March 28, 2000

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

SUBJECT: DUANE ARNOLD ENERGY CENTER - SAFETY EVALUATION OF RELIEF REQUEST FOR THE PUMP AND VALVE INSERVICE TESTING PROGRAM REGARDING EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS (TAC NO. MA6777)

Dear Mr. Protsch:

By letters dated October 4, 1999, and February 7, 2000, IES Utilities Inc. (IES) submitted a valve relief request (VR-24) for Duane Arnold Energy Center (DAEC). IES requested relief from performing American Society of Mechnical Engineers (ASME) Code required tests every refueling outage for each excess flow check valve (EFCV), and from the requirement to verify that the valve position is accurately indicated every 2 years.

The proposed alternative described in Relief Request VR-24 is identical to the technical specification (TS) amendment request for Surveillance Requirement 3.6.1.3.7 that was submitted by IES in a letter dated April 12, 1999, and approved in an Nuclear Regulatory Commission (NRC) safety evaluation (SE) dated December 29, 1999. The NRC staff SE concluded that the increase in risk associated with licensee’s request for relaxation of EFCV testing is sufficiently low and acceptable, and the alternative testing provides a high degree of valve reliability and availability. The NRC staff has reviewed the related Relief Request VR-24 submitted by the licensee in subsequent letters dated October 4, 1999, and February 7, 2000. This relief request will ensure that the TS requirement is consistent with ASME Code requirements. The NRC staff has concluded that pursuant to 10 CFR 50.55a(a)(3)(i), Relief Request VR-24 is authorized for use on the basis that the proposed alternative provides an acceptable level of quality or safety.

The NRC staff's SE is enclosed.

If you have any questions regarding this issue, please contact your Nuclear Reactor Regulation (NRR) Duane Arnold Project Manager, Brenda L. Mozafari at 301-415-2020.

Sincerely

/RA by T.J. Kim acting for/

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure: Safety Evaluation

cc w/encls: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PLAN

REQUEST FOR RELIEF

REGARDING EXCESS FLOW CHECK VALVE SURVEILLANCE REQUIREMENTS

DUANE ARNOLD ENERGY CENTER

IES UTILITIES, INC.

DOCKET NO. 50-331

1.0 INTRODUCTION

The *Code of Federal Regulations*, 10 CFR 50.55a, requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (the Code) and applicable addenda, except where alternatives have been authorized or relief has been requested by the licensee and granted by the Commission pursuant to Sections (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In proposing alternatives or requesting relief, the licensee must demonstrate that: (1) the proposed alternatives provide an acceptable level of quality and safety; (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety; or (3) conformance is impractical for its facility. Section 50.55a authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making the necessary findings. Guidance related to the development and implementation of IST programs is given in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," issued April 3, 1989, and its Supplement 1 issued April 4, 1995. Also see NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," and NUREG/CR-6396, "Examples, Clarifications, and Guidance on Preparing Requests for Relief from Pump and Valve Inservice Testing Requirements."

The 1989 Edition of the ASME Code, Section XI is the applicable Code of record for the third 10-year interval IST program at the Duane Arnold Energy Center (DAEC). Subsection IWV of the 1989 Edition, which gives the requirements for IST of valves, references Part 10 of the American National Standards Institute/ASME Operations and Maintenance Standards (OM-10) as the rules for IST of valves. OM-10 replaces specific requirements in previous editions of Section XI, Subsection IWV, of the ASME Code. Subsection IWP of the 1989 Edition, which gives the requirements for IST of pumps, references Part 6 of the American National Standards Institute/ASME Operations and Maintenance Standards (OM-6) as the rules for IST of pumps. OM-6 replaces specific requirements in previous editions of the ASME Code, Section XI, Subsection IWP.

By letters dated October 4, 1999, and February 7, 2000, IES Utilities Inc. (IES) submitted a valve relief request (VR-24) for DAEC. IES requested relief from performing ASME

Code-required tests every refueling outage for excess flow check valves, and from the requirement to verify every 2 years that the valve position is accurately indicated. The Nuclear Regulatory Commission (NRC) staff has reviewed the relief request and has provided the following safety evaluation (SE).

2.0 Valve Relief Request VR-24

VR-24 requests relief from Paragraph 4.3.2.2 of OM-10 which requires that check valves be exercised to the positions in which they perform their safety functions or examined at least once every reactor refueling outage. The licensee proposes to test the excess flow check valves (EFCVs) on a sampling basis every 24 months, such that each EFCV will be tested at least once every 10 years. Additionally, relief is requested from the requirements of valve position verification of Paragraph 4.1 of OM-10. The licensee proposes to verify the open position indication at a frequency more often than what the Code requires but verify the close position indication in conjunction with valve exercise tests.

2.1 Licensee’s Basis For Relief Request

The licensee states:

> The excess flow check valve is a simple device; the major components are a poppet and spring. The spring holds the poppet open under static conditions. The valve will close upon sufficient differential pressure across the poppet. Functional testing of the valve is accomplished by venting the instrument side of the tube. The resultant increase in flow imposes a differential pressure across the poppet, which compresses the spring and decreases flow through the valve.
>
> Excess flow check valves have been extremely reliable throughout the industry. In the first 25 years of operation at the DAEC, no excess flow check valve has failed to close due to actual valve failure (i.e., not related to test methodology). The DAEC Technical Specifications (TS) detail what frequency is required to maintain a high degree of reliability and availability, and provide an acceptable level of quality and safety. In the NRC's Safety Evaluation (dated December 29,1999) which approved the associated TS amendment, the Staff concluded, "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." The DAEC requests relief pursuant to 10 CFR 50.55a (a)(3)(i) to exercise excess flow check valves at the frequency specified in amended DAEC TS Surveillance Requirement (SR) 3.6.1.3.7.

2.2 Alternative Testing

The licensee states:

> EFCVs will be exercised at the frequency specified in Technical Specification Surveillance Requirement 3.6.1.3.7.
>
> The EFCVs have position indication in the control room. Check valve remote position indication is excluded from Regulatory Guide 1.97 as a required parameter for evaluating containment isolation. The remote position indication will be verified in the

closed direction at the same frequency as the exercise test, which will be performed at the frequency prescribed in Technical Specification Surveillance Requirement 3.6.1.3.7. After the close position test, the valves will be reset, and the remote open position indication will be verified. Although inadvertent actuation of an EFCV during operation is highly unlikely due to the spring-poppet design, the DAEC verifies the EFCVs indicate open in the control room at a frequency greater than once every 2 years.

2.3 Evaluation

An EFCV is provided in each instrument process line that penetrates the drywell and is part of the reactor coolant pressure boundary. The EFCV is designed so that it 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 has its status indicated in the control room. Because of the unique design, testing of these EFCVs and verifying their closure indication require a simulated instrument line break. With a larger number of EFCVs at DAEC, the Code-required test could result in a burden as well as significant costs for the licensee. Therefore, the licensee proposes to perform the exercise tests and valve position verification tests on a sampling basis, i.e., approximately equal number of EFCVs are tested every 24 months and each EFCV is tested at least once every 10 years.

The proposed alternative described in the relief request is identical to the technical specification (TS) amendment request for Surveillance Requirement (SR) 3.6.1.3.7 that was submitted by IES in a letter dated April 12, 1999. The NRC staff SE regarding the proposed amendment was issued on December 29, 1999, and concluded that the increase in risk associated with the licensee’s request for relaxation of EFCV testing is sufficiently low and acceptable. Additionally, an orifice is installed just inside the drywell on each of these instrument lines. The orifice limits leakage to a level where the integrity and functional performance of secondary containment and associated safety systems are maintained, and the coolant loss is within the capability of the reactor coolant makeup system. Therefore, the NRC staff finds that the proposed alternative provides an acceptable level of quality and safety and ensures valve reliability.

3.0 CONCLUSION

Pursuant to 10 CFR 50.55a(a)(3)(i), Relief Request VR-24 is authorized on the basis that the proposed alternative provides a high degree of valve reliability and availability and an acceptable level of quality and safety.

Principal Contributor: Y. S. Huang, DE/EMEB

Dated: March 28, 2000