April 27, 2001

Mr. 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 - EXEMPTION FROM THE REQUIREMENTS OF 10 CFR PART 50, SECTION 50.60(a) AND APPENDIX G (TAC NO. MB0394)

Dear Mr. Middlesworth:

The Commission has approved the enclosed exemption from specific requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 50.60(a) and Appendix G, for the Duane Arnold Energy Center (DAEC). This action is in response to your letter of October 16, 2000, that submitted new pressure-temperature (P-T) limits for DAEC. The new P-T limits were developed using the methodologies in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for ASME Section XI, Division 1," which modify the methods of the ASME Code, Section XI, Appendix G.

Your letter of October 16, 2000, also included a request to amend your license to change certain Technical Specifications. That request is being handled as a separate action.

A copy of the exemption and the supporting safety evaluation is enclosed. The exemption has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/RA/

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. Exemption
2. Safety Evaluation

cc w/encls: See next page

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Site Vice President
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<u>Distribution w/encls:</u>

PUBLIC	BBurgess, RIII	LBerry	KWichman
PD III-1/Reading	ACRS	BMozafari	ALee
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SBajwa	THarris	FLyon	TBergman
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NAME	FLyon	THarris	KWichman*	RHoefling	CCraig	SBajwa
DATE	4/2/01	4/24/01	3/21/01	4/10/01	4/24/01	4/25/01

OFFICE	DLPM/D
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DATE	4/26/01

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Duane Arnold Energy Center

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7590-01-P

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

50-301

EXEMPTION

1.0 BACKGROUND

Nuclear Management Company, LLC (NMC, the licensee) is the holder of Facility Operating License No. DPR-49 which authorizes operation of the Duane Arnold Energy Center (DAEC). The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

The facility consists of a boiling water reactor located on NMC's DAEC site, which is located in Linn County, Iowa.

2.0 PURPOSE

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G requires that pressure-temperature (P-T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G states that, "The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions." Appendix G of 10 CFR Part 50 specifies that the P-T limits must meet the safety margin requirements specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G.

To address provisions of the proposed amendments to the technical specification (TS) P-T limits, the licensee requested in its submittal dated October 16, 2000, that the staff exempt DAEC from application of specific requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, and substitute use of ASME Code Case N-640. Code Case N-640 permits the use of an alternate reference fracture toughness (K_{Ic} fracture toughness curve instead of K_{Ia} fracture toughness curve) for reactor vessel materials in determining the P-T limits. The proposed action is in accordance with the licensee's application for exemption contained in the October 16, 2000, submittal, and is needed to support the TS amendment request that is contained in the same submittal. The proposed amendment will revise the P-T limits for heatup, cooldown, and inservice test limitations for the reactor coolant system (RCS) to 25 and 32 effective full power years (EFPYs).

Code Case N-640

The licensee has proposed an exemption to allow use of ASME Code Case N-640 in conjunction with ASME Section XI, 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to determine that the P-T limits meet the underlying intent of the Nuclear Regulatory Commission (NRC) regulations.

The proposed amendment to revise the P-T limits for DAEC relies in part on the requested exemption. These revised P-T limits have been developed using the K_{Ic} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the lower bound for fracture toughness. The other margins involved with the ASME Section XI, Appendix G process of determining P-T limit curves remain unchanged.

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 licensee has determined that the use of the initial conservatism of the K_{Ia} curve when the curve was codified in 1974 was justified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, 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 operations. The operating window through which the operator heats up and cools down the RCS is determined by the difference between the maximum allowable pressure determined by Appendix G of ASME Section XI, and the minimum required pressure for the reactor coolant pump seals adjusted for instrument uncertainties.

Since the RCS P-T operating window is defined by the P-T operating and test limit curves developed in accordance with the ASME Section XI, Appendix G procedure, continued operation of DAEC with these P-T curves without the relief provided by ASME Code Case N-640 may unnecessarily restrict the P-T operating window, especially at low temperature conditions. The operating window becomes more restrictive with continued reactor vessel service. Implementation of the proposed P-T curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety. Thus, pursuant to10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

In summary, the ASME Section XI, Appendix G procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G requirements by application of ASME Code Case N-640, while maintaining, pursuant to 10 CFR50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

3.0 DISCUSSION

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50, when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (2) when special circumstances are present. The staff accepts the licensee's determination that an exemption would be required to approve the use of Code Case N-640. The staff examined the licensee's rationale to support the exemption request and concurred that the use of the code case would also meet the underlying intent of these regulations. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR Part 50, Appendix G; Appendix G of the ASME Code; and regulatory guide (RG) 1.99, Revision 2, the staff concluded that application of the code case as described would provide an adequate margin of safety against brittle failure of the RPV. This is also consistent with the determination that the staff has reached for other licensees under similar conditions based on the same considerations. Therefore, the staff concludes that requesting the exemption under the special circumstances of 10 CFR 50.12(a)(2)(ii) is appropriate and that the methodology of Code Case N-640 may be used to revise the P-T limits for the DAEC RCS.

4.0 CONCLUSION

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Therefore, the Commission hereby grants NMC an exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, for the DAEC.

Pursuant to 10 CFR 51.32, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* (66 FR 20692). Accordingly, based upon the environmental assessment, the Commission has determined that the granting of this exemption will not result in any significant effect on the quality of the human environment.

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 27th day of April, 2001

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Cynthia A. Carpenter, Acting Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AN EXEMPTION FROM THE REQUIREMENTS OF

10 CFR PART 50, SECTION 60(a) AND APPENDIX G

NUCLEAR MANAGEMENT COMPANY, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

On October 16, 2000, the Nuclear Management Company, LLC (NMC, the licensee) submitted a license amendment request to update the pressure-temperature (P-T) limit curves for the Duane Arnold Energy Center (DAEC). In the October 16, 2000, submittal, NMC also requested Nuclear Regulatory Commission (NRC) approval for an exemption to use American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code) Case N-640, as a method that would allow NMC to deviate from complying with the requirements in 10 CFR 50.60(a) and Appendix G, for generating the P-T limit curves. The proposed changes to the P-T curves are based, in part, on the use of Code Case N-640, which was reviewed by the staff. Requests for such exemptions are allowed pursuant to 10 CFR 50.60(b), which allows licensees to use alternatives to the requirements of 10 CFR Part 50, Appendices G and H, if an exemption to use the alternatives is granted by the Commission pursuant to 10 CFR 50.12. According to 10 CFR 50.12, the Commission may, upon request, grant exemptions to the requirements of 10 CFR Part 50, if the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. In considering the exemptions, the Commission will not consider granting exemptions unless special circumstances are present. These special circumstances include, but are not limited to, the following special cases:

- Pursuant to 10 CFR 50.12(a)(2)(ii), the circumstance that application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule,

- Pursuant to 10 CFR 50.12(a)(2)(iii), the circumstance that compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated, and

- Pursuant to 10 CFR 50.12(a)(2)(vi), the circumstance that there is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption.

2.0 BACKGROUND

The NRC has established requirements in 10 CFR Part 50, Appendix G, to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The Appendix to Part 50 requires the P-T limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the Code (Appendix G to the Code) were applied. The methodology of Appendix G to the Code postulates the existence of a sharp surface flaw in the reactor pressure vessel (RPV) that is normal to the direction of the maximum applied stress. For materials in the beltline and upper and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. For the case of evaluating RPV nozzles, the surface flaw is postulated to propagate parallel to the axis of the nozzle's corner radius. The basic parameter in Appendix G to the Code for calculating P-T limit curves is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. The methodology requires that licensees determine the reference stress intensity (K_{Ia}) factors, which vary as a function of temperature, from the reactor coolant system (RCS) operating temperatures, and from the adjusted reference temperatures (ARTs) for the limiting materials in the RPV. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are initiated and grown from the inside and outside surfaces of the vessel, respectively. Regulatory Guide (RG) 1.99, Revision 2, provides an acceptable method of calculating ARTs for ferritic RPV materials; the methods of RG 1.99, Revision 2, include methods for adjusting the ARTs of materials in the beltline region of the RPV, where the effects of neutron irradiation may induce an increased level of embrittlement in the materials.

The methodology of Appendix G requires that P-T curves must satisfy a safety factor of 2.0 on primary membrane and bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the RCS. Table 1 to 10 CFR Part 50, Appendix G provides the staff's criteria for meeting the P-T limit requirements of Appendix G to the Code and 10 CFR Part 50, Appendix G.

3.0 EVALUATION

3.1 Exemption to Use Code Case N-640

NMC has requested, pursuant to 10 CFR 50.60(b), an exemption to use ASME Code Case N-640 (previously designated as Code Case N-626) 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 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). Use of the K_{Ic} equation in determining the lower bound fracture toughness in the development of the P-T operating limits curve is more technically correct than the use of the K_{Ia} equation since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{Ic} equation appropriately implements the use of the static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the initial conservatism of the K_{Ia} equation since 1974 when the equation was codified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ic} equation is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P-T curves based on the K_{1c} equation will enhance overall plant safety by opening the P-T operating window with the greatest safety benefit in the region of low temperature operations.

Generating the RCS P-T limit curves developed in accordance with Appendix G to the Code, without the relief provided by ASME Code Case N-640, would unnecessarily require the RPV to be maintained at a temperature exceeding 212 degrees Fahrenheit during the pressure test. Consequently, steam vapor hazards would continue to be one of the safety concerns for personnel conducting inspections in primary containment. Implementation of the proposed curves, as allowed by ASME Code Case N-640, does not significantly reduce the margin of safety and would eliminate steam vapor hazards by allowing inspections in primary containment to be conducted at a lower coolant temperature. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served. However, since use of the K_{Ic} equation results in the calculations of less conservative P-T limits than does use of the K_{Ia} equation, licensees need staff approval to apply the Code Case methods to the P-T limit calculations.

The ASME Code's Working Group on Operating Plant Criteria (WGOPC) has concluded that application of Code Case N-640 to plant P-T limits is still sufficient to ensure the structural integrity of RPVs during plant operations. The staff has concurred with ASME's determination and has previously granted exemptions to use Code Case N-640 for the Quad Cities Nuclear Power Station (i.e., in the NRC letter to Commonwealth Edison Company dated February 4, 2000). In the staff's letter of February 4, 2000, the staff concluded that application of Code Case N-640 would not significantly reduce the safety margins required by 10 CFR Part 50, Appendix G, and would eliminate steam vapor hazards by allowing inspections in the primary containment to be conducted at a lower coolant temperature. The staff also concluded that relaxation of the requirements of Appendix G to the Code by application of Code Case N-640 is acceptable and would maintain, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety for the Quad Cities RPVs and reactor coolant pressure boundary.

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. The staff, therefore, concludes that Code Case N-640 is acceptable for application to the DAEC P-T limits.

4.0 CONCLUSION

The staff has determined that NMC has provided sufficient technical bases for using the methods of Code Case N-640 in the calculation of the P-T limits for DAEC. The staff has also determined that application of Code Case N-640 to the P-T limit calculations will continue to serve the purpose in 10 CFR Part 50, Appendix G, for protecting the structural integrity of the DAEC RPV and reactor coolant pressure boundary. In this case, since strict compliance with requirements of 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, is not necessary to serve the overall intent of the regulations, the staff concludes that application of the Code Case N-640 to the P-T limit calculations meets the special circumstance provisions in 10 CFR 50.12(a)(2)(ii), for granting an exemption to the regulations and that, pursuant to 10 CFR 50.12(a)(1), the granting of an exemption is authorized by law, will not present undue risk to the public health and safety, and is consistent with the common defense and security. The staff, therefore, grants an exemption to 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to allow NMC to use Code Case N-640 as the part of the bases for generating the P-T limit curves for DAEC.

Principle Contributors: A. Lee
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Date: April 27, 2001