April 30, 2001

Gary Van Middlesworth
Site Vice President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324-0351

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT RE: REVISED PRESSURE-TEMPERATURE CURVES (TAC NO. MB0394)

Dear Mr. Middlesworth:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 238 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 16, 2000, as supplemented December 22, 2000.

The amendment revises the TSs to incorporate new pressure and temperature (P-T) limit curves. The reactor pressure vessel P-T limit curves are updated for inservice leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality. The revised P-T limit curves are approved for an interim period not to exceed September 1, 2003.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

/RA by F. Lyon for/
Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 238 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

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*SE dated 1/16/01 **SE dated 3/21/01

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Duane Arnold Energy Center

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 238
License No. DPR-49

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC, dated October 16, 2000, as supplemented December 22, 2000, complies with the standards and requirements of the Atomic Energy Act of l954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with l0 CFR Part 5l of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 238, are hereby incorporated in the license. The licensee 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/
Claudia M. Craig, 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: April 30, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 238

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised areas are identified by amendment number and contains a marginal line indicating the areas of change.

Remove	Insert
3.4-24	3.4-24

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 238TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated October 16, 2000, Nuclear Management Company, LLC (NMC, or the licensee), submitted a license amendment request to update the pressure-temperature (P-T) limits for the Duane Arnold Energy Center (DAEC). The licensee proposed to revise the Technical Specifications (TSs) to provide new P-T limits that are valid to 25 and 32 effective full power years (EFPY). Composite curves were generated for each of the pressure test, core-not-critical and core-critical conditions at 32 EFPY. Separate P-T curves were developed for the upper vessel, beltline (at 25 and 32 EFPY), and bottom head for the pressure test and core-not-critical conditions. A composite P-T curve was also generated for the core-critical condition at 25 EFPY. The proposed changes also include appropriate changes to the TS Bases.

The Nuclear Regulatory Commission (NRC) staff had technical issues with the methodology used to derive the neutron fluence values used in the proposed licensing action. The methodology is the subject of General Electric topical report NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," which is currently under review by the staff. The fluence issues were discussed with the licensee in a teleconference on December 14, 2000. The staff concluded that these issues must be resolved in order to justify applying the fluence values for a full 32 EFPY. By letter dated December 22, 2000, NMC proposed that the staff grant interim approval of the P-T limits until September 1, 2003, to allow time for the staff to complete review of the methodology used by the licensee to determine neutron fluence. The December 22, 2000, letter 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

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2,

contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the review of P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

3.0 EVALUATION

3.1 Licensee Evaluation

The licensee requested, pursuant to 10 CFR 50.60(b), an exemption to use ASME Code Case N-640 as the basis for establishing the P-T limit curves. Code Case N-640 permits application of the lower bound static initiation fracture toughness value equation (K_{Ic} equation) as the basis for establishing the P-T curves in lieu of using the lower bound crack arrest fracture toughness

value equation (i.e., the K_{Ia} equation, which is based on conditions needed to arrest a dynamically propagating crack, and which is the method invoked by Appendix G to Section XI of the ASME Code).

The licensee submitted ART calculations and P-T limit curves valid for up to 25 and 32 EFPY. For the DAEC reactor vessel, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the lower intermediate shell plate 1-21 that was fabricated using plate heat number B0673-1. The ART values at the 1/4T location for 25 and 32 EFPY are 127.3 °F and 137.6 °F, respectively. The neutron fluence used in the ART calculation is 2.33 X 10^{18} n/cm^2 at the 1/4T location for 25 EFPY and 2.98 X 10^{18} n/cm^2 at the 1/4T location for 32 EFPY. The ΔRT_{NDT} values at the 1/4T locations for 25 and 32 EFPY are 100.3 °F and 110.6 °F, respectively. The initial RT_{NDT} for the limiting plate is 10 °F. The margin term used in calculating the ART for the limiting plate is 17 °F at 25 and 32 EFPY, as permitted by Position 2 of RG 1.99, Revision 2.

3.2 Staff Evaluation

As mentioned above, the licensee requested an exemption to use ASME Code Case N-640 as the basis for establishing the P-T limit curves. Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P-T operating limits curve is more technically correct than the K_{Ia} curve. The K_{Ic} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff concluded that P-T curves based on the K_{Ic} curve will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operation. In addition, implementation of the proposed P-T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety. Approval of the exemption was addressed in separate correspondence.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the DAEC reactor vessel is lower intermediate shell plate 1-21 that was fabricated using plate heat number B0673-1. The staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value.

The staff evaluated the licensee's P-T limit curves for acceptability by performing independent calculations, using the methodology referenced in the ASME Code (as indicated by SRP 5.3.2), and verified that the licensee's proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50. In addition, the staff independently generated P-T limit curves for normal operations and hydrostatic test pressures effective to 25 and 32 EFPYs for DAEC. By comparing the independently generated P-T curves with the licensee's curves, the staff determined that the licensee's proposed P-T limit curves meet the requirements of Appendix G of Section XI of the ASME Code, as modified by Code Case N-640. Therefore, the staff determined that the licensee's proposed P-T limit curves were acceptable for the interim period, since they meet the requirements of 10 CFR 50.60 and Appendix G of 10 CFR Part 50.

NMC's proposed revision to the P-T limits is necessary due to their proposed 20 percent power uprate to commence at the beginning of the next cycle, which is a separate licensee application and is being addressed by the staff in a separate safety evaluation. At the end of the current cycle, the plant will have accumulated 18.18 EFPY. The projected neutron fluence value was

estimated as the sum of the fluence at the end of the 18.18 EFPY and an estimate for the remaining 13.82 EFPY for which the plant will be operating at a power level 20 percent higher than the current full power level. The associated calculations for the P-T curves and the proposed fluence values were part of General Electric (GE) report GE-NE-A22-0010-08-01, "Pressure-Temperature Curves for Duane Arnold Energy Center" (non-proprietary version is publicly available). The extension requested by the licensee corresponds to the end of the current license. Staff review of the licensee's submittal and the associated GE report failed to establish the basis for the fluence used in the P-T curve evaluation. The licensee did not provide any information regarding the validity of the original estimated fluence. The second fluence value was estimated using a two-dimensional calculation, but the staff identified several shortcomings, such as a non-benchmarked code and unconventional cross sections. The staff related these concerns to the licensee and the vendor in a teleconference on December 14, 2000. In a letter dated December 22, 2000, the licensee proposed to limit the applicability of the P-T curves to the end of the next cycle, which is estimated to be completed on September 1, 2003. At that time exposure will not exceed 21 EFPY, which provides a margin of at least 4 EFPY to the least restrictive curve.

The proposed fluence values have a conservatism of about 43 percent because the calculated value was estimated for 32 EFPY. The second fluence component accounted for the 20 percent power uprate. DAEC will be operating with longer fuel cycles which entail low neutron leakage loadings, providing another source of conservatism. The staff concludes that there is reasonable assurance of safety for the interim application of the proposed P-T curves. Therefore, the fluence values are acceptable for the P-T curves until September 1, 2003, for DAEC.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 14 °F for DAEC, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation and inservice leak and hydrostatic testing.

The P-T curves for the non-beltline region were conservatively developed for a boiling-water reactor product line 6 (BWR/6) with nominal inside diameter of 251 inches. The analysis is considered appropriate for DAEC, since the plant specific geometric values are bounded by the generic analysis for large BWR/6. The generic value was adapted to the conditions at DAEC using plant specific RT_{NDT} values for the reactor pressure vessel. The application of the generic BWR/6 analysis to DAEC for the non-beltline region P-T curves is acceptable.

4.0 CONCLUSIONS

The staff concludes that the proposed P-T limits curves for each of the pressure test, core-not-critical and core-critical conditions; the separate P-T curves for the upper vessel, beltline, and bottom head; and the curve for the core-critical condition satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Case N-640, and Appendix G of 10 CFR Part 50. The proposed P-T limits also satisfy GL 88-11, because the

method in RG 1.99, Revision 2, was used to calculate the ART. Hence, the proposed P-T limit curves may be incorporated into the DAEC TSs on an interim basis until September 1, 2003. Also, the staff has no objections to the proposed change in the TS Bases. As discussed in the introduction of this safety evaluation report (SER), fluence issues must be resolved before the curves can be approved for a full 32 EFPY. When the staff's final SER on topical report NEDC-32983P is issued, the licensee can re-evaluate the fluence and submit revised P-T curves for approval through 32 EFPY.

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 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. The 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 (65 FR 77921). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Lee
L. Lois

Date: April 30, 2001