BWR OWNERS' GROUP

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BWROG-00001
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Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
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Mail Station 0-P1-17
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Subject: BWR Owners' Group Generic Response to NRC Request for Additional Information on Lead Plant Technical Specification Change Request Regarding Excess Flow Check Valve Surveillance Requirements.

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

Dear Sir(s):

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

Very truly yours,

W.G. Warren, Chairman
BWR Owners' Group

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

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PDR TOPRP

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

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

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

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

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

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

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

NRC Lead Plant Question 1:

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

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

BWROG Response to Question 1:

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

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

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

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

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

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

NRC Lead Plant Question 2:

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

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

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

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

BWROG Response to Question 2:

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

NRC Lead Plant Question 3:

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

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

BWROG Response to Question 3:

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

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

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

NRC Lead Plant Question 4:

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

BWROG Response to Question 4:

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

NRC Lead Plant Question 5:

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

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

Describe the impact/change on the release frequency estimate if

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

BWROG Response to Question 5:

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

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

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

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

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

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

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

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

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

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

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

Where:

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

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

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

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

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

Where:

x_α	is the α-point of the standard normal distribution
n	is the degrees of freedom

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

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

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

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

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

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

$$RF = I * A$$

and,

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

Where,

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

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

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

Inserting a surveillance interval of ten years,

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

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

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

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

For ten year surveillance intervals,

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

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

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

NRC Lead Plant Question 6:

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

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

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

BWROG Response to Question 6:

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

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