October 3, 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
FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION REPORT NOS. 50-315/01-17(DRP); 50-316/01-17(DRP))

Dear Mr. Bakken:

The purpose of this letter is to provide you with the final results of our significance determination of the preliminary Yellow finding identified in the subject inspection report. The inspection finding was assessed using the significance determination process and was preliminarily characterized as Yellow (i.e., a finding with substantial importance to safety that will result in additional NRC inspection and potentially other NRC action). This preliminary Yellow finding involved a failed essential service water (ESW) strainer basket, caused by inadequate strainer basket installation instructions and maintenance practices, which permitted debris to bypass the strainer and enter the ESW system, resulting in the debris intrusion event experienced at the D.C. Cook Nuclear Power Plant on August 29, 2001.

At your request, a Regulatory Conference was held on July 25, 2002, to further discuss your views on this issue (see ADAMS Accession No. ML022320960). In addition, you provided a package of supplemental information in a letter dated July, 23, 2002. During the meeting, your staff described their assessment of the significance of the finding and detailed corrective actions taken. You agreed that the installation instructions and maintenance practices for the ESW pump discharge strainer basket that led to the event were unacceptable. Your staff performed several corrective actions, including inspecting all of the ESW pump discharge strainer baskets and revising the associated maintenance instructions.

During the Regulatory Conference, you concluded that the finding was of very low safety significance (Green) for the change in Core Damage Frequency (CDF) for both units. You also concluded that, relative to the Large Early Release Frequency (LERF), the finding was of very low safety significance (Green) for Unit 2 and of low to moderate safety significance (White) for Unit 1. In reaching this conclusion, you revised, consistent with current NRC guidance, the relative probability scale used to characterize inputs into your internal significance determination. Although you determined that the change in LERF for Unit 1 was White, you stated that if conservatism regarding charging system unit cross tie capabilities and availability of 69 kV offsite power were credited, the significance for CDF and LERF would be Green for both Units.

As discussed in NRC Inspection Report 50-315/01-17(DRP); 50-316/01-17(DRP), your staff and the NRC performed an evaluation of the plant's response following a loss-of-offsite power event and concluded that the dual unit loss-of-offsite power event was the dominant contributor to both CDF and LERF. The NRC performed a revised probabilistic evaluation using information you provided just prior to and during the Regulatory Conference. The NRC evaluation used the logical sequence of steps or "Blocks" described in the inspection report to define individual probability values to reach a final significance determination.

While the NRC determined that the results presented by your staff for most of the Blocks used in the analysis were reasonable, the NRC disagreed with your results for Block 2 (Suspended Debris Is Sufficient to Challenge ESW System), Blocks 5 and 6 (Debris Bypasses the 1E ESW Strainer), Block 9 (Unsuccessful in Restoring Flow (Human Error Probability)), and LERF. In performing its review, the NRC focused on the reasonableness of your assumptions and arguments that were presented at the Regulatory Conference. The results of the NRC's evaluation of your position regarding these specific Blocks used in the analysis are discussed in Enclosure 1.

The NRC concluded that the final result for the increase in CDF is estimated to be 2.4E-06 per year (White). The final result for the increase in LERF is estimated to be 9.7E-07 per year (White). After considering the information developed during the inspection and the information you provided at the conference, the NRC has concluded that the inspection finding is appropriately characterized as White, a finding of low to moderate safety significance.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC determined that the inadequate maintenance instructions for the installation of the ESW strainer baskets constitutes a violation of 10 CFR 50, Appendix B, Criterion V, as cited in the attached Notice of Violation (Notice) (Enclosure 2). The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved, is already adequately addressed on the docket in Inspection Report No. 50-315/01-17(DRP);50-316/01-17(DRP) and our letter to you summarizing the July 25, 2002, D.C. Cook Regulatory Conference, dated August 12, 2002. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

The characterization of this finding as White, in conjunction with the one existing White finding associated with the Unit 2 Turbine-Driven Auxiliary Feedwater Pump failure to start, has resulted in two White inputs to the Mitigating Systems Cornerstone in the Reactor Safety

Strategic Performance Area. Consequently, D.C. Cook performance is in the Degraded Cornerstone Column of the NRC Action Matrix, as described in NRC Manual Chapter 0305, “Operating Reactor Assessment Program.” We will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you by separate correspondence of that determination.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC’s document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

J. E. Dyer
Regional Administrator

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74
Enclosures: 1. Evaluation and Conclusion
2. Notice of Violation
cc w/encls: 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

DOCUMENT NAME: G:\eics\01-286 DCCook ESW.wpd *SEE PREVIOUS CONCURRENCE
To receive a copy of this document, indicate in the box: "C" = Copy w/o attachment/enclosure **"E"** = Copy with attachment/enclosure **"N"** = No copy

OFFICE	RIII		RIII		RIII		RIII	
NAME	*Lambert		*Vegel		*Reynolds for Grant		*Clayton	
DATE	9/27/02		9/27/02		9/30/02		9/27/02	
OFFICE	NRR		OE		RIII		RIII	
NAME	*Carpenter[1]		*Congel[2]		*Berson		Dyer	
DATE	09/26/02		9/26/02		09/30/02		10/3/02	

OFFICIAL RECORD COPY

[1] Via telephone call with T. Vegel

[2] Via e-mail from J. Dixon-Herrity

ADAMS Distribution:
WDR
DFT
JFS2
RidsNrrDipmIipb
GEG
HBC
KAC
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1
BJK1
FJC
JGL
JLD
LAD
OEMAIL
OEWEB
MDS1
RJS2
PLA
RWB1
MRJ1
MAS
JRJ
RML2
BAB2
WMD

EVALUATIONS AND CONCLUSION

On July 25, 2002, a Regulatory Conference was held to discuss the licensee's views on a preliminary Yellow finding involving a failed essential service water (ESW) strainer basket which permitted debris to bypass the strainer and enter the ESW system, resulting in the debris intrusion event experienced at the D. C. Cook Nuclear Power Plant on August 29, 2001. The strainer basket failure was caused by inadequate installation instructions and maintenance practices. The licensee acknowledged that the maintenance practices on the ESW pump discharge strainer basket were unacceptable, but provided additional insight on the significance determination for the finding.

Summary of Licensee's Input for the SDP

The licensee provided an evaluation which concluded that the finding was of very low safety significance (Green) for the change in Core Damage Frequency (CDF) for both units. Further, the licensee concluded that, relative to the Large Early Release Frequency (LERF), the safety significance of the finding was very low (Green) for Unit 2 and of low to moderate safety significance (White) for Unit 1. In reaching this conclusion, the licensee revised, consistent with current NRC guidance, the relative probability scale used to characterize inputs into its internal significance determination. Although the licensee determined that the change in LERF for Unit 1 was White, the licensee stated that if conservatism regarding charging system unit cross tie capabilities and availability of 69 kV offsite power were credited, the significance for CDF and LERF would be Green for both Units.

NRC Evaluation of Licensee's Input

Block 2 - Suspended Debris Is Sufficient to Challenge ESW System

The licensee assigned a probability value of 0.04 for this Block. This value was composed of two major components: (1) a probability that sufficient debris would be present within the intake structure prior to the event, and (2) a probability that a back flow of water from the main condensers, through the circulating pumps, would create a debris cloud in front of the ESW pump intake area. The licensee did not consider that the inrush of lake water into the ESW forebay, following a dual unit loss-of-offsite power (DLOOP), would be more than a minor contributor.

Prior to the Regulatory Conference, the NRC assigned a probability value of 0.5 for this Block. Based upon information provided by the licensee in their letter dated July 23, 2002, and during the Regulatory Conference, and independent NRC calculations, the NRC re-assigned a probability value of 0.09 for this Block. This value was composed of two components: (1) a probability that sufficient debris would be present within the intake structure prior to the event, and (2) a probability that a debris cloud would be developed as a result of the inrush of lake water into the intake structure. The NRC determined that the licensee's assumptions regarding the probability that sufficient debris would be present within the intake structure (Item 1 above) were reasonable; therefore, the NRC used the probability value of 0.3 in our final assessment. The NRC also determined that the development of a debris cloud in front of the ESW pumps was likely lower than previously assumed. Therefore, the NRC applied a lower probability value of 0.3 for the second component (Item 2 above). The NRC evaluation did not include a potential for a debris cloud to be developed due to backflow of water from the main condenser.

This was due to the NRC's assessment that the contribution of entrained material from the backflow of water out of the condensers was minor relative to the postulated entrained material from an inrush of lake water into the ESW forebay.

Blocks 5 and 6 - Debris Bypasses the 1E ESW Strainer

At the Regulatory Conference, the licensee did not provide any new information regarding differences between their and the NRC's handling of probabilities for these two Blocks. Previously, the licensee had assigned a probability of approximately 0.15 that high flows would be present and would permit ingested debris to bypass the 1E ESW strainer.

The NRC assigned a probability value of 1.0 to this Block both prior to and following the Regulatory Conference. During previous reviews of the licensee's technical evaluation of the August 2001 event, the NRC concluded that it was not reasonable to conclude that very high flows would be required for debris to bypass the 1E ESW strainer. Specifically, the NRC noted that the licensee's evaluation of debris transport past the 1E ESW strainer did not (1) address the impacts of flow changes that would be caused by debris retained in the 1E ESW strainer housing and (2) address the impacts of the gap between the strainer basket and the down-stream ESW piping. Therefore, the NRC concluded that material entering the damaged strainer basket would be transported past the basket and into the ESW system.

Block 8 - Cooling Water Flow Degradation Impacts Emergency Diesel Generator (D/G) Function

Initially, the licensee's review of the August 2001 event concluded that three of the four D/Gs had sufficient indicated flow to fulfill their intended safety function. Therefore, the licensee initially assumed that only one of the four D/Gs would be affected by debris ingestion. However, during the Regulatory Conference, the licensee indicated that the status of the Unit 2 D/Gs could not be determined with certainty, due in part, to the declining trend in their indicated flow values. As a result, the licensee modified the probability associated with Block 8 from 0.25 per D/G to 0.50 per D/G. In using this individual D/G probability, the licensee assumed an overall probability of 0.06 for all four of the D/Gs.

Prior to the Regulatory Conference, the NRC assigned a probablity value of 0.25, to this Block, for ESW debris to impact all four D/Gs. Based upon information provided prior to and during the conference, the NRC concurred with the licensee's position that the August 2001 ESW flow data was not conclusive evidence that the Unit 2 D/Gs would have been impacted by debris in the ESW system following a DLOOP. Given the preceding information and recognizing the inherent uncertainties associated with modeling the impact of decreased ESW flow on the D/Gs, the NRC concluded that assigning a probability value of 0.06 to this Block was not unreasonable.

Block 9 - Unsuccessful in Restoring Flow (Human Error Probability)

During the Regulatory Conference, the licensee indicated that the human error probability (HEP) value of 0.13 was developed on a per unit basis. Therefore, application of the analysis to all four D/Gs (two per unit) would result in an HEP of approximately 0.02. Following the Regulatory Conference, the NRC re-evaluated this issue and still considered the licensee's

HEP value of 0.13 per unit to be an optimistic approximation of the potential for the operators to recover one D/G from either unit. Based upon the above, the NRC proposed no change in the HEP value of 0.13, for both units, used in the risk assessment for the recovery of one of the four D/Gs. This value is consistent with previous risk assessment activities involving untrained, unproceduralized, stressful evolutions.

The NRC applied a HEP of 0.13 to the operator's non-ability to recover any one of the four D/Gs that may be affected by an ingestion of debris following a DLOOP. This value was approximately equal to the non-recovery value of 0.4 per unit (0.16 for both units) developed by the NRC using human reliability analysis techniques and Standardized Plant Analysis of Risk (SPAR) Accident Sequence Precursor Model worksheets. This value was developed assuming: (1) the time available for the operators to perform the necessary tasks to clear a clogged D/G heat exchanger was approximately equal to the time required; (2) the evolution was performed in a high stress environment and was of moderate complexity; (3) nominal training, ergonomics, fitness for duty, and work processes; and (4) credit for appropriate procedural guidance.

LERF

The licensee's LERF evaluation used the model described in Figure 2-2 of NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events." In this NUREG, the probability of early containment failure for cases where the hydrogen igniters fail, as in the case of a DLOOP with loss of the D/Gs, is 0.1 for low pressure sequences and 0.2 for high pressure sequences. The licensee's use of this approach relied primarily upon estimating the timing of release sequences and determining whether these sequences resulted in a release from containment before near-site evacuation was assumed to be completed. The evaluation estimated that near-site evacuation would be completed within a 3 to 4 hour time frame.

The licensee's evaluation also presented an example of a "fast" Station Blackout Event sequence as follows: (1) declaration of general emergency at 3 hours; (2) hot leg failure at 4.7 hours; and (3) vessel failure at 7.5 hours. The evaluation also assumed no containment failure prior to 7.5 hours, providing a time window of approximately 4.5 hours for evacuation. However, the NRC noted that hydrogen combustion and containment failure is just as likely after the hot leg failure time, since the generation and release of hydrogen to the containment is essentially complete by the time of hot leg failure (4.7 hours). Therefore, the assumed evacuation time could be as short as 1.7 hours. This makes evacuation of close-in population much more unlikely.

The NRC noted that the licensee's evacuation modeling did not account for the potential for any delay in the evacuation of close-in populations (some of the earlier consequence analyses use delay times of 1, 3, and 5 hours with various probabilities). Also, the evacuation model did not consider the impacts of the severe and extreme severe weather conditions assumed as a part of the DLOOP analyses on the evacuation time estimates. Both of these considerations could increase the probability of a core damage event resulting in a large early versus late release event. The NRC noted that a later study by Sandia, NUREG/CR-6427, "Assessment of the DCH [Direct Containment Heating] Issue for Plants with Ice Condenser Containments," which was not referenced or considered by the licensee, estimated early containment failure probabilities for all Westinghouse plants with ice condensers. For station blackout events, the study estimated a conditional early containment failure probability for D.C. Cook of 0.82. Prior to and following the Regulatory Conference, the NRC used a LERF value of 0.4 based upon engineering judgement. In developing the LERF value used in the NRC risk assessment, the NRC noted that NUREG/CR-6427 was a more recent report (2000) and was being used in the NRC's rule making activities in 10 CFR 50.44 (Standards for Combustible Gas Control System in LWRs [Light Water Reactors]) and in the resolution of Generic Safety Issue (GSI) -189 (Combustible Gas Control in Pressurized Water Reactor Ice Condensers and Boiling Water Reactor Mark IIIs). This study also included new information on containment response to scenarios involving a loss of the hydrogen igniters.

Considering the differences between the methods used to calculate the LERF using the two NUREGs and the potential for site specific factors to impact this value, the NRC used a value for LERF of 0.4. Although the Sandia study would suggest the use of a conditional early containment failure probability of 0.82, the study made several assumptions that would tend to increase the conditional failure probability, such as giving no credit for random ignition prior to vessel breach (which would reduce the conditional core damage probability). The use of the 0.4 value recognized that best-estimate failure probabilities would be somewhat less than the reported values in the Sandia study.

<u>Other Considerations - Charging System Unit Cross-Tie Capabilities and Availability of 69 kV Offsite Power</u>

During the Regulatory Conference, the licensee indicated that three other conservatisms existed which would further lower the overall risk assessment of a DLOOP: (1) Technical Specification-required charging system cross-tie was not credited; (2) Technical Specification-required 69 kV offsite power source was not considered; and (3) Most limiting LERF value of 0.2 was used versus potentially lower scenario-specific values.

The NRC evaluated the above three items and noted that these items would likely have very little impact on the NRC's risk assessment. Specifically, the charging system cross-tie function would not be available due to the loss of all site alternating current electrical power during DLOOP - station black out scenarios. In addition, the initiating events for the DLOOP were assumed to be severe weather and extreme severe weather. Consequently, the severe weather conditions would be expected to disable all offsite power supplies due to the proximity of the 69 kV, 345 kV, and 765 kV power sources. The NRC evaluated the licensee's method for determining the LERF value as discussed in the previous section of this letter.

<u>Conclusion</u>

The NRC determined that the final result for the increase in Core Damage Frequency (CDF) is estimated to be 2.4E-06 per year (White). The final result for the increase in Large Early Release Frequency (LERF) is estimated to be 9.7E-07 per year (White). After considering the information developed during the inspection and the information the licensee provided at the conference, the NRC has concluded that the inspection finding is appropriately characterized as White, a finding of low to moderate safety significance.

NOTICE OF VIOLATION

Indiana and Michigan Electric Company
D. C. Cook Nuclear Power Plant, Units 1 and 2

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74
EA-01-286

During an NRC inspection conducted on August 30, 2001, through May 17, 2002, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances.
>
> Contrary to the above, the documented instructions for installation of the essential service water (ESW) strainer baskets, an activity affecting quality, were not of a type appropriate to the circumstances. Specifically, the Unit 1 East ESW pump discharge strainer basket, was improperly installed on April 18, 1989, in accordance with Job Order 723483. The strainer basket installation instructions referenced by Job Order 723483 did not contain adequate detail associated with the verification of critical parameters affecting strainer basket alignment during installation. The failure to adequately align the ESW strainer basket within the strainer housing damaged the basket vertical support bracket and allowed debris greater than 1/8 inch in size to bypass the strainer. On August 29, 2001, this pre-existing failure of the strainer basket, combined with an ESW system alignment that created a common cause failure vulnerability, resulted in degraded ESW flow to all on-site emergency diesel generators.

This violation is associated with a White Significance Determination Process (SDP) finding.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence, and the date when full compliance was achieved, is already adequately addressed on the docket in Inspection Report No. 50-315/01-17(DRP);50-316/01-17(DRP) and our letter to you summarizing the July 25, 2002, D.C. Cook Regulatory Conference, dated August 12, 2002. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or your position. In that case, or if you choose to respond, clearly mark your response as a "Reply to a Notice of Violation," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice of Violation (Notice), within 30 days of the date of the letter transmitting this Notice.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

If you choose to respond, your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). Therefore, to the extent possible, the response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 3rd day of October 2002.