testing interval, performance would not be significantly degraded.

No General Design Criteria, Regulatory Guide or ASME code requires testing of EFCVs unless they are considered to provide a containment isolation function. The results of Section 3.1 justify that the EFCVs are not needed to provide a containment isolation function.

Furthermore, the RCPB instrument lines, in which the EFCVs are installed, are similar to the Control Rod Drive System withdraw lines, in that they are normally pressurized to reactor operating pressure, are highly restricted and are unisolable from the reactor. The CRD insert/withdraw lines have been accepted by the NRC without isolation provisions.

It can be concluded that LCOs for the EFCVs are not required to be included in the Technical Specifications and that EFCV testing requirements can be deleted from plant Technical Specifications.

Reliable Design Requirements

The design requirements contained in Regulatory Guide 1.11 provide a highly reliable design with insignificant consequences in the event of an instrument line break. These design requirements are not changed by an alternate testing interval.

Performance Based Testing

Although EFCVs are not required to be tested to meet the requirements of 10CFR50, Appendix J, the approach specified in 10CFR50, Appendix J, Option B, can provide an approach familiar to the NRC for establishing an acceptable interval between EFCV tests.

10CFR50, Appendix J, Option B allows a performance-based approach for determining the leakage rate surveillance testing frequencies for Type A, Type B, and Type C containment penetrations. Extensions of Type B and Type C test intervals are allowed based upon completion of two consecutive periodic "as-found" tests where the results of each test are within a licensee's allowable administrative limits. An "as-found" test is one performed prior to any periodic maintenance. Acceptable performance history of each component is used as the basis for extending the surveillance test interval. If performance experience satisfies the criteria, test intervals may be increased up to a maximum of 120 months (10 years).

The Nuclear Energy Institute (NEI) prepared NEI 94-01 (Reference 9) to provide implementation guidelines for meeting 10CFR50, Appendix J, Option B. NEI 94-01, which is endorsed by the NRC in Regulatory Guide 1.163 (Reference 10), states:

> *"Additional considerations used to determine appropriate frequencies may include service life, environment, past performance, design, and safety impact."*

At most BWRs, historical EFCV performance justifies an extension of the testing interval based on performance based criteria, and the extension of the surveillance interval does not change the design, function, or operation of the EFCVs. When requesting test interval extension, EFCVs with similar operating and environmental conditions should be combined into groups.

6. Conclusions

Failure of RCPB EFCVs does not have adverse consequences in terms of risk to the public. EFCVs are not necessary to provide a containment isolation function. Consequently, the only licensing bases for testing requirements are contained in the Technical Specifications. The information in this report provides justification for relocating the RCPB EFCV testing requirements from Plant Technical Specifications to the Technical Requirements Manual.

Elimination of testing is not recommended. Without periodic testing, a stuck EFCV would not be detected. Automatic closure of EFCVs improves access to instrument lines in the event of a line break and improves maintainability and reduces contamination of the reactor building. Since EFCVs provide a simple method of avoiding these adverse operational considerations associated with recovery following instrument line breaks, the BWROG believes they should be retained in the plant design, and tested periodically in accordance with administrative controls. EFCV testing does not need to be retained in Plant Technical Specifications.

Reliability and consequence analyses provide a performance basis for extending the test interval up to 10 years without significantly affecting offsite risk. It is recommended that the testing interval be extended based on performance-based criteria included in 10CFR50, Appendix J Option B. An extended testing interval reduces plant costs and occupational exposures without significantly affecting offsite risk. When requesting test interval extension, EFCVs with similar operating and environmental conditions should be combined into groups. A test schedule based on actual valve performance should be established and controlled in plant administrative procedures. A staggered test schedule is suggested, in which a portion of the valves are tested each outage. Each EFCV should be tested at least once every ten years.

7. References

1. *"Standard Technical Specifications, General Electric Plants"*, NUREG 0123, Rev 3, December 1980.

2. *"Standard Technical Specifications, General Electric Plants (BWR/4)"* , NUREG 1433, April 1995.

3. "*Technical Specifications*", 10CFR50.36, July 29, 1996.

4. *"Standard Technical Specifications, General Electric Plants (BWR/6)"*, NUREG 1434, April 1995.

5. *"Code of Federal Regulations, Part 10CFR50, Appendix A"*.

6. *"Instrument Lines Penetrating Primary Reactor Containment"*, Regulatory Guide 1.11 (Safety Guide 11), 3/10/71.

7. *"Code of Federal Regulations, Part 10CFR50 Appendix J"*.

8. "*Inservice Testing of Valves in Light-Water Reactor Power Plants*", ASME OM-10, Subsection ISTC, 1995.

9. *"Industry Guidelines for implementing performance-based option of 10CFR50, Appendix J"*, NEI 94-01, July 26, 1995.

10. "*Performance Based Containment Leak-Test Program*", Regulatory Guide 1.163, September 1995.

Attachment A

Participating Utilities

Plant	*Utility*	*BWR type*	*Containment type*
Nine Mile Point 1	Niagara Mohawk Power Corporation	2	MK I
Oyster Creek	GPU Nuclear	2	MK I
Dresden	Commonwealth Edison	3	MK I
Millstone	Northeast Utilities	3	MK I
Monticello	Northern States Power	3	MK I
Quad Cities	Commonwealth Edison	3	MK I
Browns Ferry	Tennessee Valley Authority	4	MK I
Brunswick	Carolina Power & Light	4	MK I
DAEC	Alliant Utilities	4	MK I
Fermi 2	Detroit Edison	4	MK I
Fitzpatrick	New York Power Authority	4	MK I
Hope Creek	Public Service Electric & Gas	4	MK I
Limerick	PECO Energy	4	MK II
Peach Bottom	PECO Energy	4	MK I
Pilgrim	Boston Edison Company	4	MK I
Susquehanna	Pennsylvania Power & Light	4	MK II
Vermont Yankee	Yankee Atomic	4	MK I
LaSalle	Commonwealth Edison	5	MK II
Nine Mile Point 2	Niagara Mohawk Power Corporation	5	MK II
WNP2	Washington Public Power Supply System	5	MK II
Clinton	Illinois Power	6	MK III

Design Information

Plant - units	Total No. of EFCVs (all units)	Valve years	Pressure rating (psig)	Make and model	Typical Installation	Nominal line size	Internal orifice size	Minimum flow diameter	Flow controlled by body contour?	Operating pressure (psig)	Operating temp deg F
Browns Ferry 1,2,3	201	7638	Normal 1300 Design 1326 Proof 2300 Burst 4000	Marotta FVL 16D Part No. 280619	1" dia. 13/16" long EFCV inside 2 body halves connected by coupling nut	1"	Estimate 0.5mm	0.2-0.7gpm @ 1000 psig	No	1000	90
Brunswick 1,2	89	1980		Valcor	3/4" inst lines - ¼" orifice in DW	3/4"		0.25"		1030	560
Clinton	22	220	Low Press 9 - 30 High Press 100	Dragon Part No. 14455	In inst lines		No	3/8"			
DAEC	94	1974	1375	Marotta "Flo-Fuse", solenoid reset with Position Ind	In inst lines connecting to Rx coolant. 1/4" orifice in DW	1" inlet, 3/8" tube outlet	~¼"	0.25"	No	1050	575 max, 60 operating
Dresden 2,3	150	922	3000	Chemquip 50FM-11W1SS-B 50FM-920704	Instrument lines outside of containment	½"	Unknown	Unknown	Yes	1000	~100-110
Fermi 2	93	930	Design 1250	Dragon Model 10870 - ball check with PI	1" inlet 5/8" outlet	1"	0.25"	0.25"	No	1050	575
Fitzpatrick	93 Note 1	2019	Oper 150-1250 Proof 3250 Burst 10000	Marotta 1" flo-fuse, FVL16G and FVL16GB	In inst line outside containment. 1/4" orifice inside containment	1" inlet, 3/8" tube outlet		Ball seat orifice approx 5/32-1/4"	No	1040 oper, 150-1250 des	80 oper, 32-595 design
LaSalle 1,2	233		5400	Dragon Part No. 11935-1,3,5 and A11935-3,5			Reset orifice-.025/.014-.016/.0225/		Poppet config, port size, spring tension control closure	1650, 1250, 45	575, 575, 340

							.022-.028		time & flow		
Monticello	89	2314	Design 3000 Oper 1250	Chemquip 50FM-8346	1" pipe	1"	No	1/4"	Yes	1010	545
Oyster Creek	59	465	Chemquip	Chemquip							
Susquehanna 1,2	192 tested, 38 not tested	5568 for tested EFCVs	Marotta design 1510 and normal 1200. Valcor design 1500 and normal 1200	Marotta FVL16FD, Valcor V520-50-3	From Rx boundary 1" line thru a 3/8 orifice in DW to EFCV. Downstream tubing is 3/8"	1" inlet, 3/8" tube outlet	No		3/8" sharp edge orifice	1100	540
VY	82 instld, 69 testable, 8 permanently isolated, 5 used for SRV tailpipe ind and head seal leak det.	1725	3000	Chemquip 50FM-9344-1, 50FM-9345-2, 50FM-9346-3	Penetration, isol valve, EFCV, inst rack. 1/4" orifice in DW.		Yes, but internal diameter not specified	1/4" orifice installed in each line	Yes. Diameter not specified.	1015	50-100
WNP2	96	1344	1500	Dragon 12583			Valve body orifice 3/8"		No	1000	540

Note 1) In 1987, 12 valves were retired in place.

Surveillance Test Requirements

Plant	*Objective of test*	*Test method*	*Acceptance criteria*	*When test? Is it on critical path?*	*Exposure*	*Manhours*	*Test interval*
Browns Ferry	Ensure operability	Flow reduction Audible sound	Flow reduction Audible sound	22 preoutage-simplicity & no LCO 45 during startup at end of outage. Performed during In-Service leakage test, which is CP. If problems encountered it goes onto CP.	<1REM	340	Each cycle
Brunswick	Ensure operability	Closure test Vacuum pump Open test Water flow	Closure test ≤2"Hg decrease/minute Open test-evidence of backfill water flow after closure	Refueling outage. Not CP, since tested with test rigs during system window availability, not at beginning or end of outage	Normal Rx bldg dose rates during outage	135	18 months cycle
Clinton		Flow rate	<3 scfm air <1 gpm water	During outage, not CP. Performed during divisional outage	<400 mR	350	18 months
DAEC	Ensure operability	Open vent vlv	Substantial decrease in flow within 10 seconds	During refuel (or at pwr) when press >400 psi. Usually at beginning of outage or during S/U. Not currently CP.	Very little	180	Each cycle
Dresden	Ensure Operability	Flow rate	> 0.2gpm & < 2gpm @ 1000 psig	During hydro prior to startup, is critical path with the hydro.	~255 mR	314	18 month cycle
Fitzpatrick	Ensure operability	Flow rate	<3 gpm water	During startup from refuel (vssl hydro test). Near CP. Takes 22 hrs during 26 hr hydro window.	140 mR	200 + 100 for preparations	24 months cycle
LaSalle	Ensure operability Verify CR ind.	Flow rate	Demonstrate valve checks flow. Use mfrs specs (not provided)	Refuel outage. Low pressure checks not CP. High pressure checks (during hydro @ startup) are CP.			
Monticello	Satisfy Tech Spec	Flow rate	≤1.9 gpm and >0 gpm @ 900 psi	During hydro prior to S/U. Is CP with hydro.	360-720 mR	72	Each cycle
Susquehanna	Ensure Operability	Flow rate - open drain valve	Valve closes	As many as possible during S/D. Remainder during hydro @ S/U. Those during hydro are CP. Never during S/U.	3-5 mR (must be per valve)	192/unit = 384	Each cycle, presently 18 months, going to 24
VY	Measure checked flow	Verify checking action & flow rate	<1.5 gpm	During refuel, immediately following hydro (prior to S/U). Time limited by heatup to 212.	<50 mR	70-80	18 months cycle
WNP2		Flow at 85-90 psid	Flow reduction	Near end of outage. On CP because maintain pressure before or after hydro.	5 REM	360	24 months cycle

Testing Data

Plant	*EFCVs tested*	*Failures*	*Types of failures*	*Multiple failures?*	*Common cause?*	*Failure trend?*	*% requiring retest*	*Failure rate*
Browns Ferry	67 Note 1	None during most recent outage 21 @ U2 restart 5 @ U3 restart - Note 2	Testing following 6 and 10 yr. outages was done with improper procedures. Failures attributed to crud buildup and sticking, test methodology, and one broken spring - Note 3.	Yes. Attributed to long shutdown, improper test procedures, and lack of experience of test personnel.	Yes. Attributed to long shutdown, improper test procedures, and lack of experience of test personnel..	Decreasing	<1%	7% <3% in last 2 outages
Brunswick	178 Note 6	None	N/A	No	N/A	No trend	0%	
Clinton			Previous failures attributed to test methodology, not valve failures.					
DAEC	94 every cycle	None	N/A	No	N/A	No trend	Not recorded	N/A
Dresden	75 per unit Note 7	None	N/A	No	N/A	No trend	0%	0%
Fermi 2	744	None	N/A	No	N/A	No trend		
Fitzpatrick	1137	None					1.67%	0% using <3gpm criteria
LaSalle			Insufficient information	No data	Insufficient information			
Monticello	1602	1	Sticking - 1	No	N/A	No trend	0%	
Oyster Creek	59	0	N/A	No	N/A	No trend	0%	0
Susquehanna	192	None during last U1 and U2 tests	Previous failures attributed to improper test methods. Note 5			No trend	Very little	
VY	1380	1	Valve checked between 1.5 & 2.0 gpm. No cause found upon disassembly	No		No trend	0.03%	
WNP2	96	2	Failures attributed to configuration and test methodology	No		No trend	0%	<1%

Note 1) During last test

Note 2) 21 failures occurred at U2 restart in 1991, after >6 yr. outage. Improper test procedure utilized. Only 5 failures occurred at U3 restart in 1995 after >10 year outage. No failures occurred during most recent test of 67 valves. Current failure rate 0/cycle

Note 3) Some failures were probably good, but didn't "sound" good

Note 4) Until 1996, acceptance criteria values were erroneously based on equalizing leakage rather than checking flow. No valve is known to have ever exceeded plant guideline of <3gpm at normal reactor pressure

Note 5) Test by opening 3/8 inst drain valve, up to 200 ft from EFCV. EFCV design actuation pressure 3-10 psid Prior tests done at too low Rx pressure.

Note 6) Valves tested during most recent outage on Units 1 and 2.

Note 7) Valve test during the last two outages on Units 2 and 3, 53830.05 total operating hours (8.07E+06 total valve operating hours).

Maintenance Information

Plant	*Are EFCVs in a PM program?*	*Failed valves repaired or replaced?*	*What would reduce EFCV test burden?*
Browns Ferry	No	Replaced	Looked at obtaining different EFCV springs, so that vessel flood-up head would check the EFCVs in order to perform the tests in a less critical part of the outage. Not successful in this yet.
Brunswick	No	N/A	N/A
Dresden	No	Replaced, but sometimes repaired	A quick and simple test method that would not require any plant modifications that would be standard among all BWR's. Also, sample testing would reduce the burden during each outage.
Fitzpatrick	No - operating cycle surveillance sufficient	Upon failure, back-flush. If still fails, re-test at higher pressure (up to NOP), and back-flushed. If still fails, valve is replaced with pre-tested spare.	No indication that a change in valve design or maintenance program is needed. Recent changes to test method show favorable results. Fitzpatrick has concluded that actuation of the EFCV will contribute to mitigation of an instrument line break by restricting the flow area.
LaSalle		Valves are welded in place.	Yes, a change in valve design, test method, or maintenance program is needed to reduce the burden of EFCV testing.
Monticello	No	Replaced	No indication that a change in valve design, test method, or maintenance program is needed.
Susquehanna	No	Replaced	Yes, test method needs to be changed but will require modifications to the plant.
VY	No	Replaced	No indication that a change in valve design, test method, or maintenance program is needed.
WNP2	No	Valves are welded in place. Typically cut out, cleaned, and re-installed.	Presently reviewing test methodology and design basis.

Licensing

Plant	*Committed to GDC 55/56?*	*Tech Spec Rqmnt for testing?*	*Effective Inst Line break size*	*Other licensing commitments*	*Active or passive components, and why*
Browns Ferry	No, except for new mods	Verify EFCV operability once per cycle	3/8 - 3/4 inch	None known	Active - they have to check (move)
Clinton	Yes	Verify valve actuation at specified flow/DP. Verify close indication	3/4 inch	RG 1.11 is basis for licensing commitment on EFCVs	Active - required to actuate
DAEC	Yes	Test each inst line EFCV each cycle	1/4 inch orifice	Only in-service testing	Active - automatic operation is required for isolation to occur at ≤ 10 psid
Dresden	No, Dresden is a pre-GDC plant. However, the UFSAR states that the plant conforms to the intent of the GDC.	Each reactor instrumentation line excess flow check valve which fulfills a primary containment isolation function shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.	1" per UFSAR Analysis	ASME Section XI, IWV-1986	Active – Required to close to limit flow.
Fitzpatrick	Yes	Test for proper operation each cycle	1/4 inch	No	Active - ASME/ANSI Oma-1988 Part 10 § 1.3 describes active valves as those which must change obturator position
Monticello	No	Test for proper operation each cycle		Section XI	Passive
Susquehanna	Yes	Verify that valve checks flow	3/8 inch	No	Active - they have to check flow
VY	No	Verify EFCV operability once per cycle	3/8 inch	See attachment 8 to pre-license	Active - valve must open/close
WNP2	Yes, but no credit for EFCVs taken in inst line break analysis	Actuates to isolation position on 85-90 psig	3/8 inch	No	Passive - because EFCV is assumed to break off in inst line failure analysis

Inservice Testing (IST)

Plant	*How does Appendix J address EFCVs*	*IST requirements*
Browns Ferry	Excluded	Surv testing performed each cycle
Brunswick	Tested in ILRT	Verify operability per Tech Spec testing every 18 months
DAEC		Quarterly surveillance is called for. A relief request based on the risk of causing a plant transient and inoperability of safety-related instruments allows once per cycle check.
Dresden	Instrument lines are exempt from Appendix J, Type C testing provided they are not isolated from containment during the performance of a Type A, Integrated Primary Containment Leak Rate Test. (UFSAR 6.2.4.2.1)	Category A testing of containment isolation valves in accordance with the Appendix J Program. The excess flow check valves are exempt from Appendix J testing. Category C check valve closure test quarterly. Dresden has a relief request to allow closure testing to be performed during reactor refueling in accordance with Tech Specs.
Fitzpatrick	Section XI of the ASME code, 1989 edition, specified that the rules for IST of valves are stated in ASME/ANSI O&M stds Part 10. EFCVs at JAF are identified as class 1, category A/C, self-actuating ball check type	Exercise test to closed position every 3 months AND leak test for other than containment isolation valves every 2 years. JAF has opted for refuel outage justification relief, and tests as follows: exercise test to closed position every refueling outage
Monticello	Part of Section XI	<1.9 gpm and >0 gpm @ 900 psi
Susquehanna	Tested in ILRT	None
VY	EFCV treated as extension of piping, visual at pressure for external leakage	
WNP2	Excluded	Exercise every 2 years - valves are passive and do not require quarterly testing

Attachment B

Instrument Line Break Radiological Analysis

Analysis

This analysis concerns the Instrument Line Break Accident (ILBA). The calculation will be done using the standard GE ILBA based upon Reference B-1.

GE Standard Analyses

The GE standard analysis is described in Reference B-1, Chapter 8, which assumes a break in an unisolable small line (typically an instrument line) which is choked by a one-quarter inch orifice. The break continues unabated for ten minutes, at which time temperature sensors in the reactor building (or whatever area outside containment) cause the operators to respond by scramming and shutting down the reactor. The reactor follows a standard cooldown of 100°F/hr for a period of 5.5 hours, at which time the accident is terminated.

The flow rate and depressurization curves for the restricted case are given in Reference B-1, Figures 8-1 and 8-2, respectively. The total integrated release for the restricted case is approximately 33,000 lbs. For the unrestricted case, a total of 100,000 lbs of steam was assumed to be released to the reactor building.

The release of fission products is restricted to iodines, which are radiologically the most significant isotopes for site dose considerations. The reactor water inventory is assumed to be a Technical Specification limits of 0.2 μCi/gm Dose Equivalent I-131. Since the exact determination of separate isotopic species is dependent upon the dose data set used, the dose conversion factors consistent with TID-14844, Reference B-3, are used in this analysis and are given in Table B-1 below.

To account for iodine spiking source terms, the following algorithm is used:

- At accident initiation, 15% of the gap inventory of iodines is released to the vessel water and is assumed homogeneously mixed in the water. The 95% gap inventory is given in Ref. B-1 and is listed in Table B-1.

- As the reactor depressurizes, the remaining 85% is released proportional to the depressurization so that all the gap inventory is released to the vessel by the termination of the accident.

As the vessel water is released to the reactor building, it is assumed that the iodine in the water is released to the air space directly proportional to the flash fraction of the released water. This is a significant conservatism in the calculation of the released iodine.

The fission products released to the air space are assumed released to the environment at an assumed rate of 4800% per day as a cold ground level release. Operation of the Standby Gas Treatment System is not assumed. Specific computational detail is given in Table B-2.

Results

The results of an evaluation of postulated unrestricted and restricted instrument line breaks at Dresden are given in Figures B-1 and B-2. The results show the thyroid and whole body dose as a function of time for the site boundary (exclusion area boundary) and for the low population zone boundary (LPZ), and are summarized below.

Dose	Unrestricted Result (Rem)	Restricted Result (Rem)	Unrestricted fraction of 10CFR100 Limit (%)
Thyroid			
LPZ	.5	.09	.2
Site Boundary	16	.9	5.3
Whole Body			
LPZ	.003	.0006	.06
Site Boundary	.05	.008	1.0

Table B-1 Isotopic Input Data

Isotope	λ(/sec)	DCF (REM/Ci)	Conc (μCi/gm)	Gap Inven (Ci/Bundle)
I-131	9.9771E-7	1.48E+6	0.047	2.1
I-132	8.4263E-5	5.35E+3	0.415	3.2
I-133	9.2568E-6	4.00E+5	0.326	5.0
I-134	2.1963E-4	2.50E+4	1.207	5.4
I-135	2.9239E-5	1.24E+5	0.755	4.8

Table B-2 Other Input Data

Item	Value	Basis
Meteorology site boundary LPZ	 2.6E-4 sec/m3 1.1E-5	Ref. B-2, UFSAR
Number Bundles	724	Eng. Data Base
Mass RPV Water	590,000 lb.	Ref. B-2, UFSAR

Figure B-1
Thyroid Dose

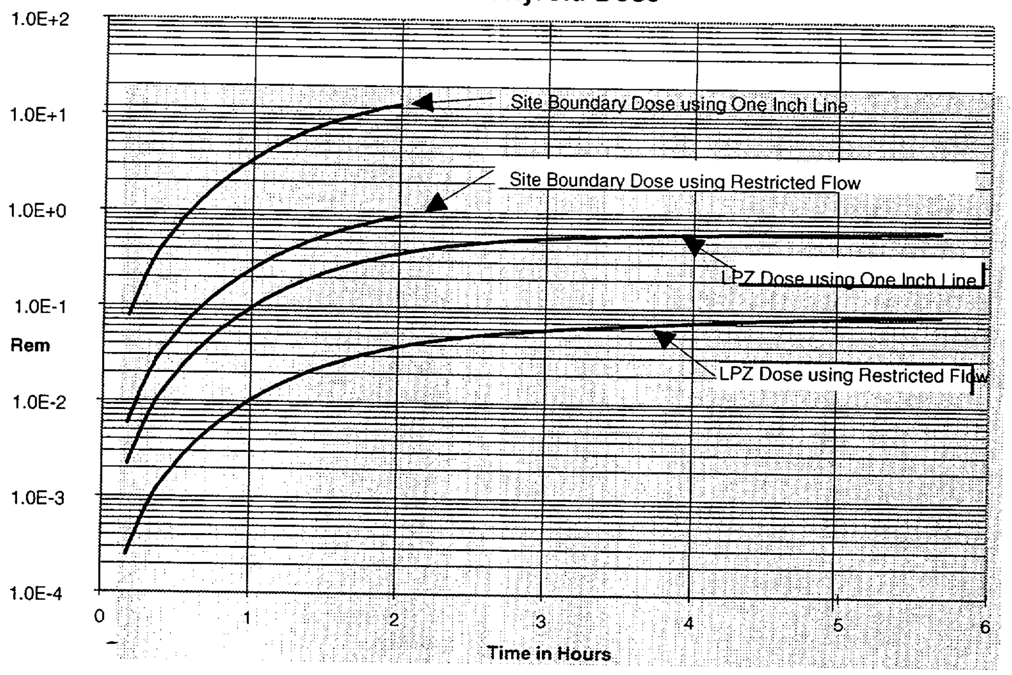
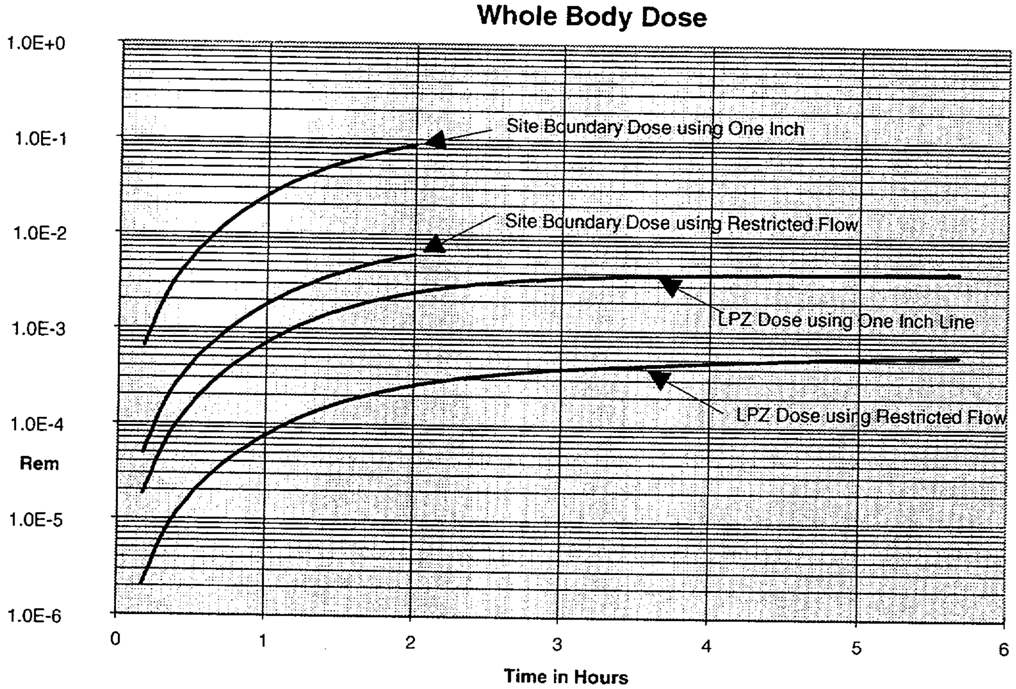


Figure B-2
Whole Body Dose



References

B-1. Careway, HA, Nguyen, VD., Stancavage, PP, "Radiological Accident Evaluation - the CONAC Code", General Electric Report NEDO-21143-1, December 1981.

B-2. Dresden UFSAR, Subsection 15.6.2.

B-3. DiNunno J.J., et al, "Calculation of Distance Factors for Power and Test Reactor Sites", Technical Information Document 14844, NTIS, March 23, 1962.

BWR OWNERS' GROUP

W. Glenn Warren, Chairman
Tel: (205) 992-5940
Fax: (205) 992-0391
wgwarren@southernco.com

c/o Southern Nuclear • 40 Inverness Center Parkway • PO Box 1295 • Birmingham, AL 35242

BWROG-00001
January 6, 2000

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: 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.

Note: This paper is written in response to NRC request for BWR Owners' Group generic response to subject Request for Additional Information

Dear Sir(s):

In 1998 the Boiling Water Reactor Owners Group prepared a Topical Report justifying the relaxation of the Surveillance intervals for testing Excess Flow Check Valves. Under Attachment 1, this report was submitted to the NRC with Duane Arnold Energy Center as the lead plant. During the NRC review of this lead plant submittal the staff requested additional information (Attachments 2 and 4). IES Utilities provided plant specific responses to these questions under Attachments 3 and 5. Generic Traveler 334 was submitted to the NRC to allow applicable BWR's to adopt conforming changes to their Technical Specifications. To simplify future plant submittals, the NRC staff requested that the BWROG provide generic responses to the questions posed to the lead plant. Attachment 1 to this letter provides these responses. Upon NRC approval of the topical report it will be reissued including the NRC Safety Evaluation Report along with this letter allowing direct referencing in future plant submittals.

Very truly yours,

W.G. Warren, Chairman
BWR Owners' Group

cc: JM Kenny, BWROG Vice Chairman
BWROG Participating Primary Representatives
TG Hurst, GE
SA Bump, GE

Attachments: (1) BWROG-00001, Letter from W. G. Warren, Chairman (BWROG) to NRC dated January 6, 2000, "Generic Response to Request for Additional Information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements."

(2) NG-99-0308, Letter from J. Franz (IES Utilities) to NRC, dated April 12, 1999, "Technical Specification Change Request (TSCR-010): Relaxation of Excess Flow Check Valve Surveillance Testing."

(3) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 27, 1999, "Request For Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)."

(4) NG-99-1358, Letter from K. Peveler (IES Utilities) to NRC, dated October 5, 1999, "DAEC Response to Request For Additional Information on Technical Specification Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements"

(5) Letter from B. Mozafari (NRC) to E. Protsch (IES Utilities), dated September 30, 1999, "Request for Additional Information on Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements at Duane Arnold Energy Center, (TAC No. MA05421)"

(6) NG-99-1383, Letter from Ken Peveler (IES Utilities) to NRC, dated October 8, 1999 "DAEC Response to Request for Additional Information On Technical Specification Change Request (TSCR) Regarding Excess Flow Change Request (TSCR) Regarding Excess Flow Check Valve Surveillance Requirements.

BWROG Generic Response to Request for Additional Information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements

NRC Lead Plant Question 1:

You have proposed a 10-year test interval for Excess Flow Check Valves (EFCVs), and have primarily referred to Option B of Appendix J to 10 CFR Part 50, as the model for doing this. However, you have neglected to address the fact that the NRC staff, through Regulatory Guide (RG) 1.163, limits containment isolation valve testing intervals to a maximum of 5 years. By licensees' requests, the RG has been incorporated by reference into the Technical Specifications (TS) of every plant that is using Option B of Appendix J. Thus, the 5-year interval is a requirement for every plant using Option B.

Insofar as your justification for a 10-year interval is, for the most part, that it is like Option B of Appendix J, provide additional justification for your proposed interval that is longer than the 5-year interval used for Option B of Appendix J.

BWROG Response to Question 1:

A cyclic nominal interval for testing a representative sample is proposed. The valves in question are of similar design, similar application, and similar service environment. Performance of the representative sample provides a strong indicator of the performance of the total population. The 10-year nominal interval solely limits the time between tests for any specific valve and provides additional assurance that all valves remain capable of performing their intended function.

The failure rate data listed in Table 4-1 of the subject report is considered the primary basis for the performance-based interval. In addition, the consequences of a failure to isolate have been evaluated and found to be acceptable with respect to off-site doses. Each site adopting this change will need to confirm the applicability of this analysis.

RG 1.163 is essentially an NRC staff endorsement, with exceptions, of a Nuclear Energy Institute (NEI) document, 94-01, concerning the performance-based option of 10 CFR Part 50, Appendix J. Per RG 1.163, "Because of uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical Type C component performance data, and because of the indeterminate time period of three refueling cycles and insufficient precision of programmatic controls described in Section 11.3.2 [of NEI 94-01] to address these uncertainties, the guidance provided in section 11.3.2 for selecting extended test intervals greater than 60 months for Type C tested components is not presently endorsed by the NRC staff."

The data provided in the BWROG report shows that bases for limiting intervals to 60 months, as stated in RG 1.163, are not applicable to EFCVs. Specifically:

- Unquantified leakage rates for test failures are not applicable because the maximum leakage through an unisolated instrument line is quantified. The dose consequences of the failure to isolate are acceptable.
- Repetitive/common-mode failures are not applicable as evidenced by the low industry failure rate and more specifically by the BWROG report, Table 4-2, "EFCV Failure Rates by Manufacturer."
- Aging effects are not a concern. The industry data already provided does not indicate any increase in failure rate with time in service.
- Historical performance data associated with EFCVs has been provided and is considered adequate to justify the proposed interval.
- There is no indeterminate time period involved with this proposed change. Every cycle a representative sample will be tested.

Therefore, we believe RG 1.163 and the 60-month limit for test intervals are not applicable to EFCV test intervals. EFCVs are not typically subject to Type C leak rate testing.

NRC Lead Plant Question 2:

Under the Appendix J, Option B, program, if a component on an extended test interval fails a test, it must be returned to the nominal test interval until subsequent testing re-establishes its reliable performance. In other words, if it doesn't continue to perform well, it gets tested more often. Your proposal has no similar well-defined feedback mechanism for EFCVs. There is only the following:

> EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. (From the proposed DAEC Bases)

The risk-informed IST Regulatory Guide, RG 1.175, also specifies the need for a feedback mechanism.

Justify the absence from your proposal of an explicit, well-defined performance feedback mechanism that requires increased testing when valves fail their tests, or add such a mechanism to your proposal.

BWROG Response to Question 2:

Each licensee who adopts the reduced surveillance intervals recommended by the subject report should ensure an appropriate feedback mechanism to respond to failure trends is in place. Generic Traveller TSTF 334 includes this commitment.

NRC Lead Plant Question 3:

The proposed Duane Arnold TS says "a representative sample" of EFCVs will be tested every 2 years. The "representative sample" is not defined. Your proposed Bases, which, you are careful to point out, are not part of your proposed license amendment and are included for information only, say you will test 20% of the valves each refueling outage and thus test all of them in a 10-year period. In fact, the proposed TS would allow you to test less than 20% each time, and the concept of "representative" could change with time to exclude certain valves that were problems (e.g. repeat leakers, hard to access). The point is not that these things will actually happen, but that that proposed TS contain virtually no actual requirements.

Justify the absence of more specific requirements in the proposed TS, or add specific requirements to the proposed TS.

BWROG Response to Question 3:

The term "representative sample," with an accompanying explanation in the TS BASES, is identical to current usage in the Standard TS (STS), NUREG-1433, Revision 1. Specifically, NUREG 1433 uses the term "representative" in TS Surveillance Requirement (SR) 3.8.6.3, in reference to battery cell testing and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. Therefore, the application of a "representative sample" for the EFCV testing SR, with its accompanying definition in the BASES is consistent with the STS usage.

In addition, as required by the Technical Specification Task Force (TSTF) process for changing the STS, a generic traveler (TSTF-334), has been submitted to the NRC for review. One of the primary reviews conducted by the TSTF committee is conformance to the Writer's Guide for TS. There were no concerns raised over the content, format or proposed use of the BASES. This traveler was approved by the TSTF on May 6, 1999 and forwarded to the NRC for review on June 23, 1999.

The BASES are routinely used to capture commitments imposed by the Staff as terms or conditions for approval of specific TS changes in their Safety Evaluation Reports (SERs). As written, the Generic Traveler is consistent with how other, similar testing programs that utilize a sampling approach are constructed in the STS. Thus, additional requirements within the TS proper are not needed.

BWROG Response to Request for Additional Information on
Technical Specification Change Request Regarding Excess
Flow Check Valve Surveillance Requirements

NRC Lead Plant Question 4:

Explain the discrepancy between page 11, Section 4.2, top paragraph that states "...a total of nine failures over 10,000 valve years of operation" and Table 4.1 on page 14 that indicates 11 failures.

BWROG Response to Question 4:

References to nine failures will be corrected when the topical report is reissued.

NRC Lead Plant Question 5:

Refer to page 12, Section 4.3, top paragraph. The single instrumentation line break frequency of 5.34e-6/year assumed was based on WASH-1400 data. Explain why a more updated value was not used. Individual Plant Examination data indicate that such frequency could be higher.

EFCV unavailability used the lambda T over two formula. Provide the basis for assuming a constant failure rate for 10 years. Explain how the nature of "stickiness" might change over such a long period (10 years) with potentially new failure mechanisms becoming dominant.

Describe the impact/change on the release frequency estimate if

(1) a more updated instrumentation line break frequency and
(2) a constant failure rate is not assumed.

BWROG Response to Question 5:

The line break frequency calculated in the GE topical report for a single instrument line is based on a break failure rate of 6.1E-12 per hour per foot of line, and a conservatively assumed average pipe length of 100 feet (6.1E-12/hr-ft * 8760 hrs/yr * 100 ft = 5.34E-6 breaks/yr). The value of 6.1E-12 per hour per foot is from WASH-1400 and is applicable to small pipe. WASH-1400 "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," was published in 1974 and therefore had a limited amount of nuclear power plant operating experience from which to base its component failure rate data. In fact, much of its data was drawn from non-nuclear facilities. More recent pipe failure rate data is published in EPRI Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants", July 1992. This report compiles failure data from approximately 1000 years of nuclear plant operating experience.

The smallest pipe size considered in the EPRI report is 1/2 inch to 2 inch diameter pipe. Failure rate data for this class of piping will be considered representative of the subject instrument line piping. Also, failure rate data is calculated and reported on a "per section" basis rather than a "per foot" basis. (This unit of measure was chosen because the influence of welds, and their adjacent heat-affected zones on the failure rates, is by far greater than the influence of length.) A pipe section is defined to be a segment of piping between major discontinuities such as valves, pumps, reducers, tees, etc.

Table 4.4-2 of EPRI TR-100380 contains recommended pipe rupture failure rates based on reactor type (Westinghouse, Babcock & Wilcox, Combustion Engineering, and General Electric) and system. The rate for reactor coolant piping in General Electric BWRs is judged to be most representative of the subject instrumentation lines. The recommended average value representing all pipe sizes in this category is 6.7E-10 failures per section per hour. A multiplier of 1.2 (derived in Section 4.4.10.2) is applied to this value to obtain the failure rate for small pipe.

1.2 * 6.7E-10/hr-section = 8.04E-10 failures per hr per section

If a typical instrument line is assumed to contain five sections (ref. UFSAR Figure 3.2-2), its rupture failure rate is:

5 sections * 8.04E-10/hr-section * 8760 hrs/yr = 35.2E-06 failures per year

This value is 6.6 times greater than the value of 5.34E-06/yr calculated in the GE Topical Report using data from WASH-1400.

The GE Topical Report determines an upper limit EFCV failure rate based upon eleven observed failures in 1.09E+08 hours of service. It can be postulated that the failure rate for EFCVs is not constant over time, but may in fact increase over time due to age related factors.

If the number of observed failures is conservatively assumed to be five times that of the actual observed number, the resulting calculated upper limit EFCV failure rate would still be acceptably small.

The formula for upper limit failure rate used in the GE Topical Report is:

$$\lambda_U = \frac{1}{2T}\chi^2_{\alpha:2r+2}$$

Where:

T is the operating time in hours

r is the number of failures

$\chi^2_{\alpha:2r+2}$ is the value taken from chi-square distribution tables which corresponds to 2r+2 degrees of freedom and 0.95 confidence level (a = 1-0.95 = 0.05)

For eleven observed valve failures, degrees of freedom is 24. The value of χ^2 for 24 degrees of freedom and a 95% confidence level is 36.415. Therefore,

$$\lambda_U = \left[\frac{1}{2*1.09E+8}\right]*36.415 = 1.67E\text{-}07 \text{ failures per hour}$$

For fifty-five observed valve failures (five times normal), degrees of freedom is 112. Chi-squared values are not typically provided for degree of freedom values above thirty because for large values, the chi-squared distribution is close to that of the standard normal distribution. In this case, χ^2 is approximated by:

$$\chi^2 = \frac{1}{2}\left[x_\alpha + \sqrt{2n-1}\right]^2$$

Where: x_α is the α-point of the standard normal distribution

n is the degrees of freedom

(Ref. CRC Standard Mathematical Tables, 18th Edition)

For a 0.95 confidence level (a = 0.05), x_α is 1.645.

$$\chi^2 = \frac{1}{2}\left[1.645 + \sqrt{(2*112)-1}\right]^2 = 137.42$$

Therefore, if the number of observed valve failures is assumed to be fifty-five, the upper limit valve failure rate would be

$$\lambda_U = \left[\frac{1}{2*1.09E+8}\right]*137.42 = 6.30E\text{-}07 \text{ failures per hour}$$

New release frequency values can be calculated from these higher values for instrument line break frequency and EFCV failure rate. Using the GE notation,

$$RF = I * A$$

and,

$$A = \lambda_U \frac{\theta}{2}$$

Where,

RF	is release frequency per year
I	is instrument line break frequency per year
A	is EFCV unavailability (failure to close probability)
λ_U	is EFCV failure rate per hour
θ	is EFCV surveillance test interval in hours

Using a surveillance interval of two years, an instrument line break frequency of 35.2E-06 per year, and an EFCV failure rate of 6.30E-07 failures per hour,

$$RF = I * \lambda_U * \left(\frac{\theta}{2}\right) = \left(\frac{35.2\text{E}-06}{\text{yr}}\right)\left(\frac{6.30\text{E}-07}{\text{hr}}\right)\left(\frac{2\text{ yrs} * 8760\text{ hrs/yr}}{2}\right) = 19.4\text{E}-08\text{ events per year}$$

Inserting a surveillance interval of ten years,

$$RF = \left(\frac{35.2\text{E}-06}{\text{yr}}\right)\left(\frac{6.30\text{E}-07}{\text{hr}}\right)\left(\frac{10\text{ yrs} * 8760\text{ hrs/yr}}{2}\right) = 97.1\text{E}-08\text{ events per year}$$

Corresponding release frequencies reported in the GE Topical Report are 0.78E-08 events per year for two year surveillance intervals and 3.91E-08 events per year for ten year surveillance intervals (Table 4-3).

For a plant with 94 instrument lines (similar to the lead plant) with two year surveillance intervals, the total release frequency of instrument line breaks with failure to isolate is,

$$RF_{plant} = 94 * 19.4\text{E-08/yr} = 1.82\text{E-05 events per year}$$

For ten year surveillance intervals,

$$RF_{plant} = 94 * 97.1\text{E-08/yr} = 9.13\text{E-05 events per year}$$

These values are sufficiently low that it can be concluded that a change in surveillance test frequency has minimal impact on the valve reliability.

The impact of an increased estimation of instrument line rupture frequency and a five-fold increase in assumed number of EFCV failures on the likelihood of a release to the reactor building environs has been calculated. The total plant release frequency for a rupture of any instrument lines and a coincident failure of the line's EFCV to isolate the break flow is 9.13E-05 events per year, which is equivalent to approximately one event in ten thousand years. The conclusion that releases would be infrequent remains valid even with significantly different assumptions on break frequency and valve failure rates.

NRC Lead Plant Question 6:

Verify if there are valves in the plant that are similar to EFCVs whose failure data may be available. If such data exist, provide the data as well as the impact of applying such data on the release frequency estimate.

In addition, ensure that you have considered in your analysis any information available on degradation mechanism(s) and root cause(s) of the failed EFCVs (or similar valves) observed at other plants. Similarly, provide assurance that this type of information (including failure rates) will be shared among the plants for future data as they become updated and available.

Provide performance criteria for EFCVs. Describe how a cause determination will be performed and determine what specific corrective action would be taken if EFCVs do not meet their performance criteria.

BWROG Response to Question 6:

EFCVs are not typically used in other applications. The GE report provides the available failure information.

Sharing of significant data from any future failures would be through applicable industry generic communication mechanisms such as the Equipment Performance and Information Exchange (EPIX), Licensee Event Reporting system, or other operating experience forums. Each plant's corrective action programs must evaluate equipment failures and establish appropriate corrective actions.

April 12, 1999
NG-99-0308

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station 0-P1-17
Washington, DC 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Technical Specification Change Request (TSCR-010):
"Relaxation of Excess Flow Check Valve Surveillance Testing"

File: A-117

Dear Sir(s):

In accordance with the Code of Federal Regulations, Title 10, Sections 50.59 and 50.90, IES Utilities Inc. hereby requests revision to the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC).

Surveillance Requirement (SR) 3.6.1.3.7 currently requires verification of the actuation capability of each reactor instrumentation line Excess Flow Check Valve (EFCV) every 24 months. This proposed change is to relax the SR frequency by allowing a "representative sample" of EFCVs to be tested every 24 months, such that each 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 (snubbers) and Option B to 10 CFR 50 Appendix J.

The basis for this amendment is consistent with that described in a Boiling Water Reactor Owners' Group (BWROG) report, B21-00658-01, dated November 1998. However, this request has been tailored to the preferences stated by NRC Staff after reviewing that report at an August 6, 1998 meeting with the BWROG. In keeping with those preferences, the DAEC is submitting this request as a lead plant and a generic TS change has been initiated for NUREG 1433.

In addition, a revision to the TS BASES has been initiated pursuant to the BASES Control Program of TS 5.5.10 and 10 CFR 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.

The DAEC Operations Committee and the Safety Committee have reviewed this application. A copy of this submittal, along with the evaluation of No Significant Hazards Consideration, is being forwarded to our appointed state official pursuant to 10 CFR 50.91. We respectfully request a 60-day implementation period for this revision.

This letter is true and accurate to the best of my knowledge and belief.

IES UTILITIES INC.

By ______________________________

John F. Franz
Vice President, Nuclear

State of Iowa
(County) of Linn

Signed and sworn to before me on this ________ day of __________________, 1999,

By ____________________________.

____
Notary Public in and for the State of Iowa

____
Commission Expires

Attachments:
1) EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92
2) PROPOSED CHANGE TSCR-010 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS
3) SAFETY ASSESSMENT
4) ENVIRONMENTAL CONSIDERATION
5) BWROG REPORT B21-00658-01, "EXCESS FLOW CHECK VALVE TESTING RELAXATION," dated November 1998.

APPROVED LICENSING TOPICAL REPORT
INCLUDED AS SEPARATE ATTACHMENT #2

cc:
J. W. Karrick
E. Protsch (w/o)
D. Wilson (w/o)
B. Mozafari (NRC-NRR)
J. Dyer (Region III)
P. Baig (State of Iowa)
NRC Resident Office
Docu

EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92

Background:

DAEC Technical Specification Surveillance Requirement (SR) 3.6.1.3.7 currently requires verification of the actuation capability of each reactor instrumentation line Excess Flow Check Valve (EFCV) every 24 months. This proposed change is to relax the SR frequency by allowing a "representative sample" of EFCVs to be tested every 24 months, such that each 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 (snubbers) and Option B to 10 CFR 50 Appendix J.

The BWROG has issued a report that provides a basis for this request. This report (B21-00658-01 dated November 1998), enclosed as Attachment 5 to this submittal, provides justification for both relocation of this SR from TS to the Technical Requirements Manual (TRM) and a relaxation in the SR frequency as described above. The report demonstrates, through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. Reliability data in the report (Table 4-1) documents zero EFCV failures (to isolate) at the DAEC.

Members of the NRC Staff and the BWROG EFCV Committee met on August 6, 1998, to discuss the report contents. Based upon the outcome of this meeting, (documented in GE meeting summary OG98-0327-213, dated August 17, 1998) the DAEC is submitting this request as a lead plant. A generic TS change request has been initiated for BWR-4 (NUREG 1433) plants. Also in keeping with the August 6th meeting, this proposed change does not relocate the SR from the TS as justified in the report. This specific request is solely for the relaxation in the SR frequency as described above, with the SR remaining in the plant TS.

In addition, a revision to the TS BASES has been initiated pursuant to the BASES Control Program of TS 5.5.10 and 10 CFR 50.36(a) and is included to assist the Staff in its review of the proposed TS change. This TS BASES change is included for information only and is not considered part of this application for license amendment. This change includes the incremental testing (approximately 20% per cycle) and actions to be taken when test failures occur as discussed at the August 6th meeting with the Staff. While the generic change includes discussions regarding grouping of the EFCVs based on valve design and environmental conditions, since all the EFCVs at DAEC are of the same design with similar environmental conditions, sub-grouping of the valves is not considered necessary at the DAEC.

IES Utilities Inc., Docket No. 50-331,
Duane Arnold Energy Center, Linn County, Iowa
Date of Amendment Request: April 12, 1999

Description of Amendment Request:

The proposed amendment:

1. Relaxes the number of EFCVs tested every 24 months from "each" to a "representative sample" every 24 months. The representative sample is based on approximately 20% of the valves each cycle such that each valve is tested every 10 years (nominal).

Basis for proposed No Significant Hazards Consideration:

The Commission has provided standards (10 CFR Section 50.92(c)) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

After reviewing this proposed amendment, we have concluded:

1) 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 instrumentation line EFCV to be tested every 24 months. The EFCVs at DAEC are designed so that they will not close accidentally during normal operation, will close if a rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate, and have their status indicated in the control room (reference DAEC UFSAR 1.8.11). This proposed change allows a reduced number of 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 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.

   Instrument lines connecting to the Reactor Coolant Pressure Boundary (RCPB) with EFCVs installed also have a flow-restricting orifice upstream of the EFCV. The consequences of an unisolable rupture of such an instrument line has been previously evaluated in response to Regulatory Guide (RG) 1.11 (DAEC UFSAR 1.8.1.1). That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (3.5 hours). Therefore, 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.

2) 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 EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry operating experience demonstrates the high reliability of these valves. The potential failure of an EFCV to isolate by the proposed reduction in test frequency 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, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created.

3) The proposed amendment will not involve a significant reduction in a margin of safety.

   The consequences of an unisolable rupture of an instrument line has been previously evaluated in response to RG 1.11 (reference DAEC UFSAR 1.8.1.1). That evaluation assumed a continuous discharge of reactor water for the duration of the detection and cooldown sequence (3.5 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 an EFCV to isolate postulated by this reduced testing frequency is bounded and does not involve a significant reduction in the margin of safety.

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

Local Public Document Room Location: Cedar Rapids Public Library, 500 First Street SE, Cedar Rapids, Iowa 52401

Attorney for Licensee: Jack Newman, Al Gutterman; Morgan, Lewis & Bockius, 1800 M Street NW, Washington, D.C. 20036-5869

ATTACHMENT 2 TO PLA-5227

NO SIGNIFICANT HAZARDS CONSIDERATION
AND
EVNIRONMENTAL CONSIDERATION

NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL CONSIDERATION

NO SIGNIFICANT HAZARDS CONSIDERATIONS

PPL Susquehanna LLC Unit 1 and Unit 2 Technical Specification Surveillance Requirement (SR) 3.6.1.3.9 currently requires verification of the actuation capability of each reactor instrumentation line Excess Flow Check Valve (EFCV) every 24 months. The proposed change relaxes the SR frequency by allowing a "representative sample" of EFCVs to be tested every 24 months, such that each 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 (snubbers) and Option B to 10 CFR 50 Appendix J.

The BWROG has issued a report that provides a basis for this request. This report NEDO-332977-A dated June 2000 and enclosed with Attachment 1 to this submittal, provides justification for a relaxation in the SR frequency as described above. The report demonstrates, through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. PPL Susquehanna LLC has determined that the SSES test experience is consistent with the findings of the BWROG report.

PPL Susquehanna LLC has evaluated the changes proposed in accordance with the criteria specified by 10CFR50.92 and has determined that the proposed changes do not involve a significant hazards consideration. The criteria and conclusions of our evaluation are presented below.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This proposed changes do not involve an increase in the probability or consequences of an accident previously evaluated.

The current SR frequency requires each reactor instrumentation line EFCV to be tested every 24 months. The EFCVs at SSES Unit1 and Unit 2 are designed so that they will not close accidentally during normal operation, will close if a rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate, and have their status indicated in the control room. This proposed change allows a reduced number of EFCVs to be tested every 24 months. There are no physical plant modifications associated with this change. Industry and SSES operating experience demonstrates a high reliability of these valves. Neither 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.

The SSES FSAR Section 15.6.2 demonstrates (consistent with the BWROG report) that the failure of an EFCV has very low consequence. SSES FSAR Section 15.6.2 evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation assumes the EFCV fails to isolate the break. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10CFR100 limits. Thus the failure of an EFCV, though not expected as a result of this proposed change, does not affect the dose consequences of an instrument line break.

Based on the above, it is concluded that the proposed change to the EFCV surveillance requirement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do 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 EFCVs to be tested each operating cycle. No other changes in requirements are being proposed. Industry and Susquehanna-specific operating experience demonstrates the high reliability of these valves. The potential failure of an EFCV to isolate by the proposed reduction in test frequency 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, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created from implementation of the proposed change.

3. **The proposed change does not involve a significant reduction in the margin of safety.**

SSES FSAR Section 15.6.2 evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation assumes the EFCV fails to isolate the break. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10CFR100 limits. Thus the failure of an EFCV, though not expected as a result of this proposed change, does not affect the dose consequences of an instrument line break.

Therefore, this proposed change does not represent a significant reduction in the margin of safety.

ENVIRONMENTAL CONSIDERATION

An environmental assessment is not required for the proposed revisions because the requested revisions conform to the criteria for actions eligible for categorical exclusion as specified in 10CFR51.22(c)(9). The requested revisions will have no impact on the environment. As discussed above, the proposed revisions do not involve a significant hazard consideration. The proposed revisions do not involve a change in the types or increase in the amounts of effluents that may be released off-site. In addition, the proposed revisions do not involve an increase in the individual or cumulative occupational radiation exposure.

ATTACHMENT 3 TO PLA-5227

MARKED-UP TECHNICAL SPECIFICATION PAGES

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.1.3.6	------------------NOTE------------------ Only required to be met in MODES 1, 2 and 3. -- Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.9	Verify ~~each~~ a representative sample of reactor instrumentation line EFCVs actuate~~s~~ to check flow on a simulated instrument line break.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	-----------------NOTE----------------- Only required to be met in MODES 1, 2 and 3. --- Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.9 5	Verify ~~each~~ a representative sample of reactor instrumentation line EFCVs actuate~~s~~ to check flow on a simulated instrument line break.	24 months
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS

(continued)

ATTACHMENT 4 TO PLA-5227

"CAMERA-READY" TECHNICAL SPECIFICATION PAGES

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	------------------NOTE------------------ Only required to be met in MODES 1, 2 and 3. -- Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.9	Verify a representative sample of reactor instrumentation line EFCVs actuate to check flow on a simulated instrument line break.	24 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	------------------NOTE------------------ Only required to be met in MODES 1, 2 and 3. -- Perform leakage rate testing for each primary containment purge valve with resilient seals.	184 days
SR 3.6.1.3.7	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the Inservice Testing Program
SR 3.6.1.3.8	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.9	Verify a representative sample of reactor instrumentation line EFCVs actuate to check flow on a simulated instrument line break.	24 months
SR 3.6.1.3.10	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS

(continued)

ATTACHMENT 5 TO PLA-5227

TECHNICAL SPECIFICATION BASES MARKUPS

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.8 (continued)

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 SR 3.3.6.1.5 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that some of these Surveillances be performed only during a unit outage since isolation of penetrations could eliminate cooling water flow and disrupt the normal operation of some critical components. 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.

SR 3.6.1.3.9

This SR requires a demonstration that ~~each~~ a representative sample of reactor instrumentation line excess flow check valves (EFCV) ~~is~~ are OPERABLE by verifying that the valve actuates to check flow on a simulated instrument line break. As defined in FSAR Section 6.2.4.3.5 (Reference 4), the conditions under which an EFCV will isolate, simulated instrument line break, are at flow rates which develop a differential pressure of between 3 psid and 10 psid. This SR provides assurance that the instrumentation line EFCVs will perform its design function to check flow. No specific valve leakage limits are specified because no specific leakage limits are defined in the FSAR. The 24 month Frequency is based on the need to perform some of these Surveillances 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.

Insert

~~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.~~

SR 3.6.1.3.10

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this

(continued)

INSERT

The representative sample consists of an approximate equal number of EFCVs such that each EFCV is tested at least once every 10 years (nominal). The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potential common problem with a specific type or application of EFCV is detected at the earliest possible time. EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 7).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 3.3 gpm when tested at 1.1 P_a. (49.5 psig). The combined leakage rates must be demonstrated in accordance with the leakage rate test Frequency required by the Primary Containment Leakage Testing Program.

As noted in Table B 3.6.1.3-1, PCIVs associated with this SR are not Type C tested. Containment bypass leakage is prevented since the line terminates below the minimum water level in the Suppression Chamber. These valves are tested in accordance with the IST Program. Therefore, these valves leakage is not included as containment leakage.

This SR has been modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3, since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

REFERENCES

1. FSAR, Chapter 15.
2. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 6.2.
5. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
6. Standard Review Plan 6.2.4, Rev. 1, September 1975
7. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation", June 2000

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.7

The Frequency of this SR is in accordance with the requirements of the Inservice Testing Program.

SR 3.6.1.3.8

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 SR 3.3.6.1.5 overlaps this SR to provide complete testing of the safety function. The 24 month Frequency was developed considering it is prudent that some of these Surveillances be performed only during a unit outage since isolation of penetrations could eliminate cooling water flow and disrupt the normal operation of some critical components. 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.

SR 3.6.1.3.9

a representative sample of

This SR requires a demonstration that ~~each~~ reactor instrumentation line excess flow check valve s (EFCV) ~~is~~ are OPERABLE by verifying that the valve actuates to check flow on a simulated instrument line break. As defined in FSAR Section 6.2.4.3.5 (Reference 4), the conditions under which an EFCV will isolate, simulated instrument line breaks are at flow rates which develop a differential pressure of between 3 psid and 10 psid. This SR provides assurance that the instrumentation line EFCVs will perform its design function to check flow. No specific valve leakage limits are specified because no specific leakage limits are defined in the FSAR. The 24 month Frequency is based on the need to perform some of these Surveillances 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.

INSERT 1

~~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.~~

(continued)

INSERT

The representative sample consists of an approximate equal number of EFCVs such that each EFCV is tested at least once every 10 years (nominal). The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potential common problem with a specific type or application of EFCV is detected at the earliest possible time. EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint (Reference 7).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.12 (continued)

Containment Leakage Rate Testing Program. If leakage from the MSIVs requires internal work on any MSIV, the leakage will be reduced for the affected MSIV to ≤ 11.5 scfh.

SR 3.6.1.3.13

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is 3.3 gpm when tested at 1.1 P_a, (49.5 psig). The combined leakage rates must be demonstrated in accordance with the leakage rate test Frequency required by the Primary Containment Leakage Testing Program.

As noted in Table B3.6.1.3-1, PCIVs associated with this SR are not Type C tested. Containment bypass leakage is prevented since the line terminates below the minimum water level in the suppression chamber. These valves are tested in accordance with the IST Program. Therefore, these valves leakage is not included as containment leakage.

This SR has been modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3, since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.

REFERENCES

1. FSAR, Chapter 15.
2. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 6.2.
5. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
6. Standard Review Plan 6.2.4, Rev. 1, September 1975
7. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation", June 2000

ATTACHMENT 6 TO PLA-5227

IST PROGRAM RELIEF REQUESTS

REFUELING OUTAGE TEST JUSTIFICATION NUMBER 20

System	P&ID	Valve
RPV	M-142	XV-14201
		XV-14202
		XV-142F041
		XV-142F043A
		XV-142F043B
		XV-142F045A
		XV-142F045B
		XV-142F047A
		XV-142F047B
		XV-142F051A
		XV-142F051B
		XV-142F055
RHR	M-151	XV-15109A
		XV-15109B
CORE SPRAY	M-152	XV-152F018A
		XV-152F018B

Category: C

Class: 1

Function: Containment Isolation

Impractical Test Requirement:

1. Exercise test valve once per 92 days.
   (OMa – 1988 Part 10 Paragraph 4.3.2)

2. Valve Position Verification once every 2 years
   (OMa - 1988 Part 10 Paragraph 4.1)

Basis for Deferment: Excess flow check valves are installed on instrument lines penetrating containment in accordance with Regulatory Guide 1.11. As such, the lines are sized and/or orificed such that off-site doses will be substantially below 10CFR100 limits in the event of a rupture. Therefore, individual leak rate testing of these valves is not required for conformance with 10CFR50, Appendix J requirements.

The excess flow check valve is a simple device; the major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the tube. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve. Functional testing is required by Technical Specification SR 3.6.1.3.9.

Instruments serviced by these valves frequently have interlock or actuation functions that would be interfered with should testing be attempted during plant operation. This would impose unnecessary challenges to safety systems and would reduce the plant's margin of safety. Also, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side.

Industry experience as documented in NEDO-32977-A indicates that EFCVs have a very low failure rate. At Susquehanna the SR failure rate has been approximately 1%. Only half of these SR failures have resulted in replacement of the EFCV. The Susquehanna test history shows no evidence of common mode failure. This Susquehanna test experience is consistent with the findings of the NEDO. The NEDO indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCVs at Susquehanna, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Therefore, PPL Susquehanna LLC requests relief pursuant to 10CFR50.55a(a)(3)(i) to test excess flow check valves at the frequency specified in the Susquehanna Technical Specifications Surveillance Requirements (SR) 3.6.1.3.9. As discussed in the Technical Specification Bases for this SR, this test provides assurance that each valve actuates to check flow on a simulated instrument line break.

Alternative Testing:

Functional testing with verification that flow is checked will be performed per SR 3.6.1.3.9.

The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.9. After the close position test, the valves will be reset, and the remote open position indication will be verified. Inadvertent actuation of an EFCV during operation is highly unlikely due to the spring-poppet design, Susquehanna verifies the EFCVs indicate open in the control room at a frequency greater than once every 2 years.

RELIEF REQUEST NUMBER 23

System	P&ID	Valve	System	P&ID	Valve
RPV	M-141	XV-141F009	RPV (cont'd)	M-142	XV-142F059B
Main Steam	M-141	XV-141F070A			XV-142F059C
		XV-141F070B			XV-142F059D
		XV-141F070C			XV-142F059E
		XV-141F070D			XV-142F059F
		XV-141F071A			XV-142F059G
		XV-141F071B			XV-142F059H
		XV-141F071C			XV-142F059L
		XV-141F071D			XV-142F059M
		XV-141F072A			XV-142F059N
		XV-141F072B			XV-142F059P
		XV-141F072C			XV-142F059R
		XV-141F072D			XV-142F059S
		XV-141F073A			XV-142F059T
		XV-141F073B			XV-142F059U
		XV-141F073C			XV-142F061
		XV-141F073D	RXR	M-143	XV-143F003A
RPV	M-142	XV-142F051C			XV-143F003B
		XV-142F051D			XV-143F004A
		XV-142F053A			XV-143F004B
		XV-142F053B			XV-143F009A
		XV-142F053C			XV-143F009B
		XV-142F053D			XV-143F009C
		XV-142F057			XV-143F009D
		XV-142F059A			XV-143F010A

RELIEF REQUEST NUMBER 23 (Cont'd)

System	P&ID	Valve	System	P&ID	Valve
RXR (Cont'd)	143	XV-143F010B	RCIC	M-149	XV-149F044A
		XV-143F010C			XV-149F044B
		XV-143F010D			XV-149F044C
		XV-143F011A			XV-149F044D
		XV-143F011B	HPCI	M-155	XV-155F024A
		XV-143F011C			XV-155F024B
		XV-143F011D			XV-155F024C
		XV-143F012A			XV-155F024D
		XV-143F012B	RHR	M-151	XV-15109C
		XV-143F012C			XV-15109D
		XV-143F012D			
		XV-143F040A			
		XV-143F040B			
		XV-143F040C			
		XV-143F040D			
		XV-143F057A			
		XV-143F057B			
RWCU	M-144	XV-14411A			
		XV-14411B			
		XV-14411C			
		XV-14411D			
		XV-144F046			

RELIEF REQUEST NUMBER 23 (Cont'd)

Category:	C
Class:	1
Function:	Containment Isolation
Impractical Test Requirement:	1. Exercise test valve one per 92 days. (OMa – 1988 Part 10 paragraph 4.3.2) 2. Valve Position Verification once every 2 years (OMa - 1988 Part 10 Paragraph 4.1)
Basis for Deferment:	Excess flow check valves are installed on instrument lines penetrating containment in accordance with Regulatory Guide 1.11. The lines are sized and/or orificed such that off-site doses will be substantially below 10CFR100 limits in the event of a rupture. Therefore, individual leak rate testing of these valves is not required for conformance with 10CFR50, Appendix J requirements. The excess flow check valve is a simple device; the major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the tube. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve. Functional testing is required by Technical Specification SR 3.6.1.3.9. Systems design does not include test taps upstream of the Excess Flow Check Valves. For this reason, the EFCV's cannot be isolated and tested using a pressure source other than reactor pressure.

The testing described above requires the removal of the associated instrument or instruments from service. Since these instruments are in use during plant operation, removal of any of these instruments from service may cause a spurious signal which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side and also will contribute to an increase in personnel radiation exposure.

Industry experience as documented in NEDO-32977-A indicates that EFCVs have a very low failure rate. At Susquehanna the SR failure rate has been approximately 1%. Only half of these SR failures have resulted in replacement of the EFCV. The Susquehanna test history shows no evidence of common mode failure. This Susquehanna test experience is consistent with the findings of the NEDO. The NEDO indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCVs at Susquehanna, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Therefore, PPL Susquehanna LLC requests relief pursuant to 10CFR50.55a(a)(3)(i) to test excess flow check valves at the frequency specified in the Susquehanna Technical Specifications Surveillance Requirements (SR) 3.6.1.3.9. As discussed in the Technical Specification Bases for this SR, this test provides assurance that each valve actuates to check flow on a simulated instrument line break.

Testing on a Cold Shutdown frequency is impractical considering the large number of valves to be tested and the condition that reactor pressure >500 psig is needed for testing. NUREG-1482 allows test deferrals to refueling outages if it is impractical to test quarterly or during cold shutdowns. In this instance, considering the large number of valves to be tested and the conditions required for testing (Reactor pressure), it is also a hardship to test all these valves during refueling outages. Recent improvements in Refueling Outage schedules (i.e. shorter outages) minimized the time that is planned for Refueling and testing activities during the outages. The appropriate time for performing these excess flow check valves tests during refueling outages is in conjunction with vessel hydrostatic testing. As a result of shorter outages, decay heat levels during hydrostatic tests are higher than in the past. If the hydrostatic test was extended to test all EFCV's, the vessel could require depressurization several times to avoid exceeding the maximum bulk coolant temperature limit. This is an evolution which challenges the reactor operators and thermally cycles the reactor vessel and should be avoided if possible. Also, based on past experience, excess flow check valve testing during hydrostatic testing becomes the outage critical path and could possibly extend the outage by 2 days if all EFCV's were to be tested during this time frame.

A proposed alternative to testing during the refueling outage would be to test certain excess flow check valves immediately preceding the refueling outage while the reactor is at power, while also instituting the appropriate administrative and scheduling controls. This provides the appropriate conditions for testing (Reactor pressure >500 psig), while also providing an acceptable level of quality and safety. Performance of the excess flow check valve testing prior to the outage will be scheduled such that, in the event of a failure, the resulting action statement and limiting condition of operation will encompass the planned shutdown for the refueling outage. Using this strategy, unplanned, unnecessary plant shutdowns as a result of excess flow check valve testing will be avoided.

In summary, considering the extremely low failure rate, personnel and plant safety concerns, the hardship of testing during refueling outages, EFCV testing during refueling outages is impractical and results in a hardship without a compensating increase in the level of safety.

Alternate Testing:

Functional testing with verification that flow is checked will be performed per TS 3.6.1.3.9, immediately preceding a planned Refueling Outage and with the appropriate administrative and scheduling controls established.

The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.9. After the close position test, the valves will be reset, and the remote open position indication will be verified. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring-poppet design, Susquehanna verifies the EFCVs indicate open in the control room at a frequency greater than once every 2 years.

REFUELING OUTAGE TEST JUSTIFICATION NUMBER 20

System	P&ID	Valve	System	P&ID	Valve
RPV	M-2142	XV-24201	RPV (continued)	M-2142	XV-242F047B
		XV-24202			XV-242F051A
		XV-242F041			XV-242F051B
		XV-242F043A			XV-242F055
		XV-242F043B	RHR	M-2151	XV-25109A
		XV-242F045A			XV-25109B
		XV-242F045B	Core Spray	M-2152	XV-252F018A
		XV-242F047A			XV-252F018B

Category: C

Class: 1

Function: Containment Isolation

Impractical Test Requirement:

1. Exercise test valve once per 92 days. (OMa – 1988 Part 10 paragraph 4.3.2)
2. Valve Position Verification once every 2 years (OMa - 1988 Part 10 Paragraph 4.1)

Basis for Deferment:

Excess flow check valves are installed on instrument lines penetrating containment in accordance with Regulatory Guide 1.11. As such, the lines are sized and/or orificed such that off-site doses will be substantially below 10CFR100 limits in the event of a rupture. Therefore, individual leak rate testing of these valves is not required for conformance with 10CFR50, Appendix J requirements.

The excess flow check valve is a simple device; the major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the tube. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve. Functional testing is required by Technical Specification SR 3.6.1.3.9.

Instruments serviced by these valves frequently have interlock or actuation functions that would be interfered with should testing be attempted during plant operation. This would impose unnecessary challenges to safety systems and would reduce the plant's margin of safety. Also, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side.

Industry experience as documented in NEDO-32977-A indicates that EFCVs have a very low failure rate. At Susquehanna the SR failure rate has been approximately 1%. Only half of these SR failures have resulted in replacement of the EFCV. The Susquehanna test history shows no evidence of common mode failure. This Susquehanna test experience is consistent with the findings of the NEDO. The NEDO indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCVs at Susquehanna, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Therefore, PPL Susquehanna LLC requests relief pursuant to 10CFR50.55a(a)(3)(i) to test excess flow check valves at the frequency specified in the Susquehanna Technical Specifications Surveillance Requirements (SR) 3.6.1.3.9. As discussed in the Technical Specification Bases for this SR, this test provides assurance that each valve actuates to check flow on a simulated instrument line break.

Alternative Testing:	Functional testing with verification that flow is checked will be performed per SR 3.6.1.3.9. The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.9. After the close position test, the valves will be reset, and the remote open position indication will be verified. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring-poppet design, Susquehanna verifies the EFCVs indicate open in the control room at a frequency greater than once every 2 years.

RELIEF REQUEST NUMBER 23

System	P&ID	Valve	System	P&ID	Valve
RPV	M-2141	XV-241F009	RPV (continued)		XV-242F059B
Main Steam	M-2141	XV-241F070A			XV-242F059C
		XV-241F070B			XV-242F059D
		XV-241F070C			XV-242F059E
		XV-241F070D			XV-242F059F
		XV-241F071A			XV-242F059G
		XV-241F071B			XV-242F059H
		XV-241F071C			XV-242F059L
		XV-241F071D			XV-242F059M
		XV-241F072A			XV-242F059N
		XV-241F072B			XV-242F059P
		XV-241F072C			XV-242F059R
		XV-241F072D			XV-242F059S
		XV-241F073A			XV-242F059T
		XV-241F073B			XV-242F059U
		XV-241F073C			XV-242F061
		XV-241F073D	RXR	M-2143	XV-243F003A
RPV	M-2142	XV-242F051C			XV-243F003B
		XV-242F051D			XV-243F004A
		XV-242F053A			XV-243F004B
		XV-242F053B			XV-243F009A
		XV-242F053C			
		XV-242F053D			
		XV-242F057			
		XV-242F059A			

RELIEF REQUEST NUMBER 23 (Cont'd.)

System	P&ID	Valve	System	P&ID	Valve
RXR (cont'd)	M-2143	XV-243F009B	RCIC	M-1249	XV-249F044A
		XV-243F009C			XV-249F044B
		XV-243F009D			XV-249F044C
		XV-243F010A			XV-249F044D
		XV-243F010B	HPCI	M-2155	XV-255F024A
		XV-243F010C			XV-255F024B
		XV-243F010D			XV-255F024C
		XV-243F011A			XV-255F024D
		XV-243F011B	RHR	M-2151	XV-25109C
		XV-243F011C			XV-25109D
		XV-243F011D			
		XV-243F012A			
		XV-243F012B			
		XV-243F012C			
		XV-243F012D			
		XV-243F040A			
		XV-243F040B			
		XV-243F040C			
		XV-243F040D			
		XV-243F057A			
		XV-243F057B			
RWCU	M-2144	XV-24411A			
		XV-24411B			
		XV-24411C			
		XV-24411D			
		XV-244F046			

RELIEF REQUEST NUMBER 23 (Cont'd.)

Category: C

Class: 1

Function: Containment Isolation

Impractical Test Requirement:

1. Exercise test valve once per 92 days. (OMa – 1988 Part 10 paragraph 4.3.2)

2. Valve Position Verification once every 2 years (OMa - 1988 Part 10 Paragraph 4.1)

Basis for Deferment:

Excess flow check valves are installed on instrument lines penetrating containment in accordance with Regulatory Guide 1.11. The lines are sized and/or orificed such that off-site doses will be substantially below 10CFR100 limits in the event of a rupture. Therefore, individual leak rate testing of these valves is not required for conformance with 10CFR50, Appendix J requirements.

The excess flow check valve is a simple device; the major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the tube. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve. Functional testing is required by Technical Specification SR 3.6.1.3.9. Systems design does not include test taps upstream of the Excess Flow Check Valves. For this reason, the EFCV's cannot be isolated and tested using a pressure source other than reactor pressure.

The testing described above requires the removal of the associated instrument or instruments from service. Since these instruments are in use during plant operation, removal of any of these instruments from service may cause a spurious signal which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process liquid will be contaminated to some degree, requiring special measures to collect flow from the vented instrument side and also will contribute to an increase in personnel radiation exposure.

Industry experience as documented in NEDO032977-A indicates that EFCVs have a very low failure rate. At Susquehanna the SR failure rate has been approximately 1%. Only half of these SR failures have resulted in replacement of the EFCV. The Susquehanna test history shows no evidence of common mode failure. This Susquehanna test experience is consistent with the findings of the NEDO. The NEDO indicates similarly that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCVs at Susquehanna, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Therefore, PPL Susquehanna LLC requests relief pursuant to 10CFR50.55a(a)(3)(I) to test excess flow check valves at the frequency specified in the Susquehanna Technical Specifications Surveillance Requirements (SR) 3.6.1.3.9. As discussed in the Technical Specification Bases for this SR, this test provides assurance that each valve actuates to check flow on a simulated instrument line break.

Testing on a Cold Shutdown frequency is impractical considering the large number of valves to be tested and the condition that reactor pressure >500 psig is needed for testing.

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NUREG-1482 allows test deferrals to refueling outages if it is impractical to test quarterly or during cold shutdowns. In this instance, considering the large number of valves to be tested and the conditions required for testing (Reactor pressure), it is also a hardship to test all these valves during refueling outages. Recent improvements in Refueling Outage schedules (i.e. shorter outages) minimized the time that is planned for Refueling and testing activities during the outages. The appropriate time for performing these excess flow check valve tests during refueling outages is in conjunction with vessel hydrostatic testing. As a result of shorter outages, decay heat levels during hydrostatic tests are higher than in the past. If the hydrostatic test was extended to test all EFCV's, the vessel could require depressurization several times to avoid exceeding the maximum bulk coolant temperature limit of 212 degrees F. This is an evolution which challenges the reactor operators and thermally cycles the reactor vessel and should be avoided if possible. Also, based on past experience, excess flow check valve testing during hydrostatic testing becomes the outage critical path and could possibly extend the outage by 2 days if all EFCV's were to be tested during this time frame.

A proposed alternative to testing during the refueling outage would be to test certain excess flow check valves immediately preceding the refueling outage while the reactor is at power, while also instituting the appropriate administrative and scheduling controls. This provides the appropriate conditions for testing (Reactor pressure >500 psig), while also providing an acceptable level of quality and safety. Performance of the excess flow check valve testing prior to the outage will be scheduled such that, in the event of a failure, the resulting action statement and limiting condition of operation will encompass the planned shutdown for the refueling outage. Using this strategy, unplanned, unnecessary plant shutdowns as a result of excess flow check valve testing will be avoided.

In summary, considering the extremely low failure rate, personnel and plant safety concerns, the hardship of testing during refueling outages, EFCV testing during refueling outages is impractical and results in a hardship without a compensating increase in the level of safety.

Alternative Testing:

Functional testing with verification that flow is checked will be performed TS 3.6.1.3.9, immediately preceeding a planned Refueling Outage and with the appropriate administrative and scheduling controls established.

The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.9. After the close position test, the valves will be reset, and the remote open position indication will be verified. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring-poppet design, Susquehanna verifies the EFCVs indicate open in the control room at a frequency greater than once every 2 years.