April 13, 2004

Mr. Mark A. Peifer
Site Vice President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324-0351

SUBJECT: DUANE ARNOLD ENERGY CENTER - RE: REQUEST FOR AUTHORIZATION OF ALTERNATIVE REGARDING PRESSURE TEST REQUIREMENTS (TAC NO. MC2328)

Dear Mr. Peifer:

By letter dated March 12, 2004, Nuclear Management Company, LLC (NMC), requested to use an alternative to pressure test requirements specified in the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, at Duane Arnold Energy Center (DAEC). The applicable ASME Code for DAEC's Repair/Replacement Program is the 1992 Edition/1992 Addenda; the applicable ASME Code for the Inservice Inspection Program is the 1989 Edition/no Addenda. NMC's request involves a repair/replacement activity associated with main steamline safety relief valve (SRV), PSV 4401.

The Nuclear Regulatory Commission (NRC) staff has evaluated the above request and concludes that NMC's proposed alternative of performing the required VT-2 examination during a system leakage test of main steamline SRV PSV 4401 performed at a minimum pressure of approximately 940 psig instead of normal operating pressure (approximately 1025 psig) during the normal plant startup sequence provides reasonable assurance of the structural integrity of the SRV bolted connection. In reaching this determination, the NRC acknowledges your commitment that if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in March 2005), another inspection of this mechanical joint will be performed to look for any evidence of leakage. Furthermore, the NRC staff concludes that complying with the ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) on a one-time-basis during the normal plant startup from the current planned maintenance outage at DAEC to replace main steamline SRV PSV 4401.

Enclosed is our safety evaluation.

Sincerely,

/RA/

L. Raghavan, 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/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVE PRESSURE TEST FOR SAFETY RELIEF VALVE PSV 4401

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

The Nuclear Management Company, LLC's (NMC's) letter of March 12, 2004, requested Nuclear Regulatory Commission (NRC) authorization to use on a one-time-basis a proposed alternative to the pressure test requirements specified in the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," at Duane Arnold Energy Center (DAEC). NMC stated that it intends to conduct a planned shutdown at DAEC to replace main steamline safety relief valve PSV 4401, due to indications of leakage through the valve. Following the SRV replacement, the ASME Code requires that a system leakage test and VT-2 examination be performed to verify leak tightness of the mechanical joints (bolted connections). The test is required to be conducted at nominal operating pressure (approximately 1025 psig). In lieu of this ASME Code requirement, NMC requested NRC authorization to perform the required VT-2 examination during a system leakage test performed at a minimum pressure of approximately 940 psig, during the normal plant startup sequence.

2.0 REGULATORY EVALUATION

Inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the applicant demonstrates that (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements set forth in the ASME Code, Section XI, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval, and subsequent intervals, comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The code of record for the DAEC Inservice Inspection Program for the third 10-year interval is ASME Code, Section XI, 1989 Edition/No Addenda. The code of record for the DAEC Repair/Replacement Program for the third 10-year interval is ASME Section XI, 1992 Edition/1992 Addenda, which was approved for use at DAEC in NRC letter dated October 19, 1999.

3.0 TECHNICAL EVALUATION

3.1 Component for Which Relief Is Requested

Main Steam Safety Relief Valve (SRV) PSV 4401

3.2 ASME Code Requirements

ASME Code, Section XI, 1992 Edition/1992 Addenda, IWA-4710(c), states, "Mechanical joints made in the installation of pressure retaining replacements shall be pressure tested in accordance with IWA-5211(a)."

ASME Code, Section XI, 1989 Edition/No Addenda, IWA-5211(a), states, "The pressure retaining components within each system boundary shall be subject to system pressure tests under which conditions visual examination VT-2 is performed in accordance with IWA-5240 to detect leakages. The required system pressure tests and examinations, as referenced in Table IWA-5210-1, may be conducted in conjunction with one or more of the following system tests or operations: (a) a system leakage test conducted following opening and reclosing of a component in the system after pressurization to nominal operating pressure."

3.3 NMC's Proposed Alternative

NMC proposed to perform the system leakage test and VT-2 examination of the mechanical joints on SRV PSV 4401 during the normal operational start-up sequence at a minimum pressure of approximately 940 psig, in lieu of the nominal operating pressure associated with 100% reactor power (approximately 1025 psig). The VT-2 examination would be performed following the hold time required by the ASME Code. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in March 2005), another inspection of the bolted connection would be performed to look for any evidence of leakage. This alternative was proposed on a "one-time-only" basis following the repair/replacement of the SRV planned for April of 2004.

3.4 NMC's Basis for Relief (as stated):

> NMC considered the available methods to reach nominal operating pressure required to perform the system leakage test and VT-2 examination. These methods are discussed below.
>
> Pressurizing System Without Withdrawing Control Rods
>
> NMC cannot isolate PSV 4401 from the reactor vessel. Thus, NMC would have to manipulate numerous valves, change system lineups, and establish procedural controls

for heating and pressurizing the primary system in order to perform the system leakage test and VT-2 examination of the mechanical joints of the SRV without withdrawing control rods, while maintaining compliance with the technical specification pressure/temperature limits. The reactor pressure vessel would need to be filled with coolant and the steam lines flooded to the inboard main steam isolation valves to provide a water-solid condition. The pressure increase would be obtained by balancing the flow into the vessel, which is provided by the control rod drive system, with the flow out of the vessel provided by the reactor water cleanup system via the dump flow control valve and flow controller. This is the method used during refueling outages to complete the reactor pressure vessel system leakage test.

This test typically takes about two days to accomplish, and the additional valve lineups and system reconfigurations necessary to support this test impose an additional challenge to the affected systems. After completion of the test, system lineups must be restored to support startup.

In addition, the decay heat load at the time of the planned maintenance outage is expected to be significantly greater than the heat load that is typically present when this test is performed following a refueling outage; this would present additional operational challenges.

<u>Pressurizing System During Normal Startup</u>

Using normal startup procedures, the allowed pressure range for conducting the test would typically not be reached until a high power level (greater than 75% of rated). If access to the primary containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to current station ALARA practices. Establishing the 1025 psig test condition at a more moderate power level and in the manner needed to address radiation concerns would require a deviation from the method in which the primary system pressure control system (Electro-Hydraulic Control (EHC) Pressure Set) is normally used, as discussed below.

During a typical plant startup, after achieving criticality, the operating procedure directs the Operator to heat up and pressurize the reactor vessel (while maintaining the heat up rate within technical specification limits) by withdrawing additional control rods or raising EHC Pressure Set to maintain a turbine bypass valve within a specified "percent open" range. Adjustments to EHC Pressure Set are stopped, by procedure, when reactor pressure reaches 940 psig. The reactor power at that point is typically between 5 and 10% of rated.

While it is technically possible to manipulate these controls to establish the nominal system pressure of 1025 psig at lower power levels, doing so will affect core reactivity and could challenge plant safety systems, such as the reactor protection system. Changing the EHC settings outside of the normal range of operation for the purpose of performing this test at nominal operating pressure would pose an operational challenge, since this would be outside the normal operating parameters for startup. Procedural revisions would be required, as well as training provided to the Operators, to enable the EHC controls to be manipulated in a manner outside the norm.

NMC's Conclusion

Compliance with the Code-required system leakage test and inspection would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Application of this alternative test maintains reasonable levels of personnel safety and reduces the opportunity for the introduction of undesirable operational challenges. While NMC does not expect that leakage will occur, any leakage at the bolted connection would be related to the differential pressure across the connection. The reduction in test pressure is less than 10%, and is not, therefore, expected to affect the ability of the VT-2 examination to detect leakage from the bolted connection. In the event that leakage occurs at the mechanical joint at the slightly higher pressure associated with 100% operating power, it would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. Leakage monitoring is required by DAEC Technical Specifications. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in March 2005), another inspection of the bolted connection will be performed to look for any evidence of leakage.

The alternative will provide an acceptable verification of the integrity of the mechanical joint without unnecessary radiation exposure and operational challenges.

NMC requests NRC authorization of the aforementioned alternative on a "one-time-only" basis following the replacement of PSV 4401.

3.5 NRC Staff's Evaluation

The NRC staff finds that testing main steamline SRV PSV 4401 at nominal operating pressure, in order to meet the above ASME Code requirements, would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee stated that performing the required testing of this safety relief valve could only be accomplished with unusual difficulty in that extensive valve manipulations, alternative system lineups and procedural controls would be required for heating and pressurizing the primary system to establish the necessary test pressure, while complying with technical specification requirements for pressure-temperature limits, without withdrawal of control rods, i.e., without using nuclear heat. In addition, testing this SRV at nominal operating pressure would require an at-power containment entry which represents an imposition on personal safety due to excessive temperature and unnecessary radiation exposure.

The NRC staff further concludes that performing the test and accompanying VT-2 examination at a minimum of 940 psig during the normal plant startup following the SRV repair/replacement activities provides reasonable assurance of structural integrity. This is because this test pressure is adequate to cause the bolted connection to leak on the replacement SRV during the test if a leak-tight connection has not been established. Should leakage occur later, DAEC

Technical Specifications require NMC to monitor reactor coolant system leakage using the drywell sump system and the primary containment air sampling system. In addition, NMC committed in its letter of March 12, 2004, that if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in March 2005), another inspection of this mechanical joint will be performed to look for any evidence of leakage. According to NMC, the drywell monitoring systems would detect leakage if the mechanical joint were to leak at the higher pressures associated with nominal reactor power. These systems include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. The NRC staff agrees that monitoring such leakage provides additional assurance of the integrity of the component.

NMC stated in their letter of March 12, 2004, that the VT-2 examination would be performed following the hold time required by the ASME Code. In a telephone conversation on April 8, 2004, the licensee clarified that they planned to reinstall the insulation on SRV PSV-4401 prior to the VT-2 examination and that they were going to use the 4 hour hold time required by the ASME Code for insulated systems.

4.0 CONCLUSION

Based on the above, the NRC staff concludes that NMC's proposed alternative of performing the required VT-2 examination during a system leakage test of main steamline SRV PSV 4401 performed at a minimum pressure of approximately 940 psig during the normal plant startup sequence provides reasonable assurance of the structural integrity of the SRV bolted connection. Furthermore, the NRC staff concludes that complying with the ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, the NRC authorizes NMC's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) on a one-time-basis during the normal plant startup from the current planned maintenance outage at DAEC to replace main steamline SRV PSV 4401.

Principal Contributor: D. Beaulieu

Date: April 13, 2004