July 9, 2002

EA-01-286

Mr. A. C. Bakken III
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan MI 49107

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
            NRC SPECIAL INSPECTION REPORT 50-315/2001-17(DRP);
            50-316/01-17(DRP); PRELIMINARY YELLOW FINDING, JUNE 10, 2002

Dear Mr. Bakken:

This provides our response to your letter dated June 24, 2002, regarding the subject inspection
report and preliminary Yellow finding. You requested additional details from the NRC
concerning certain conclusions and assumptions referenced in the inspection report in order for
your staff to prepare for the regulatory conference. I understand that the regulatory conference
has been scheduled for July 25, 2002, and you have requested receipt of the additional details
by July 9, 2002.

My staff has completed a review of your request and developed the response enclosed with this
letter. With the exception of your request for the SPAR model and SAPPHIRE engine used in
our risk analysis, the additional details you requested are consistent with information provided
previously to your staff. My staff has discussed your request for the SPAR model and the
SAPPHIRE engine used in our risk analysis with the appropriate NRC Headquarter's staff.
Based upon these discussions, we will provide the generic SPAR model for the D.C. Cook site
under separate cover. In addition, appropriate NRC staff will be available at the regulatory
conference to discuss risk insights that we gained through our use of the SPAR model and
SAPPHIRE engine.

A. Bakken   -2-

If you have need of any additional details or have further questions, please contact
David Passehl, Acting Branch Chief, at 630-829-9872.

                                             Sincerely,

                                             /RA/

                                             Geoffrey. E. Grant, Director
                                             Division of Reactor Projects

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: NRC Response to Request for Additional Information

cc w/encl: J. Pollock, Site Vice President
                M. Finissi, Plant Manager
                R. Whale, Michigan Public Service Commission
                Michigan Department of Environmental Quality
                Emergency Management Division
                  MI Department of State Police
                D. Lochbaum, Union of Concerned Scientists

NRC Response to Request for Additional Information
Letter from A.C. Bakken, III, Indiana Michigan Power [DC Cook Plant]
dated June 24, 2002 (AEP:NRC:2609)

Reference: D. C. Cook Nuclear Power Plant, Units 1 and 2
                   NRC Special Inspection Report 50-315/2001-17(DRP); 50-316/01-17(DRP);
                   Preliminary Yellow Finding, June 10, 2002

RESPONSE

Request 1:   "A description of the Significance Determination Process results for any other
              sequences determined to be greater than Green by the Nuclear Regulatory
              Commission (NRC)."

Response 1:   As discussed with your staff during the inspection effort and at the exit meeting,
the NRC staff tentatively concluded that the risk associated with the finding was
dominated by a dual unit loss of offsite power (DLOOP) initiating event. For this
initiating event, the station blackout sequences were the dominate contributor to
the overall change in core damage frequency. The station blackout condition
was developed, in part, due to the presence of a failed essential service water
pump discharge strainer and your practice of cross-connecting the emergency
service water units and trains.

Our risk assessment of the DLOOP initiating event and related sequences was
based upon the conditions observed during the events of August 29, 2001, your
staff's representations regarding the reliability of plant equipment, the plant and
electrical switchyard equipment configuration, the plant's operating history, and
generic industry information.

While the inspection report did not discuss and we did not identify any other
greater-than-Green initiating events, scenarios, or sequences, please be advised
that the risk assessment results described in the subject inspection report were
preliminary. The results were based upon our understanding, as of the date the
inspection report was issued, of those factors which could impact the risk
assessment. Consistent with NRC policy, we will continue to re-evaluate our
preliminary findings, using relevant new or different information, until the
regulatory conference is held. Additional information which may affect the
preliminary risk assessment results would include the results of subsequent
inspections, recent plant-specific or industry events, and our discussions at the
regulatory conference.

Request 2:   "The basis for the assumption that the inrush of water expected to occur
              immediately after a dual unit LOOP event has sufficient energy and flow
              velocities to cause local eddies and vertical water velocities sufficient to entrain

debris located in the previous quiescent flow areas of the intake structure. Refer to page 29 of the inspection report."

Response 2: The NRC staff's assumption that an inrush of water into the intake structure would have sufficient energy and flow velocities to cause local eddies and vertical water velocities sufficient to entrain debris was based upon plant design and operating data; simplified calculations; and engineering judgement.

The NRC staff noted that the intake structure would experience an inrush of water following a DLOOP initiating event due to differences between the intake structure and lake water levels. The staff estimated that approximately 1.5 million gallons of water would be required to equalize the water levels. This inrush volume was based on an intake structure water level, prior to the DLOOP, of minus 12.6 feet, compared to the lake level, and an intake structure free area of approximately 200,000 square feet (204' x 100'). Approximately 20% of the intake structure volume was assumed occupied by structures and equipment.

Given plant design data, the intake tunnel water velocity was estimated to be approximately 8.3 feet per second, assuming the center intake was isolated and all seven circulating water pumps were running. Based upon an intake tunnel diameter of 16 feet and the calculated fluid velocity, the incoming water flow was determined to be in the fully turbulent flow regime. Turbulent flow is characterized by the generation of eddies which have a random velocity and the destruction of laminar flow lines. Additionally, the staff noted that a dissipation of energy within the intake structure water volume would occur due to frictional interaction, further enhancing the turbulent flow conditions and generating flow eddies.

Your staff's calculations regarding the intake structure indicated that, with the center intake closed, the normal velocity of the water passing through the traveling screens, at one foot above the intake structure floor, was approximately 5.0 ft/sec. Following a DLOOP, the circulating water pumps would stop. The flow through the circulating water pumps would rapidly slow and likely reverse due to the high frictional flow losses through the condenser and the sudden, extreme drop in pressure at the discharge of the circulating water pumps. However, water flow into the intake structure would initially be expected to continue along the same streamlines because of the momentum of the flow stream, the absence of barriers with high frictional flow losses, and the continued presence of a strong driving force (i.e., the low intake structure water level relative to lake level). The flow would then be redirected perpendicular to the initial flow direction by the loss of the outlet flow path through the circulating water pumps. Because of conservation of mass and momentum, the redirected flow will have a local velocity comparable (on the same order of magnitude) of the initial flow velocity just prior to the DLOOP. The energy associated with the redirected flow will then be dissipated by frictional forces, creating additional eddies and a potential to entrain debris. As demonstrated during the emergency service water degradation event of August 29, 2001, debris was typically located in quiescent areas adjacent to the normal flow streamlines such as at the base of

the traveling screens and in front of the emergency service water pumps and would be available for entrainment by these redirected flows.

The NRC staff assumed that the intake structure inrush would occur over an approximate 1 minute time frame. The one minute time frame was based on an initial circulating water flow of approximately 1.6 million gallons/minute (7 circulating water pumps operating at 230,000 gallons per minute per pump) and the volume of about 1.6 million gallons necessary to equalize intake structure water level with lake level. This volume of water would represent less than one-quarter of the total water volume contained in the 16 foot intake tunnels (based on a 2000 foot intake tunnel length with the center intake isolated). Due to the momentum of the intake tunnel flow, the inrush transient was expected to be a fairly dynamic event, resulting in an initial intake structure water level overshoot and a dampening oscillation until an equilibrium level was established.

Given the approximate 1 minute time assumed for the intake structure and lake water levels to equalize, the staff calculated that the intake structure bulk average vertical velocity would be approximately 0.18 feet/second immediately following a DLOOP. This level of bulk average vertical velocity was greater than the licensee calculated vertical velocity necessary to entrain and sustain sand particles in a fluid flow.

In addition to the bulk average vertical velocity of the water, the staff considered the potential for localized vertical velocities. Specifically, the staff considered changes in the intake structure water velocity that would initially occur in the vicinity of the circulating water pumps. These localized disturbances in the flow profiles were similar to profiles observed when a fluid traveling with a horizontal velocity enters an enclosure and is forced to change directions such as water entering a lock and dam structure or a large tank from a pipe. Based on the momentum of the incoming flow streams, the staff assumed that the initial disturbances in the flow patterns would be localized to the east side of the traveling screens. Perturbations in the water velocity profiles on the west side of the intake structure were not considered reasonable until after flow changes, originating near the circulating water pumps, following a stopping of the pumps, worked back to the west side of the intake structure. Consequently, because the volume east of the traveling screens represented approximately one third of the intake structure water volume, the staff concluded that the average bulk vertical velocity in the volume east of the traveling screens would be initially about three times the overall average bulk vertical velocity. Therefore, in addition to localized high velocity eddies, a bulk vertical velocity between the traveling screens and the east wall of the intake structure of up to 0.54 feet per second was considered possible. Licensee calculations indicated that a vertical velocities of 0.30 feet per second were sufficient to entrain and transport sand and shells.

Request 3: "The details from recent NRC studies indicating that the conditional probability of large early release, given core damage, is approximately 0.82. Refer to page 27 of the inspection report."

Response: The recent NRC studies, referenced in the inspection report and discussed with your staff during the inspection and several associated meetings, were documented in NUREG/CR-6427 [SAND99-2553], "Assessment of the DCH [Direct Containment Heating] Issue for Plants with Ice Condenser Containments," April 2000. Table 4.21, "Recommended DCH Containment Over pressure Failure Probabilities For Extrapolation Evaluations Assuming A DCH Event Occurs," on page 67, recommends a value of 0.82 for the conditional probability of a large early release at the D.C. Cook plant.

Request 4:   "The basis for using a large early release frequency value of 0.4, since the value appears to exceed the maximum conditional probability value provided in NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," dated January 1999. Refer to page 27 of the inspection report."

Response 4:   The staff utilized engineering judgement in the development of the large early release frequency value of 0.4 employed in our risk assessment. The value was developed using guidance provided in NUREG/CR 6427, as referenced above, and the staff's review of information supplied by your staff relative to their development of a similar factor using the guidance of NUREG/CR-6595. Your staff should be prepared to justify their basis for a large early release frequency value other than 0.4.

Request 5:   "The basis for and method used to combine the individual block evaluations into a "D/G common cause failure factor," including the final value reached. Also, please provide a description of how the SPAR model was modified to account for this factor. What failure modes were considered, a failure of individual emergency diesel generators (EDG) in any combination or failure of the 4 EDGs as a set? Refer to Page 27 of the inspection report."

Response 5:   The emergency diesel generator (D/G) common cause failure factor was developed based upon NRC staff review of information provided in your staff's analysis entitled, "Debris Intrusion Into the Essential Service Water System Probabilistic Evaluation," April 2002, and other related data. The common cause failure factor was used as a direct adjustment to the SPAR model core damage frequencies. Therefore, the SPAR model was not specifically modified to account for this factor. The common cause failure factor was based upon a common cause failure of all four D/Gs; therefore, individual or combinations of individual D/G failures were not addressed. The SPAR model outputs were focused on station blackout sequences; a condition that could only occur given a failure of all four D/Gs.

The D/G common cause failure factor included inputs related to Blocks 2, 3, 4, 7, and 8 of your staff's analysis. The NRC staff developed probabilities for these blocks; specifically, 0.5, 1.0, 0.77, 0.25, and 0.25, respectively. The D/G common cause failure factor for a DLOOP was 0.024, as discussed in the inspection report. The common cause failure factor did not include inputs for Blocks 1 and 9 of your staff's analysis due to these items having been previously accounted for in the SPAR model results. Your staff's assumptions, used to

4

develop information associated with Blocks 5 and 6, could not be confirmed or supported; therefore, the NRC staff did not include these items in development of the D/G common cause failure factor.

Request 6:   "The SPAR model and SAPPHIRE engine used to perform the risk analysis."

Response 6:   My staff has discussed your request for the SPAR model and the SAPPHIRE engine used in our risk analysis with the appropriate NRC Headquarter's staff. Based upon these discussions, we will provide the generic SPAR model for the DC Cook site under separate cover. In addition, appropriate staff will be available at the regulatory conference to discuss risk insights that we gained through our use of the SPAR model and SAPPHIRE engine.