May 28, 2003

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 - CORRECTION TO ISSUANCE OF AMENDMENT (TAC NO. MB4752)

Dear Mr. Peifer:

On March 21, 2003, the U. S. Nuclear Regulatory Commission issued Amendment No. 249 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. The amendment changed the surveillance requirement of Technical Specification 5.5.12, "Primary Containment Leakage Rate Testing Program," to allow a one-time 5-year extension to the 10-year interval for performing the next Type A containment integrated leakage rate test.

The amendment (page 2), including the technical specification replacement page 5.0-17 and certain pages of the associated safety evaluation report (pages 2, 3, 6, and 9), included typographical errors and inconsistencies with your application for amendment by letter dated March 29, 2002, as supplemented January 24, 2003. Please use the enclosed, corrected pages to replace the corresponding pages previously provided for Amendment No. 249.

Sincerely,

/RA/

Darl S. Hood, Senior Project Manager
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: As stated

cc w/encls: See next page

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 249, are hereby incorporated in the license. NMC shall operate the facility in accordance with the Technical Specifications. |

3. The license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 21, 2003

Corrected by NRC letter dated May 28, 2003

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

2. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

3. A required system redundant to support system(s) for the supported systems (1) and (2) above is also inoperable.

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

The first Type A test after the September 1993 Type A test shall be performed no later than September 2008.

(continued)

Corrected by NRC letter dated May 28, 2003

The regulation at 10 CFR Part 50, Appendix J, Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. DAEC TS 5.5.12 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163. RG 1.163, Section C, "Regulatory Position" states, "licensees intending to comply with the Option B in the amendment to Appendix J should establish test intervals based upon the criteria in Section 11.0 of NEI 94-01 [Nuclear Energy Institute report 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995], rather than using test intervals specified in ANSI/ANS-56.8-1994. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision in NEI 94-01 for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at DAEC has been successful, so the current interval requirement is 10 years.

Thus, the licensee is requesting an addition to TS 5.5.12 that would permit an exception from the guidelines of RG 1.163 regarding the Type A test interval by extending the currently specified 10-year interval to a 15-year interval on a one-time basis. Specifically, the proposed TS states that the first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008. The proposed change does not involve any change to a code, regulatory requirement, or acceptance criteria.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of the proposed license amendment as described in the licensee's application dated March 29, 2002, and the additional information provided in the licensee's letter dated January 24, 2003. The detailed evaluation below will support the conclusion 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

By Amendment No. 219, dated October 4, 1996, the NRC previously approved a revision to the DAEC TSs to incorporate the requirements of Option B for the Type A tests. The change proposed by the licensee's application dated March 29, 2002, involves only the extension of the interval between Type A containment leakage tests; Type B and C containment leakage tests will continue to be performed at the frequency currently required by DAEC's TSs.

3.1 Inservice Inspection for Primary Containment Integrity

The leak rate testing requirements of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by 10 CFR 50.55a complement each other in ensuring the leaktightness and structure integrity of the containment. Therefore, a detailed evaluation related to the inservice inspection of the containment and potential areas of weaknesses in the containment is performed in the following section.

DAEC utilizes a General Electric boiling water reactor (BWR) enclosed within a Mark I containment. The containment design includes a drywell and suppression chamber with interconnecting vent pipes and bellows, primary containment access penetrations, and other process piping and electrical penetrations. The drywell is a steel pressure vessel (0.75 to 3.0 inches thick) with a spherical lower portion and cylindrical upper portion. It is enclosed in 4- to 7-feet-thick, reinforced concrete for shielding, and provides additional resistance to deformation and buckling over areas where concrete backs up against the steel shell. Below the drywell head flange and above the foundation transition zone, the drywell is separated from the reinforced concrete by a gap of approximately 2 inches to allow for thermal expansion. Shielding over the top of the drywell is provided by removable, segmented, reinforced-concrete shield plugs.

The drywell is provided with a removable head to facilitate refueling, one combination double door personnel access lock/equipment lock, one equipment hatch, one personnel access hatch, and one control rod removal hatch. The head and hatches are all bolted in place and have double seals and provisions for leak tests.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus, located below and encircling the drywell. The suppression chamber will transmit seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside the chamber for inspection. Access to the chamber is provided by two 4-foot diameter manhole entrances with double gasket (leak testable) bolted covers connected to the chamber by 4-foot diameter steel pipe inserts.

Eight 4-foot, 9-inch diameter vent pipes connect the drywell and the pressure suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces or projectiles which might accompany a pipe break in the drywell. The vent pipes are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber. These bellows have test connections which allow for leak testing and for determining that the passages between the two-ply bellows are not obstructed.

The drywell vents are connected to a 3-foot, 6-inch diameter vent header in the form of a torus which is contained within the air space of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, 24 inches in diameter and terminating a minimum of 3 feet below the water surface of the pool and approximately 7 feet above the bottom of the Torus.

The last ILRT for DAEC was performed in September 1993. With the extension of the ILRT time interval, the next overall verification will be performed no later than September 2008. In its letter dated January 24, 2003, the licensee responded to the NRC staff's request for additional information for five issues regarding the ISI of the containment and discussed potential areas of weaknesses in the containment that may not be apparent in the risk assessment. The NRC staff's evaluation of the licensee's response to these issues is discussed below.

3.1.1 DAEC's ISI Program

The licensee states that ISI program is established in accordance with the requirements of the 1992 Edition with the 1992 Addenda of Subsections IWE of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The first 10-year containment inspection is divided into three periods as follows:

Regardless of the above schedule, any repairs or disassembly of a component with a seal, gasket, or bolted connection requires a post-maintenance Appendix J Type B test.

Thus, the licensee does not rely solely on the Type A testing for seals, gaskets, or bolted connections at DAEC.

3.1.4 Integrity of Stainless Steel Bellows

In the past, the NRC staff has found that two-ply stainless steel bellows are susceptible to trans-granular stress corrosion cracking, and the leakage through them is not detectable by Type B testing (see NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing"). The licensee states that in response to Information Notice 92-20, it evaluated and modified the test method used to measure leakage for DAEC's two-ply bellows to include provisions to detect potential damage to the bellows prior to determining leakage rate. The drywell-torus vent bellows are tested at a 120-month interval in accordance with the DAEC Performance Based Containment Testing Program and Regulatory Guide 1.163. The licensee states that there have been no local leak rate test (LLRT) failures of the drywell-torus vent bellows in the last ten refueling outages.

3.1.5 Inspection of Embedded Side of the Containment Steel Shell

The NRC staff is concerned that inspections of some reinforced concrete and steel containment structures at other nuclear power plants have identified degradation (e.g., corrosion) on the embedded side of the containment steel shell of the primary containment that cannot be inspected. The major areas of the Mark I containment that cannot be inspected are the vertical portion of the drywell shell and that part of the shell that is sandwiched between the drywell floor and basemat. A summary of work performed in the inaccessible region of the containment follows:

The licensee states that the inspections of the containment are performed during the time between ILRTs. The extension of time between ILRTs will not affect the inspections. The performance-based ILRT program guidance (NEI 94-01 and RG 1.163) requires a minimum of three visual examinations of accessible interior and exterior surfaces of the containment system to allow for early revealing of evidence of structural deterioration. The discrepancies identified in the liner, penetrations, and concrete are documented and dispositioned in accordance with the appropriate Code and design requirements. The details are as discussed above. The licensee states that approximately 85 percent of the containment surface area is accessible (i.e., approximate 15 percent of the area is inaccessible).

The licensee states that the potential leakage due to age-related degradation in areas that cannot be inspected is factored into DAEC's risk assessment which supports the requested ILRT interval extension from 10 to 15 years.

3.1.6 Maintaining Positive Pressure in The Containment

The licensee stated that during power operation the primary containment atmosphere is inerted with nitrogen to ensure that no external sources of oxygen are introduced into containment. The Containment Atmosphere Control System provides a supply of makeup nitrogen to is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in the ASME Code, Section XI, Subsections IWE/IWL). The most recent visual examination of the Duane Arnold containment was performed in 2001. The next scheduled IWE/IWL containment visual examination is 2003. Visual examinations are expected to be effective in detecting large flaws in the visible regions of the containment, and would reduce the impact of the extended test interval on LERF. The licensee performed additional risk analysis to consider the potential impact of corrosion in inaccessible areas of the containment shell on the proposed change. The risk analysis considered the likelihood of an age-adjusted flaw that would lead to a breach of the containment. The risk analysis also considered the likelihood that the flaw was not visually detected but could be detected by a Type A test. When possible corrosion of the containment surfaces is considered, the increase in LERF resulting from a change in the Type A test interval from the original 3 in 10 years to 1 in 15 years is estimated to be 3.8×10^{-8}/year. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The conditional containment failure probability increases about 1.3 percentage points for the cumulative change of going from a test interval of 3 in 10 years to 1 in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the change in the conditional containment failure probability for the proposed amendment.

On the basis of the above, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATIONS

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 or 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 the amendment involves no significant hazards consideration and there has been no public comment on such finding (67 FR 21291). Accordingly, the amendment meets the eligibility