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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 230 TO FACILITY OPERATING LICENSE NO. DPR-49

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated April 12, 1999, as supplemented by letters dated October 5 and 8,1999, IES Utilities Inc, the licensee for Duane Arnold Energy Center (DAEC), requested a license amendment to allow relaxation of the frequency of surveillance testing of excess flow check valves (EFCVs) in reactor instrumentation lines. The proposed change to the DAEC Technical Specifications (TSs) would relax the surveillance frequency to allow testing of a "representative sample" of EFCVs every 24 months, rather than testing each EFCV every 24 months, as currently required by TS. The licensee's intent is to test approximately 20% of the EFCVs each 24 months such that each EFCV will be tested at least once every 10 years (nominal). The licensee states that its basis for the request is a high degree of reliability associated with the EFCVs and the low consequences from an EFCV failure. The analysis to support this conclusion was based on the Boiling Water Reactor (BWR) Owners Group Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" by General Electric Nuclear Energy. The October 5 and 8, 1999, letters provided clarifying information that was within the scope of the original *Federal Register* notice, and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

EFCVs in reactor instrumentation lines are used in BWR containments to limit the release of fluid from the reactor coolant system in the event of an instrument line break. Examples of EFCVs include Reactor Pressure Vessel Level/Pressure instrument, Main Steam Line Flow instrument, Recirculation Pump Suction Pressure instrument, and Reactor Core Isolation Cooling Steam Line Flow instrument. EFCVs are not required to close in response to a containment isolation signal and are not postulated to operate under post-LOCA conditions. The topical report states that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with a design basis LOCA would be of a sufficiently low probability to be outside of the design basis.

DAEC TS Surveillance Requirement (SR) 3.6.1.3.7 currently requires verification of the actuation (closing) capability of each reactor instrumentation line EFCV every 24 months. This is typical for BWR TS. The proposed change would relax the SR frequency by allowing a "representative sample" of EFCVs to be tested every 24 months. The licensee's intent is to test approximately 20% of the EFCVs every 24 months such that each EFCV will be tested at least once every 10 years (nominal). The proposed change is similar in principle to existing performance-based testing programs, such as Inservice Testing of snubbers and Option B of Appendix J to 10 CFR Part 50.

Licensees make changes to their TS Bases sections without the need for prior NRC review or approval. Nevertheless, the licensee has included in its submittal, for information, a revised Basis for TS 3.6.1.3.7. The revised Basis states, in part:

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

3.0 EVALUATION

3.1 Systems Review

The topical report provides detailed information about EFCV surveillance testing at 12 BWR plants. Testing history indicates that there is a low failure rate in EFCV surveillance testing (see Section 3.2.1, below). At DAEC, there have been no failures in approximately 25 years of testing 94 valves. Thus, EFCVs have been very reliable performers, in general, and notably so at DAEC.

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

3.1.1 Request For Additional Information

We sent the licensee a request for additional information (RAI) dated September 27, 1999, regarding certain system aspects of the TS amendment application. The licensee responded on October 5, 1999. The following three sections discuss these issues.

3.1.1.1 Test Interval Increase

The topical report, and the licensee's plant-specific submittal, compare this situation to Option B of Appendix J to 10 CFR Part 50. We revised Appendix J in 1995 by adding Option B, which provides a risk-informed, performance-based approach to leakage rate testing of containment isolation valves. We also developed Regulatory Guide 1.163, "Performance-Based

Containment Leak Test Program," dated September 1995, as a method acceptable to us for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to us for complying with Option B, with four exceptions that are described in the Regulatory Guide.

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

Topical Report B21-00658-01 states that the NEI document allows a 10-year test interval, and that Regulatory Guide 1.163 endorsed the NEI document, without mentioning our exception to 10 years. We asked the licensee to justify its proposal for a 10-year testing interval. Its response stated that the comparison to Option B of Appendix J was conceptual and that the topical report established its own basis for the testing relaxation, i.e., high reliability, low risk, and low radiological consequences. Nevertheless, the licensee addressed the Regulatory Guide's reasons for not accepting a 10-year interval as follows:

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.

2. Unquantified leakage rates for test failures are not applicable because the maximum leakage through an unisolated instrument line is quantified as discussed in UFSAR [Updated Final Safety Analysis Report] 1.8.11. The dose consequences of the failure to isolate, as discussed in UFSAR 1.8.11, are acceptable (see Section 3.2.2, below).

3. Repetitive/common-mode failures are not applicable as evidenced by the low industry failure rate and more specifically by the topical report, Table 4-2, "EFCV Failure Rates by Manufacturer."

- Aging effects are not a concern. The industry data already provided does not indicate any increase in failure rate with time in service.

- Historical performance data associated with EFCVs has been provided and is considered adequate to justify the proposed interval.

- There is no indeterminate time period involved with this proposed change. Every 24 months, approximately 20% of the total population (e.g., about 19 valves at DAEC) will be tested.

We generally agree with the licensee's assessment, except for the aging question that is addressed below in Section 3.2.1. Regulatory Guide 1.163 considered all varieties of containment isolation valves, from a fraction of an inch to several feet in diameter, carrying liquid or gas in a wide range of temperatures and pressures. It had to account for different types of valves, e.g., gate, globe, check, made of various materials, by different manufacturers, and with varying levels of safety significance. On the other hand, EFCVs in reactor instrumentation lines are a very specific, narrow class of valves. Their history and performance are well-documented. Based on their historically high reliability and their low risk significance and radiological consequences should they fail, we accept the proposed extended test intervals.

3.1.1.2 Failure Feedback Mechanism

We pointed out to the licensee that, under Appendix J, Option B, testing programs, a valve that fails a test after having been put on an extended test interval must return to its original interval until it once again shows good performance (i.e., passes two consecutive tests). Risk-informed inservice testing Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," also specifies the need for a failure feedback mechanism. Topical Report B21-00658-01 has no specific failure feedback mechanism, nor does the licensee's plant-specific submittal. The licensee replied that its existing general Corrective Action Program, along with its Maintenance Rule (10 CFR 50.65) program, would provide appropriate actions to correct future valve failures (see a more detailed discussion in Section 3.2.1, below).

Thus, considering the historically high reliability of the EFCVs and their low risk significance and radiological consequences should they fail, we find that the licensee's program for responding to future test failures is sufficient.

3.1.1.3 Technical Specification Level of Detail

The proposed TS states that "a representative sample" of EFCVs will be tested every 2 years. The "representative sample" is not defined in the TS itself. The proposed Bases say that the licensee will test 20% of the valves each refueling outage and, thus, test all of them in a 10-year period. We asked the licensee to justify placing the specific requirements in the Bases, rather than in the proposed TS.

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

The licensee also provided additional examples and explanations to demonstrate that its proposed TS is consistent with current STS practices.

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

3.2 Risk and Radiological Review

In Topical Report B21-00658-01, the licensee provided: (1) estimate of the steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close and (2) assessment of the radiological consequence of such release. Below is our review of this report as it pertains to DAEC as well as the licensee's subsequent response to an RAI dated October 8, 1999.

The instrument lines at DAEC include a 1/4-in flow restriction orifice upstream of the EFCVs to limit reactor water leakage in the event of a rupture. As discussed below, in Section 3.2.2, previous evaluation of such an instrument line rupture in DAEC UFSAR Section 1.8.11, for which the EFCVs are designed to mitigate, do not credit the isolation of the line by the EFCVs. Thus, a failure of an EFCV is bounded by the previous evaluation of an instrument line rupture. This analysis also showed that the resulting offsite doses would be well below regulatory limits.

The operational impact of an EFCV that is connected to the reactor pressure vessel (RPV) boundary failing to close is based on the environmental effects of a steam release in the vicinity of the instrument racks. The environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the released steam into the reactor building. However, the topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems and that the integrity and functional performance of secondary containment following instrument line break would be met. The separation of equipment in the reactor building is also expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. Nevertheless, the presence of an unisolated steam leak into the reactor building requires the licensee to shut down the reactor and depressurize to allow access to manually isolate the line.

The licensee's evaluation of the frequency of steam release caused by an instrument line break concurrent with an EFCV failing to close is reviewed in Section 3.2.1 of this report. The assessment of the radiological consequences of such release is reviewed in Section 3.2.2.

3.2.1 Estimation of Release Frequency

In estimating the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency and (2) the probability of EFCV failing to close. The licensee assumed an instrument line break frequency of 3.52E-05/year. Thus, for DAEC whose total number of instrument line/EFCVs is 94, the total plant instrument line break frequency was estimated at 3.31E-3/year. This estimate was based on the EPRI Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants," dated July 1992. This frequency corresponded to pipe sizes between ½ inch to 2 inches in diameter. The licensee considered these pipe sizes to represent the subject instrument line piping. The NRC staff considers this pipe break frequency to represent recent pipe failure data and the application of this frequency to represent the instrument line failure frequency to be acceptable.

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

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

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

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

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

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

The topical report determined an upper limit EFCV failure rate based on 11 observed failures in about 12424.5 years of service for 12 BWR plants in the U.S. (Note: 12424.5 years was determined by multiplying the number of tested EFCVs with the time period during which the number of occurring failures was reported). For 11 observed EFCV failures, the EFCV failure rate, λ, was estimated to be 1.57E-3.

However, the above formula for EFCV unavailability, $\bar{A}$, assumes that the EFCV failure rate is constant over time. The licensee, in its response to the NRC's RAI, reported that it was not currently aware of any study that explores the causes of EFCV failures, or changes in EFCV failure rate over time. Nevertheless, to account for the possibility that the failure rate for EFCV may change over time, potentially due to age-related factors, the licensee conservatively assumed the EFCV failure rate to change by five fold. In addition, the licensee assured that any EFCV failure would be documented in the DAEC Corrective Action Program as a surveillance test failure and that EFCV test failures would be trended to determine whether additional surveillance testing would be warranted. The licensee also indicated that its adherence to 10 CFR 60.65 Maintenance Rule Performance Criteria would ensure that EFCV performance remains consistent with the extended test interval and that the new performance criterion is less than or equal to one failure per year on a 3-year rolling average.

We consider the licensee's method to account for a potential change in EFCV failure rate to be acceptable. In addition, we consider the licensee's commitment to document and trend EFCV failure rates to determine if frequency of surveillance tests need to be increased to be both prudent and necessary.

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

$$\chi^2 = ½ (Z + (2n-1)^{½})^2$$

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

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

$$\chi^2 = ½ (1.645 + (2*112-1)^{½})^2 = 137.42$$

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

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

The release frequency was then calculated by the formula:

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

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

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

We consider the estimated increase in release frequency, 7.3E-5/year, to be sufficiently low. This is based on a qualitative analysis that an instrument line break with a concurrent failure of EFCV to close is not a significant contributor to core damage accidents. In addition, this increase in release frequency is lower than the DAEC large-break loss of coolant accident (LOCA) frequency of 3E-4/year that has the potential to lead to a core damage accident, whereas the instrument line break concurrent with EFCV failing to close does not. Based on these factors, we do not consider the estimated increase in release frequency to be significant.

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

3.2.2 Radiological Consequences

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

4.0 EVALUATION CONCLUSION

The impact of an increase in EFCV surveillance test intervals to 10 years along with an assumed five-fold increase in the EFCV failure rate on the likelihood of a release to the reactor building was shown to result in a release frequency of about 9.1E-5/year. This represents an increase of about 7.3E-5/year from the current release frequency estimate (for 2-year surveillance test interval) of about 1.8E-5/year. We consider this estimate to be sufficiently low, especially since the consequence of such an accident is unlikely to lead to core damage. We also agree with the licensee that the consequence of steam release from the depicted events is not significant as it was supported by a previous analysis. Based on the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with unlikely impact on core damage and negligible consequence of a release in the reactor building, we conclude that the increase in risk associated with the licensee's request for relaxation of EFCV surveillance testing to be sufficiently low and acceptable.

Therefore, we find the proposed TS change to be acceptable.

The NRC staff's initial proposed no significant hazards considerations determination was noticed in the *Federal Register* on July 14, 1999 (64 FR 38028). The October 5 and October 8, 1999, letters from the licensee provided clarifying information within the scope of the original application and did not change the NRC staff's initial proposed no significant hazards consideration determination.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (64 FR 38028). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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