October 22, 2002

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, UNITS 1AND 2 - COMPLETION OF LICENSING ACTION FOR GENERIC LETTER (GL) 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS" (TAC NOS. M96801 AMD M96802)

Dear Mr. Bakken:

The Nuclear Regulatory Commission (NRC) staff issued Generic Letter (GL) 96-06 on September 30, 1996, to all holders of operating licenses for nuclear power reactors, except for those licenses that have been amended to possession-only status. GL 96-06 requested information from licensees related to two concerns: (1) water hammer and two-phase flow in the cooling water systems that serve the containment air coolers and (2) thermally-induced overpressurization of isolated water-filled piping sections in containment. On November 13, 1997, the staff issued Supplement 1 to GL 96-06 informing licensees about ongoing efforts and new developments associated with GL 96-06 and providing additional guidance for completing corrective actions. You responded to the GL and Supplement 1 by letters dated October 24, 1996, January 28 and May 20, 1997, August 15, September 8, and November 7, 2000, August 31, 2001, and June 28, 2002.

The NRC staff's review of your responses to GL 96-06, as documented in the enclosed safety evaluation, concludes that all requested information has been provided; therefore, we consider GL 96-06 to be closed for your facility.

Sincerely,

/RA/

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

Docket Nos. 50-315 and 50-316

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION OF THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RESPONSE TO GENERTIC LETTER (GL) 96-06

"ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING

DESIGN-BASIS ACCIDENT CONDITIONS"

DOCKET NOS. 50-315 AND 50-316

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) staff issued GL 96-06 on September 30, 1996, to all holders of operating licenses for nuclear power reactors, except for those licenses that have been amended to possession-only status. GL 96-06 requested information from licensees related to two concerns: (1) water hammer and two-phase flow in the cooling water systems that serve the containment air coolers and (2) thermally-induced overpressurization of isolated water-filled piping sections in containment. On November 13, 1997, the staff issued Supplement 1 to GL 96-06 informing licensees about ongoing efforts and new developments associated with GL 96-06 and providing additional guidance for completing corrective actions. You responded to the GL 96-06 and its Supplement by letters dated October 24, 1996, January 28 and May 20, 1997, August 15, September 8, and November 7, 2000, August 31, 2001, and June 28, 2002.

2.0 EVALUATION

2.1 Water Hammer and Two-Phase Flow

The containment structure at D. C. Cook is the Westinghouse ice condenser design. The design includes a lower compartment that contains the reactor vessel, steam generators and associated piping; an ice condenser; and an upper compartment. Following a high-energy line break in the lower compartment, steam and air will be forced into the ice condenser compartment through a series of doors where the steam will be condensed. The air will flow into the upper compartment. The containment cooling system at D.C. Cook does not perform a safety-related function, but serves to maintain the containment within acceptable limits for operation of equipment and personnel access during normal operation. There are four upper compartment coolers, four lower compartment coolers and two instrument room coolers within the containment building of each unit. Cooling water for the fan cooling coils is supplied by the non-safety-related non-essential service water (NESW) system. The containment isolation valves for the NESW system automatically close on high containment pressure, such as would occur following a large loss-of-coolant accident or main steamline break. The system would not isolate following smaller breaks for which the containment pressure would not reach the isolation setpoint. The safety function of the NESW is to maintain its integrity during design basis events so as to not provide a path for containment leakage. For this reason, the licensee evaluated water hammer within the system to ensure that piping failure will not occur.

The NESW system is an open loop system that uses water from Lake Michigan. After passing through the containment air coolers, the water is returned to the lake. All the containment air coolers are located at elevations in excess of 32 feet above the lake which is the maximum height that a column of water can be supported by atmospheric pressure. If offsite power were to be lost, all NESW pumps would lose power and coast down. Following loss of pumping power, the pressure within the fan coolers will decrease to the saturation pressure of the water within the fan cooler coils. That pressure will be very low. As the pressure drops to the saturation pressure, the water within the fan coolers will boil and release both steam and some of the air dissolved in the water. The NESW pumps would be loaded onto the diesel generators approximately 25 seconds after offsite power were lost. During the time when pumping power is lost, the containment coolers will drain thereby creating the potential for water hammer when power is restored. At the time just before power is restored, the licensee calculated that pressure within the NESW would drop to approximately 1.0 psia and the system would still be draining. Under these conditions, the upper compartment coolers and their piping would be mostly voided while the lower compartment coolers would still be full of water, although some voiding would occur in the higher regions of the piping attached to the lower coolers. There would also be some voiding in the loop seals of the supply line for the containment coolers outside the containment building.

Condensation induced water hammer (CIWH) has occurred when steam rapidly condenses on the surface of colder water in partially filled horizontal piping runs. Work by Griffith in Reference 1 indicates that pressure pulses from CIWH are not of significance for low pressure piping systems and should therefore not be of significance as the NESW system is draining following a loss of pumping power. In low pressure piping systems, the magnitude of pressure pulses from the starting of a pump in a partially voided system would exceed those from CIWH.

The licensee evaluated the consequences of a column closure water hammer that might occur following restart of the NESW pumps on emergency generator power using the SYSFLO computer code. The SYSFLO code was developed by MPR Associates to analyze water hammer events. The location for the maximum water hammer was calculated to occur in the main supply line outside the containment building producing a water hammer pulse of 200 psia. That pulse would not be transferred within the containment because of voiding within the system at that time. Smaller pressure pulses were calculated within the containment as the coolers and piping filled.

Although the maximum pressure pulse in the NESW supply header was outside the containment and would be attenuated by voiding, the licensee used this pulse to evaluate stresses within the NESW system assuming that the pulse was transmitted by a water filled system. The peak pipe stress was calculated to be 10.1 ksi.

To evaluate acceptable piping stresses, the criteria from Section III, Appendix F of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code were used. Since the NESW piping is not safety related, the acceptance criteria for the piping analysis was based on assuring that the piping will not catastrophically fail due to the hydraulic load with some yielding allowed. Using this methodology piping stresses were judged to be acceptable as long as the bending stress is less than 25 ksi. The licensee concluded that a considerable margin would exist to piping failure from the 10.1 ksi calculated piping stress.

The NRC staff has not reviewed the SYSFLO code that the licensee used to evaluate water hammer and asked that the licensee compare the SYSFLO code predictions with the Joukowski equation that is recommended in NUREG/CR-05220 (Ref. 2). The principal input variables in the Joukowski equation are the closure velocity during the collapse of a voided region in a pipe and the speed of sound in the water. The licensee developed a sample test problem and determined that for the sample problem the SYSFLO code results approximated those from the Joukowski equation for a speed of sound of 4000 feet per second. The speed of sound in unaerated cold water has a maximum value of approximately 4800 ft/sec. For actual piping systems for which the walls of the pipe can expand, a lower value (approximately 4500 ft/sec.) is appropriate. A sudden pressure decrease such as would occur during system draining following a loss of offsite power occurrence would cause air bubble formation. The effect of entrained air bubbles in water is to produce a dramatic reduction in the speed of sound (Ref. 3). Air bubbles have been found to quickly form within liquid water that experiences a sudden decrease in pressure (Ref. 4). The NRC staff believes a speed of sound of 4000 feet per second would be conservative following the depressurization that would occur following a loss of offsite power at D. C. Cook.

For further verification of the licensee's analysis, the staff performed calculations of maximum pressure pulses using flow rates calculated by the licensee in the water hammer analysis and maximum design flow rates from the Updated Final Safety Evaluation Report. The staff performed bounding calculations using the Joukowski equation to evaluate water hammer in the containment cooler supply pipe, the upper and lower containment coolers, and the instrument room coolers. The results by the staff were in some cases higher than those calculated by the licensee but were nevertheless well within the margin to piping failure.

The design of the NESW at D. C. Cook is that at any time when the pumps are stopped, voids will be formed within the system. The NRC staff requested that the licensee provide any experience with prior water hammers in the system. During startup testing of Unit 2 in 1978, an incorrectly installed expansion joint was found to be damaged as the result of an apparent water hammer. The condition was corrected. No other occurrence of water hammer damage has been recorded for the NESW.

In addition to water hammer, Generic Letter 96-06 is concerned with the occurrence of two phase flow conditions within containment air coolers that might affect the assumptions used for heat removal during design-basis accidents. The containment air coolers at D. C. Cook are not relied on to mitigate design-basis accidents, therefore, this aspect of the Generic Letter does not apply to the containment air coolers at D. C. Cook.

2.1.1 Summary Water Hammer and Two-Phase Flow

Based on the forgoing considerations, the NRC staff finds that the occurrence of a water hammer event such as postulated in Generic Letter 96-06 is highly unlikely at D. C. Cook

Nuclear Plant Units 1 and 2. Furthermore the staff finds that the licensee has provided the required evaluations and has adequately addressed the issues raised in Generic Letter 96-06 regarding the potential for water hammer and two-phase flow.

2.2 Thermally Induced Overpressurization

In the submittal of January 28, 1997, the licensee identified several pipe lines that were susceptible to thermally-induced pressurization at D. C. Cook Units 1 and 2. In the May 20, 1997, submittal, the licensee indicated that the evaluation of the affected pipe lines was complete and that the Updated Final safety Anaylsis Report allowable stress criteria for emergency conditions would not be exceeded. In submittals of August 15, 2000, and November 7, 2000, the licensee provided revised responses to GL-96-06 for D. C. Cook Units 1 and 2. The November 7, 2000, submittal indicated that a design change to install bypass check valves would be implemented for three lines in Unit 1. The submittal also identified 21 lines that are susceptible to thermally induced pressurization in Unit 1 that were analyzed using ASME Code Appendix F criteria. The August 15, 2000, submittal indicated that relief valves were added to seven lines in Unit 2 and that bypass check valves had been installed in three other lines. The submittal also identified four lines that are susceptible to thermally-induced pressurization in Unit 2. The licensee indicated that these lines had been analyzed and determined to be acceptable using ASME Code Section III, Appendix F criteria. By letter dated June 6, 2001, the NRC staff requested that the licensee provide additional information regarding the details of its analyses using the ASME Code Section III, Appendix F criteria.

In its submittal of August 31, 2001, the licensee indicated that the maximum pressure in 18 Unit 1 pipe runs would be limited by lifting of the diaphragm valves under overpressure conditions. The licensee provided a sample calculation to demonstrate the methodology used for the calculation. The licensee indicated that the calculated pressure required to open the valve 10 percent of its full stroke was used to evaluate the pipe run. The licensee indicated that the resulting calculation would bound any uncertainty in the calculated lift off pressure. The calculated pressure was used to calculate the stress in each pipe run. The licensee also described the methodology used to calculate the maximum temperature and pressure for the remaining 3 Unit 1 and four Unit 2 lines that did not contain diaphragm that lift to relieve the overpressure. The licensee indicated that the maximum water temperature was determined using either forced convection or condensation heat transfer on the outside surface of the pipe and natural convection heat transfer on the pipe's inner surface. The licensee provided sample calculations to demonstrate the methodology. The associated maximum calculated pressure was used to calculate the stress in each pipe run. The licensee provided the maximum calculated stress and strain for each pipe run in Tables 3 and 5 of its submittal for Units 1 and 2 respectively. The licensee's results indicate that the calculated membrane stress intensity is within the ASME Code Section III, Appendix F limit of 0.7 Su (ultimate stress) for all pipe runs. The staff considers the use of the ASME Code Section III, Appendix F stress limit reasonable and appropriate for this application.

2.2.1 Summary Thermally-Induced Overpressurization

The staff concludes that the licensee's corrective actions and evaluations provide an acceptable resolution for the issue of thermally-induced pressurization of piping runs penetrating containment. Furthermore, the staff finds that the licensee has provided the required evaluations and has adequately addressed the issues raised in GL-96-06 regarding thermally induced over pressurization of GL 96-06 for D. C. Cook, Units 1 and 2.

3.0 CONCLUSION

Based on the above evaluation, the staff concludes that all requested information has been provided; therefore, the staff considers GL 96-06 to be closed for your facility.

4.0 REFERENCES

1. P. Griffith, “Screening Reactor Steam/Water Piping Systems for Water Hammer,” NUREG/CR-6519, Massachusetts Institute of Technology, September 1997.

2. M. G. Izenson, P. H. Rothe and G. B. Wallis, “Diagnosis of Condensation-Induced Water hammer,” NUREG/CR-5220 Vol. 1, October 1988.

3. J. Paul Tullis, “Pumps, Valves, Cavitation, and Transients,” John Wiley & Sons, New York, 1989.

4. W. Zielke, H-D Perko and A. Keller, “Gas Release in Transient Pipe Flow,” Proc. 6th International Conference on Pressure Surges, BHRA, Cambridge, England October 4-6, 1989.

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Date: October 22, 2002