NMC
Committed to Nuclear Excellence

Duane Arnold Energy Center
Operated by Nuclear Management Company, LLC

March 12, 2004

NG-04-0176
10 CFR 50.55a(a)(3)(ii)

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Request for Authorization of Alternative Regarding Pressure Test Requirements

Nuclear Management Company, LLC (NMC) intends to conduct a planned shutdown at the Duane Arnold Energy Center (DAEC) to replace a safety relief valve (SRV). Increasing tailpipe temperatures have been observed on main steamline safety relief valve PSV 4401, indicating leakage through the valve. The replacement of the SRV is a repair/replacement activity associated with a mechanical joint in accordance with American Society of Mechanical Engineers (ASME) Section XI Code. The applicable Code for the DAEC Repair/Replacement Program is the 1992 Edition/1992 Addenda; the applicable Code for the Inservice Inspection Program is the 1989 Edition/no Addenda.

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). The enclosure contains a request for 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. As discussed in the request, as-low-as-reasonably-achievable (ALARA) considerations and other plant operational constraints make the performance of the examination and test at 1025 psig problematic, and constitute undue hardship without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(a)(3)(ii), authorization is requested for the enclosed alternative on a "one-time-only" basis. NMC requests authorization prior to April 12, 2004 to support planning for the shutdown. The NRC approved a similar request for the Monticello Nuclear Generating Plant on June 13, 2003 (One-Time Inservice Inspection Program Plan Relief Request No. 8 for Leak Testing the "B" and "G" Main Steam Safety Relief Valves, TAC No. MB9538, ADAMS Accession Number ML031640464).

A047

This letter makes the following commitment:

> 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.

Please contact Steve Catron of DAEC Regulatory Affairs at (319) 851-7234 with any questions regarding this submittal.

Mark A. Peifer
Site Vice-President, Duane Arnold Energy Center
Nuclear Management Company, LLC

cc: Regional Administrator, USNRC, Region III
Project Manager, DAEC (NRC-NRR)
NRC Resident Inspector (DAEC)

Enclosure

ENCLOSURE

Alternative to Examination at Nominal Operating Pressure

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(ii)

1. ASME Code Component(s) Affected

Code Class:	1
References:	Section XI, 1992 Edition/1992 Addenda, IWA-4710(c) Section XI, 1989 Edition/No Addenda, IWA-5211(a)
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	System pressure test and accompanying VT-2 examination at nominal operating pressure following repair/replacement activities involving mechanical joints
Component:	Main Steam Safety Relief Valve (SRV) PSV 4401

2. Applicable Code Edition and Addenda

ASME Section XI 1992 Edition/1992 Addenda is applicable to the DAEC Repair/Replacement Program for the Third Ten-Year Interval.

ASME Section XI 1989 Edition, no Addenda is applicable to the DAEC Inservice Inspection Program for the Third Ten-Year Interval.

3. Applicable Code Requirement

IWA-4710(c):
Mechanical joints made in the installation of pressure retaining replacements shall be pressure tested in accordance with IWA-5211(a).

IWA-5211:
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.

4. Reason for Request

A planned shutdown will be conducted at the DAEC to replace an SRV (PSV 4401). The SRV is connected to the main steam piping with a bolted, mechanical joint. Replacing the SRV for maintenance is considered a Repair-Replacement activity under the rules of ASME Section XI, 1992 Edition/1992 Addenda (the current code of record for DAEC Repair/Replacement Program). Following the replacement, a system leakage test and VT-2 examination are required. The system leakage test is required to be performed at the nominal pressure associated with the reactor at 100% power (approximately 1025 psig).

Several conditions associated with such testing represent an imposition on personnel safety, personnel radiation exposure, and challenges to the normal mode and manner of equipment operation. The SRV is not isolable from the reactor vessel; in order to perform this test, the primary system would need to be pressurized to the inboard system isolation valves. A leakage test and inspection at about 1025 psig cannot be performed during a normal plant startup, due to the excessive temperature and radiological exposure conditions to which licensee personnel would be exposed in the primary containment during the required VT-2 inspection. 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 the Technical Specification (TS) requirements for Pressure-Temperature (P/T) Limits, without withdrawal of control rods, i.e., without using nuclear heat.

5. Proposed Alternative and Basis for Use

Proposed Alternative

Pursuant to 10 CFR 50.55a(a)(3)(ii), NMC proposes 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 will 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 will be performed to look for any evidence of leakage. This alternative is proposed on a "one-time-only" basis following the repair/replacement of the SRV planned for April of 2004.

Basis for Use

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 TS P/T Limits. The reactor pressure vessel (RPV) would need to be filled with coolant and the steam lines flooded to the inboard main steam isolation valves (MSIVs) 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 (CRD) system, with the flow out of the vessel provided by the reactor water cleanup (RWCU) system via the dump flow control valve and flow controller. This is the method used during refueling outages to complete the RPV 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 start-up.

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.

Pressurizing System During Normal Start-Up

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 TS 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 (RPS). 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.

Conclusion

Compliance with the Code-required system leakage test and inspection would result in a 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 would occur 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.

6. Duration of Proposed Alternative

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

7. Precedent

The NRC authorized use of a similar alternative on a "one-time-only" basis for the Monticello Nuclear Generating Plant. (Letter dated June 13, 2003, One-Time Inservice Inspection Program Plan Relief Request No. 8 for Leak Testing the "B" and "G" Main Steam Safety Relief Valves, TAC No. MB9538, ADAMS Accession Number ML031640464).