

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.

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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.

cc: J. W. Karrick
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EVALUATION OF CHANGE PURSUANT TO 10 CFR SECTION 50.92Background:

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.

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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:

<u>Page</u>	<u>Description of Changes</u>
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.



GE Nuclear Energy

B21-00658-01
Class I
November 1998

Excess Flow Check Valve Testing Relaxation

**Prepared for:
Excess Flow Check Valve Committee
BWR Owners' Group**

**Prepared by: PD Knecht
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**Project Manager: original signed by Rick Hill
R.A. Hill**

**GE Nuclear Energy
San Jose, California
November 16, 1998**



IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of General Electric Company (GE) respecting information in this document are contained in the contract between the Boiling Water Reactors Owners' Group (BWROG) and GE, as identified in the respective utilities' BWROG Standing Purchase Order for the performance of the work described herein, and nothing in this document shall be construed as changing those individual contracts. The use of this information, except as defined by said contracts, or for any purpose other than that for which it is intended, is not authorized, and with respect to any other unauthorized use, neither GE, nor any of the contributors to this document makes any representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

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This report is for your use. Comments regarding the appropriateness and completeness of the schedule and the good practices provided in this report should be directed to the GE Project Manager of the BWROG Excess Flow Check Valve Committee at Mail Code 182, GE Nuclear Energy, 175 Curtner Avenue, San Jose, CA 95125.



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 lies 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	.75	
In-situ Testing	92	.6	
Bench Testing	224	0	
Instrument line refill	38	.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	<i>Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.</i>	EFCVs do not satisfy this criterion
Criterion 2	<i>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.</i>	EFCVs do not satisfy this criterion
Criterion 3	<i>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.</i>	EFCV closure is not needed for accident mitigation (see Section 3)
Criterion 4	<i>A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.</i>	EFCV closure is not significant to public health and safety (see 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) <i>Should satisfy the requirements for redundancy, independence, and testability of the protection system</i>	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) <i>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,</i> <i>(1) the leakage is reduced to the maximum extent practical consistent with other safety requirements,</i> <i>(2) the rate and extent of coolant loss are within the capability of the reactor coolant makeup system,</i> <i>(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</i> <i>(4) the potential offsite exposure will be substantially below the guidelines of 10CFR100.</i>	This requirement is not applicable to EFCV testing. Item b) (3) may not be satisfied if the EFCVs fail to function. (See section 3.2) Section 3 and Attachment B demonstrate that the designs generally meet the intent of this requirement.



<p>Lines that are part of the protection system and Lines that are not part of the protection system</p>	<p>c) <i>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></i></p> <ul style="list-style-type: none"> (1) <i>will not close accidentally during normal reactor operation,</i> (2) <i>will close or be closed if the instrument line integrity outside containment is lost during normal reactor operation or under accident conditions, and</i> (3) <i>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.</i> 	<p>EFCVs generally comply with this requirement.</p> <p>Reliability of EFCVs is discussed further in Section 4.</p>
<p>Lines that are part of the protection system and Lines that are not part of the protection system</p>	<p>d) <i>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.</i></p>	<p>Instrument line separation generally addresses this requirement.</p> <p>This requirement is not applicable to EFCV testing.</p>
<p>Lines that are part of the protection system and Lines that are not part of the protection system</p>	<p>e) <i>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.</i></p>	<p>Instrument line orificing generally does not impact the performance of the instrumentation.</p> <p>This requirement is not applicable to EFCV testing.</p>

As shown in Table 2-2 the design criteria of RG 1.11 are generally met in 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 (see 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 withdraw 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 HCU's 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), and
- 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 1/4" 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 (see Attachment B) indicate that even without a 1/4" 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 1/4" 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 standby gas treatment system



(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 9 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:

$$\lambda_{U} = \frac{1}{2T} \chi_{\alpha; 2r+2}^2 \quad (1)$$

Where:

λ_{U} is the upper limit failure rate per hour

T is the operating time in hours

r is the number of failures

$\chi_{\alpha; 2r+2}^2$ is 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 (see 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:

$$RF = I * \bar{A} \quad (2)$$

and $\bar{A} = \lambda \frac{\theta}{2} \quad (3)$

Where:

- RF is the release frequency per year
- I is the instrument line break frequency per year (Section 4.3)
- \bar{A} is the EFCV unavailability (failure to close probability)
- λ is the EFCV failure rate per hour (Section 4.2)
- θ is the 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 months surveillance test interval $86 * 5.86E-9 = 5.04E-7$ events/year



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

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

<i>Plant [Note 1]</i>	<i>Make of EFCV</i>	<i>Operating Time [years] [Note 2]</i>	<i>Operating Time [hours] [Note 2]</i>	<i>Number of Failures</i>	<i>Best Estimate Failure Rate [1/h] [Note 3]</i>	<i>Upper Limit Failure Rate [1/h]</i>	<i>Notes</i>
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 \bar{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 1/4" 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 $\sim 3E-5$ 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 $\sim 2E-4$ 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 bases 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 and 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 tests of the EFCVs.

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 basis 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 EFCVs 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

<i>Plant</i>	<i>Utility</i>	<i>BWR type</i>	<i>Containment type</i>
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

<i>Plant - units</i>	<i>Total No. of EFCVs (all units)</i>	<i>Valve years</i>	<i>Pressure rating (psig)</i>	<i>Make and model</i>	<i>Typical Installation</i>	<i>Nominal line size</i>	<i>Internal orifice size</i>	<i>Minimum flow diameter</i>	<i>Flow controlled by body contour?</i>	<i>Operating pressure (psig)</i>	<i>Operating temp deg F</i>
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 - 1/4" 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	1/4"	0.25"	No	1050	575 max, 60 operating
Dresden 2,3	150	922	3000	Chemquip 50FM-11WISS-B 50FM-920704	Instrument lines outside of containment	1/2"	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			Reset orifice-025/014-		Poppet config, port size, spring tension	1650, 1250, 45	575, 575, 340



GE Nuclear Energy

				A11935-3,5			.016/.0225/ .022-.028		control closure 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

<i>Plant</i>	<i>Objective of test</i>	<i>Test method</i>	<i>Acceptance criteria</i>	<i>When test? Is it on critical path?</i>	<i>Exposure</i>	<i>Manhours</i>	<i>Test interval</i>
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

<i>Plant</i>	<i>EFCVs tested</i>	<i>Failures</i>	<i>Types of failures</i>	<i>Multiple failures?</i>	<i>Common cause?</i>	<i>Failure trend?</i>	<i>% requiring retest</i>	<i>Failure rate</i>
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) 20 failures occurred at U2 restart in 1993, 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

<i>Plant</i>	<i>Are EFCVs in a PM program?</i>	<i>Failed valves repaired or replaced?</i>	<i>What would reduce EFCV test burden?</i>
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

<i>Plant</i>	<i>Committed to GDC 55/56?</i>	<i>Tech Spec Rqmnt for testing?</i>	<i>Effective Inst Line break size</i>	<i>Other licensing commitments</i>	<i>Active or passive components, and why</i>
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 scrambling and shutting down the reactor. The reactor follows a standard cool down 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 $\mu\text{Ci/gm}$ 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 (μ Ci/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/m ³ 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

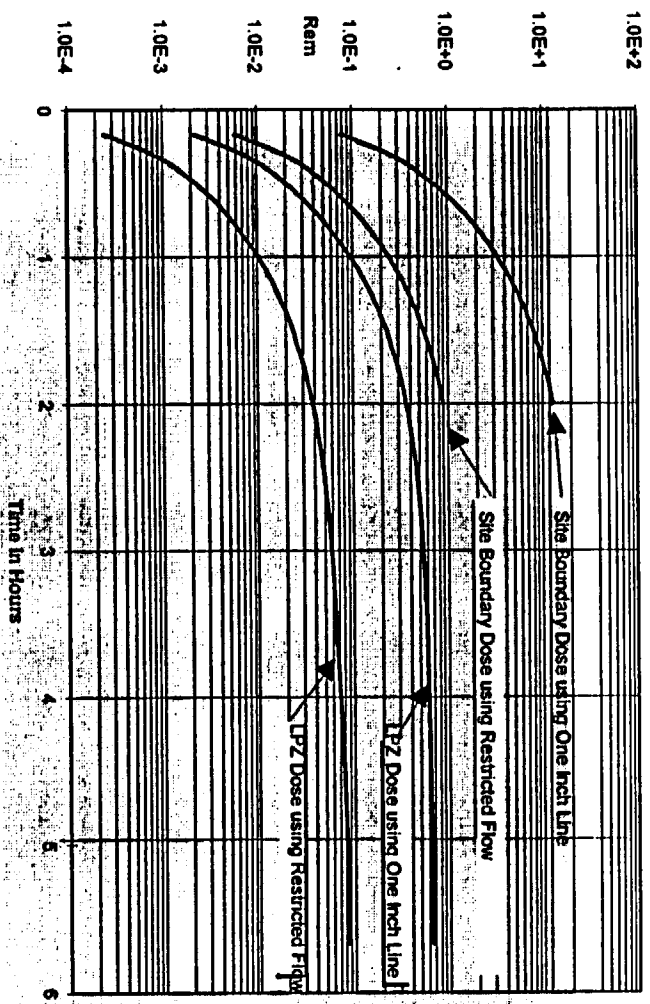
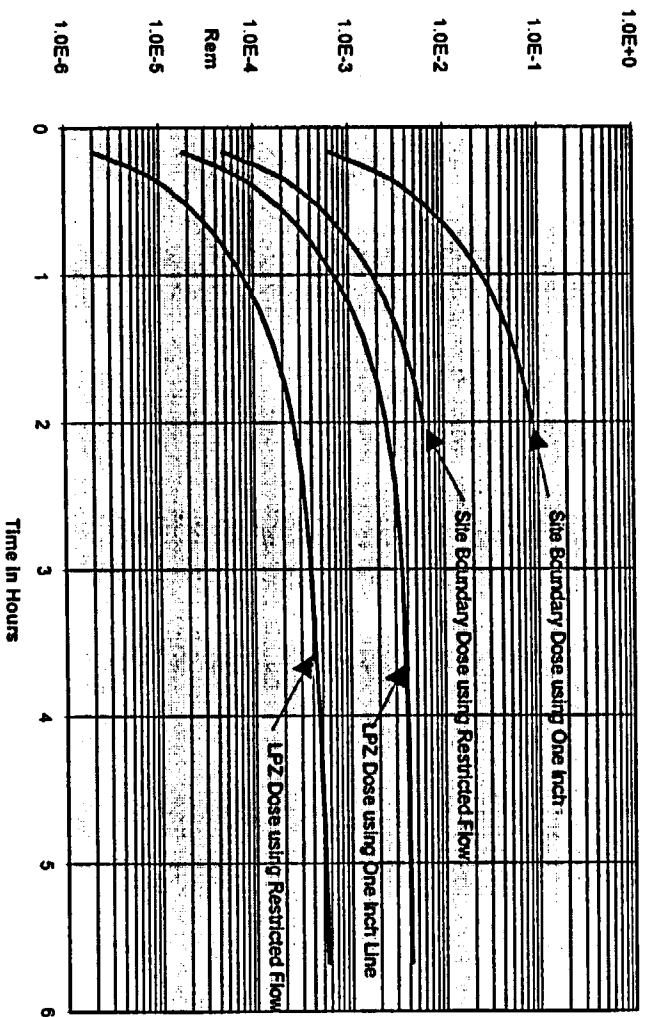


Figure B-2
Whole Body Dose





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.



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,

A handwritten signature in cursive script that reads "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.

October 5, 1999
NG-99-1358
Page 2

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 contained 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 Excess 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

6/2/99

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

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 3.6.1.3.7NOTE..... <div style="border: 1px solid black; padding: 5px; width: fit-content; margin: 10px auto;"> Only required to be met in MODES 1, 2 and 3. </div> 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 \geq [2] seconds and \leq [8] seconds.	<div style="border: 1px solid black; padding: 5px; width: fit-content; margin: 10px auto;"> In accordance with the Inservice Testing Program or 18 months </div>
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 reactor instrumentation line EFCV actuates [on a simulated instrument line break to restrict flow to \leq 1 gph]. <i>(S)</i>	[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

INSERT 1

This SR requires a demonstration that ~~each reactor~~ ⁵ instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve [reduces flow to ≤ 1 gph on a simulated instrument line break]. 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.

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)



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

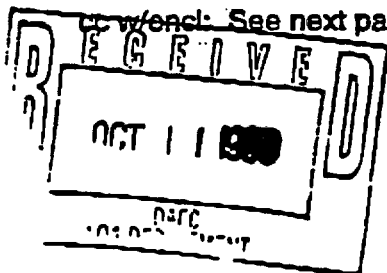
Brenda L. Mozafari, Project Manager, Section 1 Project Directorate III Division of Licensing Project Management Office of Nuclear Reactor Regulation

- NUCLEAR LICENSING Resp. Dues Yes* Post. Req'd NO VP-Nuc. Div. Asst: VP Nuc. Plant Mgr. Mgr.-Reg. Perf. Mgr.-Eng. Mgr.-EP Mgr.-Training Mgr.-Business Mgr.-Dist. Supp. Mgr.-QA CIPCO Combatt GDS A. Gutterman AR Coord. Security Rad. Pro. Maint. Operations Proj. Engr. Syst. Engr. Prog. Engr. Licensing Safety Comm. CFC TMAP Coord. Mat. Mgmt. OPS SUPPORT

Docket No. 50-331

Enclosure: As stated

cc w/enc: See next page



* Response Completed 09-13-99

**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.

October 8, 1999

NG-99-1383

Page 2

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.

$$1.2 * 6.7E-10/\text{hr-section} = 8.04E-10 \text{ failures per hr per section}$$

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

$$5 \text{ sections} * 8.04E-10/\text{hr-section} * 8760 \text{ hrs/yr} = 35.2E-06 \text{ failures per year}$$

This value is 6.6 times greater than the value of $5.34E-06/\text{yr}$ 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:

$$\lambda_U = \frac{1}{2T} \chi_{\alpha; 2r+2}^2$$

Where:

T is the operating time in hours

r is the number of failures

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

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,

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

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 ($\alpha = 0.05$), 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

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

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,

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

Inserting a surveillance interval of ten years,

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

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,

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

For ten year surveillance intervals,

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

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", 1/2", 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
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MA05421)."**

File: A-107a, A-117