July 18, 2003

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NOS. MB7162 AND MB7163)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 278 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 10, 2002.

The amendment consists of changes to the Donald C. Cook Nuclear Plant (D. C. Cook) Unit 1 TSs related to the reactor pressure vessel (RPV) operating limits at low temperatures. The amendment approves revised pressure-temperature (P-T) limits for the RPV to be applicable for a maximum of 32 effective full-power years of facility operation. These changes were based, in part, on the use of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Case N-641.

In addition, your letter dated December 10, 2002, requested an exemption from the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, to allow application of ASME Code Case N-641. This Code case permits the use of an alternate reference fracture toughness for reactor vessel materials in determining the revised P-T curves, to maintain operator flexibility and safety during heatup and cooldown conditions. Based upon review of the information provided, the NRC staff has determined that application of ASME Code Case N-641 is acceptable for D. C. Cook Unit 1. Accordingly, the NRC staff, pursuant to 10 CFR 50.12(a), has issued an exemption from the requirements of Appendix G of 10 CFR Part 50 for D. C. Cook Unit 1.

A copy of our related Safety Evaluation supporting the amendment, and a copy of the exemption are also enclosed. The exemption and the Notice of Issuance for the amendment will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-315

Enclosures: 1. Amendment No. 278 to DPR-58
2. Safety Evaluation
3. Exemption

cc w/encls: See next page

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 278
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated December 10, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, 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-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 278, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to startup from Unit 1 refueling outage 19.

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: July 18, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 278

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 278 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By application dated December 10, 2002, the Indiana Michigan Power Company (the licensee or IMP) requested an amendment to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Unit 1. The proposed amendment would revise the Unit 1 reactor coolant system (RCS) pressure-temperature (P-T) curves in TS Figures 3.4-2 and 3.4-3 and associated TSs Bases.

The proposed changes to the P-T limits were based, in part, on the use of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) Case N-641. The licensee also requested an exemption from the requirements of Appendix G to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR Part 50), which mandates use of Appendix G to Section XI of the ASME Code for developing reactor pressure vessel (RPV) P-T limits, in order to utilize ASME Code Case N-641.

2.0 REGULATORY EVALUATION

2.1 10 CFR Part 50, Appendix G - Requirements for Generating P-T Limits for Light-Water Reactors

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in Appendix G of Part 50 to Title 10, *Code of Federal Regulations* (10 CFR Part 50, Appendix G), to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. 10 CFR Part 50, Appendix G, requires that the P-T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the ASME Code (henceforth Appendix G to Section XI) were used to generate the P-T limits. The methods of Appendix G to Section XI postulate the existence of a sharp surface flaw in the 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 thickness and a length equal to 1.5 times the thickness. For the case of evaluating an RPV nozzle, the surface flaw is postulated to propagate parallel to the axis of the nozzle's corner radius.

The basic parameter used in Appendix G to Section XI for calculating P-T limit curves is the stress intensity factor, K_I factor, which is a function of the stress state at the crack-tip and flaw configuration. The methods of Appendix G to Section XI require, in part, that licensees calculate the maximum allowable stress intensities (K_I factors) and pressures for the RPV as a function of temperature and based on use of the lower bound crack arrest fracture toughness equation (K_{Ia} equation) for the limiting adjusted reference temperature value (RT_{NDT} value) material 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, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, provides an acceptable method of calculating RT_{NDT} values for ferritic RPV materials. The methods of RG 1.99, Revision 2, include methods for adjusting the RT_{NDT} values 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 methods of Appendix G to Section XI also require that P-T curves must satisfy a safety factor of 2.0 on stress intensities arising from primary membrane stresses and primary bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on stress intensities arising from primary membrane stresses and primary 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 NRC staff's criteria for meeting the P-T limit requirements of Appendix G to Section XI and as well as the minimum temperature requirements of the rule for bolting up the vessel during normal and pressure testing operations.

2.2 Exemptions to the Requirements of 10 CFR Part 50, Appendix G

In the license's amendment request dated December 10, 2002, the licensee also requested NRC approval of an exemption to use Code Case N-641 as an alternative to the specific requirements in 10 CFR Part 50, Appendix G, for generating the P-T limit curves. Pursuant to 10 CFR 50.60(b), licensees may use alternatives to the requirements of 10 CFR Part 50, Appendix G, if an exemption to use the alternatives is granted by the Commission pursuant to 10 CFR 50.12. According to 10 CFR 50.12(a)(1), 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 a special circumstance is present. The special circumstances in 10 CFR 50.12 include, but are not limited to, the following special case:

- pursuant to 10 CFR 50.12(a)(2)(ii), the case 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.

3.0 TECHNICAL EVALUATION

The basic parameter used in Appendix G to Section XI for calculating P-T limit curves is the stress intensity factor, K_I factor, which is a function of the stress state at the crack-tip and flaw configuration. The methods of Appendix G to Section XI require, in part, that licensees calculate the maximum allowable stress intensities (K_I factors) and pressures for the RPV as a function of temperature and based on use of the lower bound crack arrest fracture toughness equation (K_{Ia} equation) for the limiting adjusted reference temperature value (RT_{NDT} value) material in the RPV. 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. RG 1.99, Revision 2, provides an acceptable method of calculating RT_{NDT} values for ferritic RPV materials. The methods of RG 1.99, Revision 2, include methods for adjusting the RT_{NDT} values 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 methods of Appendix G to Section XI also require that P-T curves satisfy a safety factor of 2.0 on stress intensities arising from primary membrane stresses and primary bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on stress intensities arising from primary membrane stresses and primary bending stresses when leak-rate or hydrostatic pressure tests are performed on the RCS.

The licensing basis for the P-T limit curves at D.C. Cook 1, as given in the TS limiting conditions of operation, includes two figures:

- TS Figure 3.4-2, which provides the P-T limit curves for normal operations of the reactor, including heatups at 60 °F/hr, operations with the core critical, and leak-rate/pressure test operations, and which includes requirements for minimum boltup temperatures.

- TS Figure 3.4-3, which provides the P-T limit curves for normal cooldown operations of the reactor at cooldown rates of 0 °F/hr (i.e., the steady state cooldown rate), 20 °F/hr, 40 °F/hr, 60 °F/hr, and 100 °F/hr.

In the amendment request of December 10, 2002, the licensee submitted the following updated P-T curves for NRC staff review and approval:

- new P-T limit curves (TS Figure 3.4-2) effective to 32 Effective Full-Power Years (EFPY) for normal operations of the reactor, including heatups at 60 °F/hr, operations with the core critical, and leak test operations, and including requirements for minimum boltup temperatures.

- new P-T limit curves (TS Figure 3.4-3) effective to 32 EFPY for normal cooldown operations of the reactor at cooldown rates of 0 °F/hr (i.e., the steady state cooldown rate), 20 °F/hr, 40 °F/hr, 60 °F/hr, and 100 °F/hr.

The new P-T limit curves in TS Figures 3.4-2 and 3.4-3 are based on use of ASME Code Case N-641. Section IV.A.2 of 10 CFR Part 50, Appendix G, requires the P-T limit curves to be at least as conservative as if the methods and criteria in Appendix G to Section XI were used to

generate the P-T limit curves.[1] Use of Code Case N-641 will generate P-T limit curves that are less conservative than would be generated if the methods of evaluation in Appendix G to Section XI were used to generate the P-T limit curves. Therefore, pursuant to the provisions in 10 CFR 50.60(b) and 10 CFR 50.12(a)(2)(ii), licensees must be granted exemptions against the requirements of 10 CFR Part 50, Appendix G, if they seek to apply the code case methods to the generation of their P-T limits. The NRC staff previously approved the methods in ASME Code Case N-641 for application to D. C. Cook, Unit 2 P-T limit curves by exemption dated March 20, 2003.

The NRC staff performed an independent assessment of the P-T limit curves submitted by the licensee for heatups and cooldowns of the reactor at heatup/cooldown rates of 60 °F/hr in order to verify compliance with requirements of Section IV.A.2 of 10 CFR Part 50, Appendix G, as exempted for use of the methods of evaluation in Code Case N-641. The NRC staff confirmed that use of 199 °F and 143 °F as the 1/4T and 3/4T RT_{NDT} input values for the generation of the P-T limit curves was valid for operations of the reactor effective to 32 EFPY and acceptable when evaluated against the recommended guidelines of RG 1.99, Revision 2. The NRC staff also confirmed that the proposed P-T limit curves generated by the licensee were at least as conservative as those that would be generated if the criteria and methods in the 1995 Edition of Appendix G to Section XI were used to generate the curves, as modified by the criteria and evaluation methods in ASME Code Case N-641. The NRC staff also confirmed that IMP's P-T limit curves included appropriate minimum temperature requirements that were at least as conservative as those required in Table 1 to 10 CFR Part 50, Appendix G. Based on this independent confirmation, the NRC staff concluded that IMP's P-T limit curves for normal and leak-rate test/pressure test operations are acceptable with respect to the requirements of 10 CFR Part 50, Appendix G, as exempted for permission to use of the ASME Code Case N-461 methodology for calculation of the P-T limits.

The fluence calculational methodology was described in WCAP-12483, an attachment to the license amendment request. The methodology used approximations, geometrical description, and cross sections in accordance with the guidance in RG 1.190. In addition, the neutron sources reflected the proposed power uprate from Cycle 18 on. The calculational results were compared to updated results for capsules T, X, and Y, which further validate the methodology. Surveillance capsules T, X, and Y were removed earlier than capsule U and before the cross sections in the BUGLE library were updated. The NRC staff review finds that the methodology adheres to the guidance of RG 1.190; therefore, the results of the fluence calculation are acceptable.

New Critical Element

The PT limit curve calculations are shown in the licensee's December 10, 2002, application. Recalculation of the adjusted reference temperature (ART) for all of the elements in the beltline region indicates that the intermediate to lower shell circumferential weld (heat 1P3571) has the highest ART (228 °F) at 32 EFPYs. The calculation was carried out using the fluence value of

(1) Acceptable editions of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code, as currently endorsed by reference in 10 CFR 50.55a, "Codes and Standards," are editions through the 1995 Edition of the Code, inclusive of the 1996 Addenda.

$1.8x10^{19}$ n/cm^2 which is the peak value at the inside surface of the vessel, thus, is acceptable. The PT limit curves, the low-temperature overpressure protection setting, and the RT_{PTS} to the end of the current license are determined by the properties of the intermediate to lower shell circumferential weld.

RT_{PTS} to 32 EFPYs

10 CFR 50.61 requires recalculation of the RT_{PTS} value whenever there are significant changes in the conditions of the beltline region. The planned power uprate and the change in critical element warrant recalculation of the RT_{PTS}.

The related analysis was performed by Westinghouse and is documented in WCAP-15789 which was included in the licensee's December 10, 2002, application. The 32 EFPY fluence was derived from WCAP-12483 and is an acceptable value. The evaluation of RT_{PTS} followed the guidance in RG 1.99. The estimated value is 242 °F. The relevant screening criterion in 10 CFR 50.61 is lower than 300 °F, therefore, the RT_{PTS} is acceptable.

4.0 SUMMARY

The NRC staff has determined that the proposed P-T limit curves are consistent with the alternate assessment criteria and methods of ASME Code Case N-641, and satisfy the requirements of 10 CFR 50.60(a)and(b), "Acceptance Criteria for Fracture Prevention Measures for Light-water Nuclear Power Reactors for Normal Operation;" Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements;" and Appendix G to Section XI (1995 Edition), as exempted by the methods of analyses in the code case. The proposed curves for D.C. Cook, Unit 1 are, therefore, approved for incorporation into the technical specifications.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or the use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents 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 15762). 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.

7.0 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 Contributor: J. Medoff
L. Lois

Date: July 18, 2003

7590-01-P

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-315

EXEMPTION

1.0 BACKGROUND

Indiana Michigan Power Company (the licensee) is the holder of Facility Operating License No. DPR-58 which authorizes operation of the Donald C. Cook (D. C. Cook) Nuclear Plant, Unit 1. The licensee provides, among other things, that the facility is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (NRC, the Commission) now or hereafter in effect.

The facility consists of a pressurized water reactor located in Stevensville, Michigan.

2.0 REQUEST/ACTION

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, Appendix G to 10 CFR Part 50 states that "[t]he appropriate requirements on...the pressure-temperature limits and minimum permissible temperature must be met for all conditions." Further, Appendix G of 10 CFR Part 50 specifies that the requirements for these limits are based on the application of evaluation procedures given in Appendix G to Section XI of the American Society of Mechanical

Engineers (ASME) *Boiler and Pressure Vessel Code* (Code). In this exemption, consistent with the current provisions of 10 CFR 50.55(a), all references to the ASME Code denote the 1995 Edition through the 1996 Addenda of the ASME Code.

In order to address provisions of amendments to the D. C. Cook, Unit 1, Technical Specification (TS) P-T limit curves, the licensee requested in its submittal dated December 10, 2002, that the NRC staff exempt D. C. Cook, Unit 1, from application of specific requirements of Appendix G to 10 CFR Part 50, and substitute the use of ASME Code Case N-641. ASME Code Case N-641 permits the use of an alternate reference fracture toughness curve for RPV materials and permits the postulation of a circumferentially-oriented flaw for the evaluation of circumferential RPV welds when determining the P-T limits. The proposed exemption request is consistent with, and is needed to support, the D. C. Cook, Unit 1, TS amendment that was contained in the same submittal. The proposed D. C. Cook, Unit 1, TS amendment will revise the P-T limits for heatup, cooldown, and inservice test limitations for the reactor coolant system (RCS) through 32 effective full power years of operation.

Code Case N-641

The licensee has proposed an exemption to allow the use of ASME Code Case N-641 in conjunction with Appendix G to ASME Section XI, 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to establish the P-T limits for the D. C. Cook, Unit 1 RPV.

The proposed TS amendment to revise the P-T limits for D. C. Cook, Unit 1, relies in part, on the requested exemption. These revised P-T limits have been developed using the lower bound K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A-2200-1, in lieu of the lower bound K_{IA} fracture toughness curve of ASME Section XI, Appendix G, Figure G-2210-1, as the basis fracture toughness curve for defining the D. C. Cook Unit 1 P-T limits. In addition, the revised P-T limits have been developed based on the use of a postulated circumferentially-oriented flaw for the evaluation of RPV circumferential welds in lieu of the

axially-oriented flaw which would be required by Appendix G to Section XI of the ASME Code. 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 as the basis fracture toughness curve for the development of P-T operating limits is more technically correct than use of the K_{IA} curve. The K_{IC} curve appropriately implements the use of a relationship based on static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a RPV, whereas the K_{IA} fracture toughness curve codified into Appendix G to Section XI of the ASME Code was developed from more conservative crack arrest and dynamic fracture toughness test data. The application of the K_{IA} fracture toughness curve was initially codified in Appendix G to Section XI of the ASME Code in 1974 to provide a conservative representation of RPV material fracture toughness. This initial conservatism was necessary due to the limited knowledge of RPV material behavior in 1974. However, additional information has been gained about RPV materials which demonstrates that the lower bound on fracture toughness provided by the K_{IA} fracture toughness curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure.

Likewise, the use of a postulated circumferentially-oriented flaw in lieu of an axially-oriented one for the evaluation of a circumferential RPV weld is more technically correct. The flaw size required to be postulated for P-T limit determination has a depth of one-quarter of the RPV wall thickness and a length six times the depth. Based on the direction of welding during the fabrication process, the only technically reasonable orientation for such a large flaw is for the plane of the flaw to be circumferentially-oriented (i.e., parallel to the direction of welding). Prior to the development of ASME Code Case N-641 (and the similar ASME Code Case N-588), the required postulation of an axially-oriented flaw for the evaluation of a circumferential RPV weld provided an additional, unnecessary level of conservatism to the overall evaluation.

In addition, P-T limit curves based on the K_{IC} fracture toughness curve and postulation of a circumferentially-oriented flaw for the evaluation of RPV circumferential welds 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. A narrow operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent overpressure protection system actuation due to pressure surges associated with normal plant evolutions such as RCS pump starts and swapping operating charging pumps with the RCS in a water-solid condition.

Since application of ASME Code Case N-641 provides appropriate procedures to establish maximum postulated defects and to evaluate those defects in the context of establishing RPV P-T limits, this application of the Code Case maintains an adequate margin of safety for protecting RPV materials from brittle failure. Therefore, the licensee concluded that these considerations were special circumstances pursuant to 10 CFR 50.12(a)(2)(ii), “[a]pplication 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.”

In summary, the ASME Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning reactor coolant pressure boundary materials and the estimated effects of operation. Since 1974, the level of knowledge about the fracture mechanics behavior of RCS materials has been greatly expanded, especially regarding the effects of radiation embrittlement and the understanding of fracture toughness properties under static and dynamic loading conditions. 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-641, while maintaining, 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 against brittle failure of the RPV.

The NRC staff has reviewed the exemption request submitted by the licensee and has concluded that an exemption should be granted to permit the licensee to utilize the provisions of ASME Code Case N-641 for the purpose of developing D. C. Cook, Unit 1, RPV P-T limit curves.

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.

Special circumstances, pursuant to 10 CFR 50.12(a)(2)(ii), are present in that continued operation of D. C. Cook, Unit 1, with the P-T curves developed in accordance with ASME Section XI, Appendix G, without the relief provided by ASME Code Case N-641 is not necessary to achieve the underlying purpose of Appendix G to 10 CFR Part 50. The underlying purpose of the regulations in Appendix G to 10 CFR Part 50 is to provide an acceptable margin of safety against brittle failure of the RCS during any condition of normal operation to which the pressure boundary may be subjected over its service lifetime. Application of ASME Code Case N-641 in lieu of the requirements of ASME Code Section XI, Appendix G provides an acceptable alternative methodology which will continue to meet the underlying purpose of Appendix G to 10 CFR Part 50.

The NRC staff examined the licensee's rationale to support the exemption request, and agrees within the licensee's determination that an exemption would be required to approve the

use of Code Case N-641. The NRC staff agrees that the use of ASME Code Case N-641 would meet the underlying intent of Appendix G to 10 CFR Part 50. The NRC staff concludes that the application of the technical provisions of ASME Code Case N-641 provides sufficient margin in the development of RPV P-T limit curves such that the underlying purpose of the regulations (Appendix G to 10 CFR Part 50) continue to be met so that the use of all provisions in Appendix G to Section XI of the ASME Code are not necessary. Therefore, the NRC staff concludes that the exemption requested by the licensee is justified based on the special circumstances of 10 CFR Part 50(a)(2)(ii), "[a]pplication 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."

Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of Appendix G to 10 CFR Part 50; Appendix G to Section XI of the ASME Code; and Regulatory Guide 1.99, Revision 2; the staff concludes that application of ASME Code Case N-641 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 NRC 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-641 may be used to revise the P-T limits for the D. C. Cook, Unit 1, RPV.

4.0 CONCLUSION

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are present. Therefore, the Commission hereby grants the licensee an exemption from the

requirements of 10 CFR 50.60 and 10 CFR Part 50, Appendix G, to allow application of ASME Code Case N-641 in establishing TS requirements for the reactor vessel pressure limits at low temperatures for D. C. Cook, Unit 1.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (68 FR 42137).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 18th day of July 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation