August 25, 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 - ISSUANCE OF AMENDMENT REGARDING PRESSURE AND TEMPERATURE LIMIT CURVES (TAC NO. MB8750)

Dear Mr. Peifer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 253 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 May 2, 2003, as supplemented June 30, July 30, and August 8 and 18, 2003.

The amendment updates the existing Reactor Coolant System pressure and temperature limit curves (TS Figure 3.4.9-1) and extends their applicability to 32 effective full-power years.

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

Sincerely,

/RA/

Darl S. Hood, Sr. 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. 253 to
License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page

August 25, 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 - ISSUANCE OF AMENDMENT REGARDING PRESSURE AND TEMPERATURE LIMIT CURVES (TAC NO. MB8750)

Dear Mr. Peifer:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 253 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 May 2, 2003, as supplemented June 30, July 30, and August 8 and 18, 2003.

The amendment updates the existing Reactor Coolant System pressure and temperature limit curves (TS Figure 3.4.9-1) and extends their applicability to 32 effective full-power years.

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

Sincerely,
/RA/
Darl S. Hood, Sr. 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. 253 to
License No. DPR-49
2. Safety Evaluation

DISTRIBUTION:

GHill (2)	DHood	MMitchell	WRuland
PUBLIC	GGrant, RIII	LLois	LRaghavan
PDIII-1 Reading	SMagruder	NRay	
ACRS	OGC	SCoffin	
THarris	GWilson, SRI	JUhle	

**See previous concurrence
*Provided SE input by memo

ADAMS Accession No. ML032310536

OFFICE	PM:PDIII-1	LA:PDIII-1	SRXB:SC**	EMCB:SC*	OGC	SC:PDIII-1
NAME	DHood	BClayton for THarris	JUhle	SCoffin	SCole	LRaghavan
DATE	08/20/03	08/20/03	08/20/03	08/06/03	08/21/03	08/25/03

OFFICIAL RECORD COPY

cc:

Mr. John Paul Cowan
Executive Vice President &
Chief Nuclear Officer
Nuclear Management Company, LLC
700 First Street
Hudson, MI 54016

John Bjorseth
Plant Manager
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324

Steven R. Catron
Manager, Regulatory Affairs
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, IA 52324

Regional Administrator
U. S. NRC, Region III
801 Warrenville Road
Lisle, IL 60532-4531

Jonathan Rogoff, Esquire
General Counsel
Nuclear Management Company, LLC
700 First Street
Hudson, WI 54016

Bruce Lacy
Nuclear Asset Manager
Alliant Energy/Interstate Power
and Light Company
3277 DAEC Road
Palo, IA 52324

Daniel McGhee
Utilities Division
Iowa Department of Commerce
Lucas Office Buildings, 5th floor
Des Moines, IA 50319

Chairman, Linn County
Board of Supervisors
930 1st Street SW
Cedar Rapids, IA 52404

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 253
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 (NMC, the licensee) dated May 2, 2003, as supplemented by letters dated June 30, July 30, and August 8 and 18, 2003, 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 10 CFR Part 51 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. 253 , 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 by September 1, 2003.

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: August 25, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 253

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 page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3.4-24	3.4-24

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 253 TO FACILITY OPERATING LICENSE NO. DPR-49

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By application dated May 2, 2003, as supplemented on June 30, July 30, and August 8 and 18, 2003, the Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Duane Arnold Energy Center (DAEC). The supplemental letters provided additional information that clarified the application and corrected calculation errors and the designation of certain information as proprietary. The supplemental letters did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 27, 2003 (68 FR 28855).

The proposed amendment would change the TSs by replacing the existing Reactor Coolant System pressure and temperature (P-T) limit curves for inservice leakage and hydrostatic testing, non-nuclear heatup and cooldown, and criticality (Figure 3.4.9-1, "Pressure Versus Minimum Temperature Valid to Thirty-two Full Power Years, per Appendix G of 10 CFR 50") with new, updated P-T limit curves. The replacement curves were generated using an NRC-approved methodology (General Electric Report NEDC-32983PA, Revision 1, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations," December 2001) for determining the neutron fluence on the Reactor Pressure Vessel (RPV) and extend the RPV beltline region to encompass a new limiting component, the recirculation inlet nozzle. The new curves would continue to be based upon the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-640. The change to Figure 3.4.9-1 would also delete the existing notation that states: "(Interim Approval Until September 1, 2003)."

2.0 REGULATORY EVALUATION

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 NRC staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"; Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"; GL 92-01, Revision 1 and Supplement 1 to Revision 1, “Reactor Vessel Structural Integrity”; Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”; and Standard Review Plan (SRP, NUREG-0800) Section 5.3.2, “Pressure Temperature Limits and Pressurized Thermal Shock.” In GL 88-11, the NRC staff advised licensees that it would use RG 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. In GL 92-01, Revision 1, the NRC staff requested that licensees submit RPV data for their plants. In Supplement 1 to GL 92-01, Revision 1, the NRC staff requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the NRC 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 ASME 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 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 methodology of Appendix G to Section XI of the ASME Code 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 depends upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor depends upon the neutron fluence at the maximum postulated flaw depth. The margin term depends upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Revision 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.

The regulatory requirements for RPV fluence calculations are specified in 10 CFR Part 50 Appendix A by General Design Criteria (GDC) 30, “Quality of Reactor Coolant Pressure Boundary,” and GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary.” In March 2001, the NRC staff issued RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron fluence.” The NRC staff has approved RPV neutron fluence calculational methodologies that satisfy the requirements of GDC 30 and 31 and adhere to the guidance in RG 1.190. Fluence calculations are acceptable if they are done with approved methodologies or with methods that are shown to conform to the guidance in RG 1.190.

3.0 TECHNICAL EVALUATION

3.1 Neutron Fluence Evaluation

The methodology that the licensee used for the RPV fluence calculation is described in GE Report NEDC-32983PA. This methodology follows the guidance in RG 1.190 and has been approved by the NRC staff by letter dated September 14, 2001. The methodology recognizes that DAEC began an extended power uprate at 18.18 effective full-power years (EFPYs) after the NRC staff issued Amendment 243, dated November 6, 2001, authorizing an increase in DAEC's maximum power level from 1658 megawatts thermal (MWt) to 1912 MWt (i.e., an increase of 15.3 percent, or 20 percent of DAEC's original rated power of 1593 MWt).

The neutron transport calculation was performed using the two-dimensional discrete ordinates neutral particle transport computer code DORT, "TORT-DORT Two-and Three-Dimensional Discrete Ordinates Transport Version 2.8.14," dated January 1994 and distributed by the Radiation Shielding Information Center at Oak Ridge National Laboratory, Computer Code Collection CCC-543. In this instance, the flux distribution at the vessel was estimated with two two-dimensional calculations in (r, θ) and (r, z) geometry. The scattering cross section, the quadrature approximations, and the azimuthal, radial, and axial meshes adhere to the guidance in RG 1.190. The cross sections were processed in the same manner as in the approved version of the methodology.

The value of the water density in each reactor cell was assumed to be the cycle-average value.

During its review, the NRC staff requested additional information regarding the effect of the power uprate on the axial location and size of the neutron leakage peak. In its letter dated July 30, 2003, the licensee responded with additional information that accounted for the relocation of the peak and the rearrangement of the power distribution. The NRC staff found the response to be acceptable.

The result of the licensee's RPV fluence calculation for use in the P-T limit curve evaluation for a 40-year plant life with an 80 percent operation capacity factor, is 4.17×10^{18} n/cm^2. On the basis of its review of the licensee's information and methodology, the NRC staff finds that the conservative value of the source, the approximations in the representation of the scattering and the quadrature, the number of mesh points, and the representation of the distribution of the water density meet the guidance in RG 1.190; therefore, the proposed peak vessel fluence of 4.17×10^{18} n/cm^2 at 32 EFPYs is acceptable.

3.2 P-T Limit Curves Evaluation

The licensee submitted ART calculations and P-T limit curves valid for up to 32 EFPYs. For DAEC's RPV, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the N2 Recirculation Inlet Nozzle (heat or heat Lot Q2Q6VW). The licensee's ART value at the 1/4T location for 32 EFPYs is 119.2°F. The neutron fluence used in the licensee's ART calculation is 0.585 X 10^{18} n/cm^2 at the 1/4T location for 32 EFPYs. The licensee's ΔRT_{NDT} value at the 1/4T location for 32 EFPYs is 45.2°F. The initial RT_{NDT} for the limiting nozzle is

40°F. The margin term that the licensee used in calculating the ART for the limiting material is 34°F as suggested by RG 1.99, Revision 2.

By letter dated April 27, 2001, and pursuant to 10 CFR 50.60(b), the NRC staff has previously issued an exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G, authorizing the licensee to use ASME Code Case N-640 as the basis for establishing DAEC's P-T limit curves. Accordingly, the licensee's proposed P-T limit curves for DAEC continue to use this code case. ASME 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). The K_{IA} equation is based on conditions needed to arrest a dynamically propagating crack, and is the method invoked by Appendix G to Section XI of the ASME Code. Using the K_{IC} curve in determining the lower bound fracture toughness in the development of P-T operating limits is more technically correct than using the K_{IA} curve.

The NRC staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. On the basis of these calculations, the NRC staff verified that the licensee's limiting material for DAEC's RPV is the Recirculation Nozzle N2 (Heat or Heat Lot Q2Q6VW). The NRC staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value.

The NRC 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 NRC staff independently generated P-T limit curves for normal operations and hydrostatic test pressures effective to 32 EFPYs for DAEC. By comparing the independently generated P-T curves with the licensee's curves, the NRC 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 ASME Code Case N-640. Therefore, the NRC staff finds that the licensee's proposed P-T limit curves are acceptable.

In addition to RPV beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based upon 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. The NRC staff has determined that the proposed P-T limits satisfy these requirements for the flange for normal and hydrostatic pressure and leak tests.

The licensee conservatively developed the P-T curves for the non-beltline region (upper vessel and bottom head) on the basis of a Boiling Water Reactor (BWR)/6 with nominal inside diameter of 251 inches. This is appropriate since DAEC's geometric values are bounded by the generic analysis for the large BWR/6. The licensee adapted the generic value to the conditions at DAEC using plant specific RT_{NDT} values for the RPV. On the basis of its review, the NRC staff finds the application of the generic BWR/6 analysis to the nonbeltline region P-T curves for DAEC to be acceptable.

The NRC staff concludes that the proposed P-T limit curves for each of the pressure tests, core not critical and core critical conditions; the separate P-T curves for the upper vessel, beltline, and bottom head satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by ASME 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. Accordingly, on the basis of the NRC staff's review of the information provided by the licensee and the NRC staff's independent calculations, the NRC staff finds the proposed P-T limit curves acceptable for use at DAEC for up to 32 EFPYs.

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. 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 (68 FR 28855). 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.

6.0 CONCLUSION

The NRC 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: M. Mitchell
N. Ray
L. Lois

Date: August 25, 2003