UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOCKETED
USNRC

'99 FEB 16 P4:36

OFFICE OF THE SECRETARY

February 11, 1999

Mr. David A. Lochbaum
Union of Concerned Scientists
1616 P Street, NW., Suite 310
Washington, D.C. 20036-1495

Dear Mr. Lochbaum:

I am responding to the Petition you submitted pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206) on October 9, 1997, and the additional information you provided in an Addendum to the Petition dated January 12, 1998. You submitted these two submittals, hereinafter referred to as the Petition and Addendum, on behalf of the Union of Concerned Scientists (UCS), the Petitioner. In accordance with 10 CFR 2.206, the Office of Nuclear Reactor Regulation was assigned to prepare a response to your request.

You requested that the U.S. Nuclear Regulatory Commission (NRC) modify, revoke, or suspend the operating licenses for Donald C. Cook Nuclear Plant, Units 1 and 2, (D. C. Cook) operated by Indiana Michigan Power Company (the Licensee) until there is reasonable assurance that the Licensee's systems are in conformance with design-basis and licensing-basis requirements. You also requested that a public hearing into this matter be held in the Washington, D.C. area before the first unit at D. C. Cook is authorized to restart. Finally, you requested that the NRC protect public health and safety by preventing the units at D. C. Cook from operating until such time as there is reasonable assurance that significant non-compliances have been identified and corrected. By letter dated December 9, 1997, the NRC acknowledged the receipt of your October 9, 1997, Petition.

By letter dated January 12, 1998, UCS sent an Addendum to the Petition. The Addendum discusses the 10 CFR 2.206 process and six specific concerns covering a broad range of issues. The specific concerns raised in the January 12, 1998, letter are: (1) ice condenser issues identified at Watts Bar Nuclear Power Plant and their applicability at D. C. Cook; (2) the Licensee's 10 CFR 50.59 safety evaluation process; (3) the scope of the Licensee's review of engineering calculations; (4) missing and inaccurate net positive suction head (NPSH) calculations; (5) the credibility of the Licensee's response to the confirmatory action letter (CAL); and (6) the NRC inspection process.

By letter dated February 23, 1998, the NRC acknowledged the receipt of your January 12, 1998, Addendum to the Petition. In the acknowledgment letter, you were informed that the issues raised in the Addendum as they relate to the Petition and D. C. Cook would be reported on in the Director's Decision. All other issues raised in the Addendum would be evaluated and transmitted to you in separate correspondence.

As the basis for these requests, you stated that the NRC completed an architect/engineering (AE) design inspection at D. C. Cook (NRC Inspection Report (IR) No. 50-315, 50-316/97201, November 26, 1997). NRC findings during the AE inspection led to the Licensee taking action to declare the emergency core cooling system (ECCS) inoperable at both units at D. C. Cook.

As a result, the Licensee shut down both units in accordance with their Technical Specifications (TS). As stated in the Petition, the systems reviewed during the AE inspection were the same systems that the Licensee had previously reviewed as part of its design-basis reconstitution program, and the program did not identify any deficiencies concerning system operability. Therefore, you asserted that the design-basis documentation reconstitution programs at D. C. Cook lacked the necessary rigor and focus to identify potential design-related operability issues. You further asserted that deficiencies in the Licensee's design-control programs may also be responsible for similar issues in safety systems that the NRC has not examined. On the basis of this potential, you also requested that the NRC increase the inspection scope at D. C. Cook.

For the reasons stated in the Director's Decision, the actions requested in the Petition and Addendum have been satisfied. The NRC has acted to ensure that public health and safety are protected and that a unit at D. C. Cook can only be restarted after the Licensee has demonstrated there is reasonable assurance the units will be operated in accordance with their license and within the provisions of the rules and regulations of the NRC. The specific request in the Petition to modify, revoke, or suspend the operating licenses for D. C. Cook, Units 1 and 2 is not necessary. In regard to the public hearing, the NRC granted the request for holding a public hearing before restarting a unit at D. C. Cook. On August 18, 1998, an informal public hearing was held at NRC headquarters offices in Rockville, Maryland, to discuss the issues raised in the Petition and the Addendum. The NRC is addressing the issues raised in the Addendum.

We appreciate your willingness to bring these concerns regarding nuclear safety to our attention. Public health and safety are served when individuals speak out and make the NRC staff aware of potential problems with licensee performance or the NRC inspection program. The issues you raised in the Petition and the Addendum paralleled and provided additional information regarding the issues being followed by the NRC at the D.C. Cook plant.

A copy of the Director's Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206(c). As provided for by this regulation, the Decision will constitute the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, institutes review of the Decision in that time.

I have enclosed a copy of the notice "Issuance of Director's Decision Under 10 CFR 2.206," which contains the complete text of DD-99-03 that is being filed with the Office of the Federal Register for publication.

Sincerely,

/s/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures: 1. Director's Decision DD-99-03
2. Federal Register Notice

cc w/encls: See next page

Indiana Michigan Power Company

Donald C. Cook Nuclear Plant
Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI 49127

Jeremy J. Euto, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Regulatory Affairs Director
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

John R. Sampson, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

DOCKETED
USNRC

DD-99-03

'99 FEB 16 P4:37

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

OFFICE OF SECRETARY
ACKNOWLEDGED

Samuel J. Collins, Director

In the Matter of	)	Docket Nos. 50-315
	)	50-316
	)	
INDIANA MICHIGAN POWER COMPANY	)	License Nos. DPR-58
	)	DPR-74
	)	
(Donald C. Cook Nuclear Plant, Units 1 and 2)	)	(10 CFR 2.206)
	)	

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. INTRODUCTION

On October 9, 1997, Mr. David A. Lochbaum submitted a Petition to the Executive Director for Operations of the U.S. Nuclear Regulatory Commission (NRC) pursuant to Section 2.206 of Title 10 of the Code of Federal Regulations (10 CFR 2.206). The Petition was submitted on behalf of the Union of Concerned Scientists (UCS or Petitioner) and requested that the operating licenses for the Donald C. Cook Nuclear Plant, Units 1 and 2 (D. C. Cook) be modified, revoked, or suspended to prevent operation of the units until there is reasonable assurance that significant non-compliances have been identified and corrected so that systems are in conformance with their design-basis and licensing-basis requirements. The Petitioner also requested that a public hearing into this matter be held in the Washington, D.C. area before the first unit at D. C. Cook is authorized to restart. The Petitioner indicated that the basis for his request was derived from a completed NRC architect/engineering[1] (AE) design

[1]NRC Inspection Report (IR) No. 50-315, 50-316/97201, November 26, 1997.

inspection at D. C. Cook. Findings by the NRC during the AE inspection led to the Licensee declaring the emergency core cooling system (ECCS) inoperable at both units at D. C. Cook. As a result, the Licensee shut down both units in accordance with their Technical Specifications (TS). As stated in the Petition, the systems reviewed during the AE inspection were the same systems that the Licensee had reviewed earlier as part of its design-basis documentation reconstitution program. This review did not identify any deficiencies concerning equipment operability. Therefore, the Petitioner asserted that the D. C. Cook design-basis documentation reconstitution programs lacked the necessary rigor and focus to identify potential design-related operability issues. The Petitioner further asserted that deficiencies in the Licensee's design control programs may also be responsible for similar issues in safety systems that have not been examined by the NRC. On the basis of this potential, the Petitioner also requested that the NRC increase the inspection scope at D. C. Cook.

On December 9, 1997, the NRC acknowledged receipt of the Petition and informed the Petitioner that the Petition had been assigned to the Office of Nuclear Reactor Regulation (NRR) to prepare a response and that action on the specific concerns raised in the Petition would be taken within a reasonable time.

By letter dated January 12, 1998, the Petitioner submitted an Addendum to the Petition. The Addendum raised additional issues concerning D. C. Cook and provided additional information concerning the Petition. In addition, the Addendum raised concerns dealing with the 10 CFR 2.206 process, the NRC inspection process, and generic concerns with ice condenser containments. On February 23, 1998, the NRC acknowledged receipt of the additional information and informed the Petitioner that the specific concerns related to the D. C. Cook plant and the Petition would be considered in the Director's Decision. Further, the NRC informed the Petitioner that the concerns not directly applicable to the requests in the Petition

would be evaluated and transmitted in separate correspondence. By letters dated July 10 and December 28, 1998, the NRC sent the Petitioner the status of the review of these issues not related to D. C. Cook or the Petition.

II. DISCUSSION

A. Request To Modify, Revoke, or Suspend the Operating Licenses for D. C. Cook Nuclear Plant, Units 1 and 2

The Petitioner based his request on the fact that the NRC had recently completed an AE design inspection at D. C. Cook and the inspection identified a number of issues concerning design and procedural controls, safety evaluations, use of engineering judgment, adequacy of operability determinations, temporary modifications, and consistency between the updated Final Safety Analysis Report (UFSAR) and the TS. The Petitioner asserted that the Licensee's design control programs were inadequate and there was the potential that similar issues could exist in other safety-related systems that the NRC had not inspected. The Petitioner requested that the units at D. C. Cook be prevented from operating until such time that there is reasonable assurance that significant non-compliances have been identified and corrected. The Petitioner stated in the Petition that the system certification process used at the Salem Nuclear Generating Station and the Millstone Nuclear Power Station would provide such reasonable assurance.

On September 8 and 9, 1997, the Licensee shut down both Unit 1 and Unit 2, respectively, because of inspection findings made by the NRC during the AE inspection. These findings led the Licensee to question the operability of the ECCS. Upon further investigation, the Licensee determined that the ECCSs in both units were inoperable and, in accordance with the TS, the Licensee shut down both units. By letter dated September 18, 1997, the Licensee identified several issues and corrective actions it would take preceding restart of either unit at

D. C. Cook. By letter dated September 19, 1997, the NRC issued a confirmatory action letter (CAL) confirming that nine specific issues from the Licensee's September 18, 1997, letter would be addressed by the Licensee before a unit at D. C. Cook would be restarted. In addition, the NRC recognized that the AE inspection was a limited-scope inspection and that the inspection findings were substantial. For this reason, the NRC confirmed that the Licensee, before restart of a unit at D. C. Cook, would perform an assessment to determine whether the type of inspection findings discovered during the AE inspection existed in other safety-related systems and whether they affected system operability.

By letters dated December 2, December 24, and December 31, 1997, the Licensee responded to the CAL. In these letters, the Licensee described the corrective actions, the root-cause analysis, and the reasons why the units at D. C. Cook were ready to restart. The NRC held public meetings with the Licensee on December 10 and December 22, 1997, and January 8, 1998, to discuss the Licensee's CAL responses.

The Petition raised concerns involving the Licensee's design control program and requested that a public hearing be held in the Washington, D.C. area before restarting either unit at D. C. Cook. The NRC staff reviewed the Petition thoroughly and determined that no new information was provided concerning D. C. Cook. The NRC staff came to this conclusion because the Petitioner based his concerns on the Licensee's design control program deficiencies that were identified in the NRC AE inspection. A CAL had been issued which confirmed that the Licensee would bound the problems discovered by the AE inspection and implement adequate corrective actions before restarting either unit at D. C. Cook. Therefore, following the guidelines contained in NRC Management Directive (MD) 8.11, "Review Process for 10 CFR 2.206 Petitions," the NRC staff came to the conclusion that new information was not provided and a hearing was not warranted.

In a telephone conversation on January 5, 1998, the NRC Petition Manager informed the Petitioner that new information was not provided in the Petition and, in accordance with MD 8.11, a public hearing would not be granted. By letter dated January 6, 1998, the Petitioner protested the NRC's decision not to hold a public hearing concerning the Petition. In that letter, the Petitioner stated that information concerning ice condenser issues was presented to the NRC Inspector General's Office and since D. C. Cook's containment operability relies on an ice condenser system this constituted new information. The Petitioner also stated that the Petition was developed and submitted in haste because NRC Region III officials indicated that the Licensee was planning to restart a unit at D. C. Cook in mid-October 1997 and the Petitioner wanted to submit the Petition before the first unit at D. C. Cook was restarted. For this reason, the Petition had not been fully developed and additional information would be forthcoming. On the basis of concerns that the Petitioner raised in the January 6, 1998, letter, and the assertion that the Petitioner potentially had new information, the NRC held a public meeting with the Petitioner on January 12, 1998. During the meeting, the Petitioner raised general concerns about the 10 CFR 2.206 process and addressed the following six specific concerns covering a broad range of issues:

(1) ice condenser concerns

(2) 10 CFR 50.59 Safety Evaluation process

(3) engineering calculations

(4) net positive suction head (NPSH) calculations

(5) licensee's response to the CAL

(6) NRC inspection process

By letter dated January 12, 1998, the Petitioner issued an Addendum to the Petition documenting the issues discussed during the January 12, 1998, public meeting. By letter dated February 23, 1998, the NRC acknowledged the receipt of the Addendum. Issues 1 through 5, as they relate to D. C. Cook and the Petition, are discussed individually in Sections II.B through II.F of this Director's Decision. As stated above, all issues raised in the Addendum not related to D. C. Cook or the Petition are being evaluated and will be addressed independent of the 10 CFR 2.206 process in separate correspondence.

The NRC staff reviewed the new information provided in the Addendum according to the guidelines of MD 8.11 and concluded that the additional information presented in the January 12, 1998, Addendum met the criteria for holding an informal public hearing. As a result, the NRC granted the Petitioner's request for an informal public hearing. On August 19, 1998, an informal public hearing was held at NRC headquarters in Rockville, Maryland. Both the Petitioner and the Licensee made presentations at the hearing. The hearing gave the Petitioner an opportunity to clarify the issues raised in the Petition and the Addendum. During the hearing, the Petitioner reported being pleased with the NRC oversight activities at D. C. Cook. Further, the Petitioner indicated he would like to see a Millstone scale civil penalty issued to the Licensee to ensure that the Licensee will maintain the proper safety culture in the future. During the hearing, the Petitioner also requested that the NRC investigate the potential that the Licensee's December 2, 1997, letter contained material false statements concerning the readiness of a unit at D. C. Cook to restart. This issue has been referred to the NRC Region III office for resolution and the results will be forwarded to the Petitioner under a separate cover.

In an effort to assess the effectiveness of the Licensee's corrective actions and the readiness of the units at D. C. Cook to restart, NRC performed an inspection of the CAL issues.

The results of the inspection are documented in NRC Inspection Report (IR) No. 50-315, 50-316/98004. The team of inspectors reviewed the nine specific issues identified in the CAL and considered them adequately addressed. The inspection team concluded that the short term assessment items were appropriate and bounded the AE inspection concerns. However, as described in the NRC July 30, 1998, letter to the Licensee, the CAL remains open pending the resolution of concerns involving the adequacy of the Licensee's assessment to determine whether the type of issues discovered during the AE inspection existed in other safety-related systems. By letter dated January 15, 1998, the Petitioner requested a copy of the inspection report, even if it was a preliminary version subject to revision, at least 1 business day before closing the CAL. In the NRC's February 23, 1998, letter, the request to release the draft inspection report was denied. As stated in the February 23, 1998, letter, it is not NRC policy to release draft predecisional information. This policy is intended to prevent improper influences and assure that predecisional information, or contemplated enforcement actions, are not compromised by a premature release. In accordance with MD 8.11, once the Petition was received, the Petitioner was placed on distribution for correspondence between the NRC and D. C. Cook. The Petitioner has subsequently received a copy of the IR.

The NRC expanded the scope of inspections of the D. C. Cook facility based on findings of the resident inspector staff, concerns that came to the NRC's attention regarding the ice condenser issues emanating from the AE inspection, and information brought to our attention by the Petitioner. This expanded scope of inspection satisfied the request in the Petition. From November 1997 until April 1998, the NRC performed inspections of the containment (IR No. 50-315, 50-316/97017), ice condenser (IR No. 50-315, 50-316/98005), hydrogen mitigation systems (IR No. 50-315, 50-316/98009), and the design-basis (IR No. 50-315, 50-316/98004). The inspections identified that NRC requirements had been violated. The apparent violations

were discussed at a public predecisional enforcement conference held at the NRC Region III office on May 20, 1998, with video viewing by the NRC headquarters staff, the Petitioner, and other members of the public in the NRC headquarters offices located in Rockville, Maryland.

During the predecisional enforcement conference, the Licensee admitted to all the apparent violations that formed the basis for the conference, described its assessment of the root causes, and presented its proposed corrective actions to address these issues. The Licensee stated that a root cause for many of these apparent violations was the failure to establish and communicate adequate performance standards.

As documented in the IRs, extensive degradation of the design of each unit's ECCS, ice condenser, refueling water storage tanks (RWSTs), and containment sumps, impaired the ability of the barriers (fuel cladding and containment) to prevent fission product release to the environment in the event of a design-basis loss-of-coolant accident (LOCA). With regard to the fuel cladding barrier, deficiencies were identified involving (1) a large quantity of fibrous materials within containment which would likely have clogged the ECCS sump screens in the recirculation mode, (2) a single-failure ECCS vulnerability, and (3) the insufficient amount of water available in the ECCS sump which represents a challenge to cool the fuel post LOCA. With regard to the containment barrier, the effects on the degraded ice condenser from blocked ice bed flow passages, missing ice segments, and ice basket damage represented a serious challenge to the ability of the ice condenser to perform its intended function to condense steam and suppress containment pressure. These conditions seriously impaired the safety function of the ECCS and the containment. Further, beyond the specific systems addressed by this enforcement action, two additional systems related to the containment, the hydrogen ignition and containment spray systems, were also degraded during the same period and, following analysis, the Licensee declared these systems inoperable.

During the informal public hearing, the Petitioner requested that the NRC issue a "Millstone" scale [2] civil penalty for the violations of NRC requirements at D. C. Cook. The violations were collectively categorized in accordance with the NRC Enforcement Policy (NUREG-1600) as a Severity Level II violation. This severity level was warranted for the breadth and number of the violations that, taken in total, resulted in a lack of reasonable assurance that following a design-basis accident, the ECCS and containment would have performed their intended functions.

On October 13, 1998, the NRC issued the Notice of Violation and associated proposed civil penalty to the Licensee. Accordingly, after considering the information obtained during the informal public hearing and predecisional enforcement conference, and after consultation with the Commission, the NRC staff chose to exercise discretion pursuant to Section VII.A.1 of the NRC Enforcement Policy and assessed a penalty in the amount of $500,000. Specifically, the escalated civil penalty reflected the consideration of the poor performance by the Licensee, the duration of the problems, the adverse impact on the ECCS and the containment, and the NRC's concerns regarding the violations. The purpose of the enforcement action was to emphasize the need for (1) taking timely and effective corrective actions for identified deficiencies, (2) effective surveillance testing and for plant personnel to challenge and investigate discrepancies identified during surveillance activities, (3) rigorous safety evaluations to determine whether changes to the plant or procedures constitute unreviewed safety questions, (4) maintaining the plant's design and licensing bases, and (5) a strong self-assessment program. The NRC staff would have proposed a larger civil penalty had it not been for the Licensee's decision to take

---

[2] On December 10, 1997, the NRC issued Enforcement Action EA 96-34 to Northeast Utilities which included Severity Level II violations and $2.1 million civil penalty.

comprehensive corrective actions and a commitment to keep the facility shut down until these problems are resolved.

Compliance with regulations, license conditions, and TS, and operation of a facility in accordance with the licensing basis is mandatory. However, the NRC also recognizes that plants will not operate trouble-free.[3] This is clearly articulated in Criterion XVI, Appendix B, Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Criterion XVI states that "measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." The appropriate response to an identified deficiency can and should vary, depending on the safety significance of the deficiency.

The conduct of NRC regulatory oversight at the D. C. Cook site is based on the recognition that it is the Licensee's responsibility to comply with its license and safety requirements and to take corrective actions when deficiencies are identified. Thus, the Licensee must determine that a unit is in conformance with applicable NRC regulations, its

---

[3] The NRC's regulations for protection of public health and safety embrace the philosophy of defense-in-depth, which supports the identification and correction of degraded or nonconforming conditions previously discussed. Briefly stated, this philosophy (1) requires the application of conservative codes and standards to establish substantial safety margins in the design of nuclear plants; (2) requires high quality in the design, construction, and operation of nuclear plants to reduce the likelihood of malfunctions, and promotes the use of automatic safety system actuation features; (3) recognizes that equipment can fail and operators can make mistakes, and therefore, requires redundancy in safety systems and components to reduce the chances that malfunctions or mistakes will lead to accidents that release fission products from the fuel; and (4) recognizes that, in spite of these precautions, serious fuel damage accidents can happen and, therefore, requires containment structures and safety features to mitigate the release of fission products. In the unlikely event of an offsite fission product release, emergency plans are in place to provide reasonable assurance that protective actions can and will be taken to protect the population around nuclear power plants. These emergency plans are coordinated with local and State officials and the Federal Emergency Management Agency.

license conditions, its UFSAR, and that applicable licensing commitments have been met before a unit is ready to restart. The Licensee's conformance with NRC regulations, license conditions, and licensing commitments is fundamental to the NRC's confidence in the safety of licensed activities. In short, the Licensee has the primary responsibility for the safe operation of its facilities.

By letter dated March 7, 1998, the Licensee docketed the D. C. Cook Nuclear Plant Restart Plan (Restart Plan). The Restart Plan is the principal program to provide reasonable assurance that weaknesses at the D. C. Cook units are identified and appropriate corrective actions are implemented. The Restart Plan includes efforts to understand and correct the licensing- and design-bases issues that, in part, led to the Licensee shutting down both units at D. C. Cook and the NRC taking escalated enforcement action. Revision 4 of the Restart Plan was submitted by the Licensee on December 16, 1998. The Licensee's Restart Plan included system readiness reviews for the most risk-significant systems at D. C. Cook. The reviews included evaluation of the UFSAR and TS design requirements, surveillance tests for the system, a review of design modifications, and a review of temporary modifications.

The NRC, in an effort to assess the effectiveness of the system readiness reviews, scheduled a safety system functional inspection (SSFI) on the auxiliary feedwater (AFW) system. The Licensee requested permission to conduct and subsequently conducted the SSFI, using independent contractors. The NRC provided oversight of the Licensee's SSFI through an inspection team. The NRC IR No. 50-315, 50-316/98017 associated with the oversight of the Licensee's SSFI was issued on January 28, 1999. In a public meeting on October 22, 1998, the Licensee presented the preliminary findings from the SSFI. The SSFI identified a number of issues, including findings that questioned the operability of the AFW system under certain accident conditions. These findings had not been identified by the Licensee's AFW system

readiness review. In a public meeting on December 22, 1998, the Licensee stated that enhancements would be made to the system readiness review process and a more thorough review of the most risk-significant systems would be performed before restart of a unit at D. C. Cook. These changes will be incorporated into the Licensee's Restart Plan.

Through the implementation of the Restart Plan, the Licensee has documented a large number of deficiencies that vary in scope and safety significance for each unit. The Licensee has identified deficiencies that must be corrected before restart. In its continuing review of the Licensee's corrective actions, the NRC will determine whether the Licensee has appropriately scheduled safety-significant items for completion before restart and whether the decision to defer selected corrective actions until after restart is appropriate for each unit. The results of these efforts will be documented in NRC IRs.

The NRC has developed a comprehensive and multifaceted oversight process to provide reasonable assurance that the Licensee has identified necessary issues and implemented required corrective actions. Because of the extent of issues discovered at D. C. Cook, the NRC has chosen to use the guidelines contained in NRC Inspection Manual Chapter (MC) 0350, "Staff Guidelines for Restart Approval" to conduct the oversight of the Licensee's corrective actions. MC 0350 establishes the guidelines for approving the restart of a nuclear power plant after a shutdown resulting from a significant event, a complex hardware problem, or serious management deficiencies. The primary objective of the guidelines in MC 0350 is to ensure that NRC's restart review efforts are appropriate for the individual circumstances, are reviewed and approved by the appropriate NRC management levels, and provide objective measures of restart readiness. In accordance with MC 0350, a restart panel has been established. Members include senior managers from both NRC Region III and the NRR offices. The NRR project manager and the senior resident inspector are also on the

panel. The panel meets internally to discuss restart issues on a weekly basis, and holds meetings approximately monthly with the Licensee to discuss the Licensee's corrective actions and schedules. The monthly meetings with the licensee are noticed and are open to the public.

By letters dated July 30 and October 13, 1998, the NRC forwarded to the Licensee the Case Specific Checklist for D. C. Cook in accordance with the MC 0350 guidelines. The checklist specified the activities the NRC considers necessary to be addressed before the restart of a unit at D. C. Cook. The items on the list were derived from the NRC's review of inspection activities, the CAL, and the Licensee's Restart Plan. As new issues emerge the Case Specific Checklist will be changed, and new issues necessary to be addressed before restart will be added to the list.

B. Ice Condenser Concerns

In the Addendum, the Petitioner identified problems in the configuration and testing of the ice condenser at the Watts Bar Nuclear Power Plant. The Addendum specifically identified problems with the inlet bay doors, floor upheaval, and ice basket components. The Addendum also stated that those problems were known, but were not properly reported by the Watts Bar Licensee (the Tennessee Valley Authority), the D. C. Cook Licensee (Indiana Michigan Power Company), the McGuire Licensee (Duke Power), and the vendor (Westinghouse). The Petitioner questioned if the Watts Bar ice condenser problems were valid and if they applied to the D. C. Cook facility. In the NRC's February 23, 1998, acknowledgment letter, the Petitioner was informed that the specific concerns regarding ice condenser issues at D. C. Cook would be addressed in the Director's Decision. All other issues concerning ice condensers at other facilities and the vendor will be reported on in separate correspondence. By letters dated July 10 and December 28, 1998, the Petitioner was informed of the review status of these issues.

As a result of concerns with the ice condensers at the D. C. Cook facility, the NRC Region III office initiated an inspection of the ice condensers. The Petitioner's concerns raised in the Addendum were incorporated into that inspection. In addition to the concerns raised in the Addendum, the inspection also reviewed activities associated with the surveillance test program of the ice condensers, the corrective actions performed on the ice condensers, and how the Licensee maintained the design-basis documentation concerning the ice condensers. The findings of the inspection were documented in NRC IR No. 50-315, 50-316/98005.

The inspectors determined that the overall material condition of the ice condensers was poor and some of the concerns raised by the Petitioner were confirmed. The issues raised in the Addendum concerning the inlet bay doors and the floor upheaval were not substantiated. The team inspected the doors of the ice condensers and found them to be functional but in poor material condition. In addition, the team identified deficiencies in the design-basis testing of the inlet bay doors. The team also inspected the ice condenser floor sections, which have the potential to heave and prevent the bay doors from operating properly. No signs of floor upheaval or degradation were detected. Concerning the issue of deficiencies of ice basket components, the team identified defective and damaged ice baskets. Examples include the following: (1) dented and buckled ice basket webbing, (2) missing sheet metal screws used to couple the ice baskets together, (3) loose and missing U-bolt nuts on lower ice basket assemblies, (4) separated ice baskets, and (5) failed fillet welds at the ice basket bottom hold-down bar. The team inspection identified 29 apparent violations of NRC requirements. As stated in Section II. A. of the Director's Decision, these violations were part of the overall enforcement action taken by the NRC.

In the Addendum, the Petitioner raised the concern that the Licensee was aware of the deficiencies with the ice condenser and did not properly report the deficiencies. While the

Licensee's staff had knowledge of some of the inspection issues, it was not apparent that the Licensee was aware of the significance of those issues until they were discovered by the NRC and followed up by the Licensee during the inspection. Contributing to the Licensee's failure to recognize the significance of those issues was the breakdown of the corrective action program. As stated in Section II.A these issues were a part of the overall enforcement action. Therefore, the problems the Licensee's staff identified with the ice condenser were not properly resolved or reported by the Licensee. Following the inspection, the Licensee has submitted several LERs reporting on the deficiencies identified with the ice condenser in accordance with 10 CFR 50.73. In addition, on July 30, 1998, the Licensee issued a report in accordance with 10 CFR Part 21 informing the NRC of potential defects with failed fillet welds at the ice basket hold down bar.

In March 1998, the Licensee decided to completely melt out the ice condensers of both units to allow thorough inspections and comprehensive repairs of the ice condensers. Following the meltout of the ice condensers, the Licensee discovered foreign material in the ice baskets. Some material appeared to be from the original construction. Also, the Licensee identified damage to the ice baskets and other ice condenser components. The restoration of the ice condenser has been incorporated into the Licensee's Restart Plan. The Licensee chose to repair damaged components and reinspect the ice condensers to assure that corrective actions have been adequately implemented and the material condition of the ice condensers has been returned to its original design-basis. In addition to the physical repairs to the ice condenser, the Licensee has reviewed the ice condenser surveillance program and intends to complete revised ice condenser surveillance tests to ensure that the ice condensers are operable and will perform their intended function.

Resolution of the ice condenser problems is an item on the MC 0350 Case Specific Checklist and the Licensee's corrective actions are monitored by the MC 0350 restart panel.

Corrective actions implemented by the Licensee will be inspected before the restart of a unit at D. C. Cook.

C. 10 CFR 50.59 Safety Evaluation Process

During the AE inspection, the NRC inspectors identified problems with the 10 CFR 50.59 process at D. C. Cook. In the Addendum, the Petitioner raised a concern that the Licensee's 10 CFR 50.59 safety evaluation preparation process was "bad" and that a thorough enough review of old 10 CFR 50.59 safety evaluations had not been performed. Further, the Petitioner questioned if safety evaluations prepared using the "bad" 10 CFR 50.59 process potentially could mean that unidentified safety problems remain at D. C. Cook.

Following the AE inspection, the Licensee initiated corrective actions to address the 10 CFR 50.59 issues identified during the AE inspection. The Licensee assessed the 10 CFR 50.59 process in December 1997. The Licensee reviewed 10 CFR 50.59 screenings and unreviewed safety question determinations performed between January 1996 and September 1997. The Licensee identified several administrative or procedural problems. The Licensee's assessment did not identify issues that would have an impact on the technical conclusions reached in any safety evaluation prepared in accordance with the 10 CFR 50.59 process.

To evaluate the corrective actions taken by the Licensee following the AE inspection, the NRC performed an inspection of the 10 CFR 50.59 process at D. C. Cook. The inspectors reviewed procedure and design change safety evaluations. The team did not identify any safety evaluations performed by the Licensee using the "old" 10 CFR 50.59 process that resulted in a safety system operability concern, or where the change would have resulted in an unreviewed safety question determination. The inspection did, however, identify apparent violations of 10 CFR 50.59 concerning the failure to perform safety evaluations for proposed changes to the plant design basis. The violations resulted from the Licensee's failure to

recognize that implemented changes constituted a change to the plant's design basis as described in the UFSAR. Violations were also identified pertaining to the adequacy of safety evaluations. The inspection made it evident that weakness still existed in the Licensee's 10 CFR 50.59 program and substantiated the concerns raised in the Addendum with the Licensee's 10 CFR 50.59 process. The specific details of the findings are contained in the IR No. 50-315, 50-316/98004.

As a result of the inspection findings from both the AE inspection and IR No. 50-315, 50-316/98004, the Licensee has performed three additional self-assessments of the effectiveness of its 10 CFR 50.59 program. The Licensee's review sample was selected from a population of 50.59 safety evaluations beginning in the 1980s. As a result of the deficiencies identified through these self-assessments, the Licensee committed to implement a number of programmatic changes to improve the 10 CFR 50.59 process at D. C. Cook. Further, the Licensee has committed to perform enhanced system readiness reviews as stated above. These commitments have been incorporated into the Licensee's Restart Plan and will be implemented before restart of a unit at D. C. Cook.

Inspections to date of the Licensee's 10 CFR 50.59 process have not identified any safety evaluations performed by the Licensee that resulted in safety system operability concerns. However, the Licensee's enhanced system readiness reviews may discover 10 CFR 50.59 safety evaluations that are inadequate and that may result in safety system operability concerns. Because of the nature and number of 10 CFR 50.59 violations, the NRC placed the 10 CFR 50.59 process on the MC 0350 Case Specific Checklist. Corrective actions taken by the Licensee will be inspected by the NRC staff before restart of a unit at D. C. Cook to assure that the 10 CFR 50.59 program implementation at D. C. Cook provides adequate assurance of safety.

D. Engineering Calculations

In the Addendum, the Petitioner identified concerns involving engineering calculations at D. C. Cook. The Petitioner questioned whether the population of calculations, reviewed by the Licensee as part of the corrective actions taken in response to inspection findings from the AE inspection, was a representative sample. In addition, the Petitioner questioned whether the NRC was satisfied with corrective actions taken by the Licensee in response to the calculation weaknesses identified by the NRC during the AE inspection.

The NRC inspected the corrective actions taken by the Licensee in this area. The NRC inspection findings were documented in NRC IR No. 50-315, 50-316/98004. The inspection concluded that the older calculations (early 1970 vintage) appeared to satisfy their intended purpose; however, problems still existed with calculations at D. C. Cook and the initial corrective actions implemented by the licensee had been unsuccessful in bounding the problem.

On the basis of the inspection findings, the Licensee chose to expand the scope of engineering calculations to be reviewed to determine the quality, level of detail, completeness and accuracy of the calculations before restart of a unit. The Licensee expanded its review to include a significant sample of the calculations for the most risk significant systems. The Licensee's expanded review identified a number of deficiencies in engineering calculations. As a result of these deficiencies, the Licensee has committed to corrective actions to change the calculation preparation procedure and to train all calculation preparers, verifiers, and approvers on the new procedures.

In summary, because of the extent of the problems with engineering calculations and design control at D. C. Cook, the MC 0350 restart panel incorporated this issue into the Case Specific Checklist. Before restart of a unit at D. C. Cook, the NRC will evaluate corrective actions taken by the Licensee to assess whether the Licensee has been successful in

correcting the weakness in the engineering calculation program at D. C. Cook and that the calculation adequacy provides reasonable assurance of safety.

E. Net Positive Suction Head (NPSH) Calculations

In the Addendum, the Petitioner stated that from the time the Petition was submitted on October 9, 1997, until the time the Licensee responded to the CAL on December 2, 1997, the Petitioner received concerns from an individual at D. C. Cook indicating problems with NPSH calculations. The alleged problems involved both missing and inaccurate calculations. The Petitioner questioned if safety-related pumps at D. C. Cook have adequate NPSH as shown by quality calculations.

In response to the concerns raised in the Addendum, the NRC staff requested by letter dated June 8, 1998, that the Licensee provide (1) the NPSH calculations for all safety-related pumps, (2) a description of the calculation technique, and (3) all assumptions used in the calculations. By letters dated July 22, July 31, and August 5, 1998, the Licensee provided the requested information.

The NRC staff reviewed the NPSH calculations for each safety-related pump at D. C. Cook. With the exception of the containment spray (CTS) and the residual heat removal (RHR) systems, the NRC found that the calculations submitted by the Licensee supported adequate NPSH for the safety related pumps. For the CTS and RHR systems the values used for the pump run out flows in the UFSAR did not match the values used in the NPSH calculations. Because of the inconsistencies in the values used for the pump run-out flows, the NRC was unable to determine whether the NPSH calculations of record for the CTS and RHR systems demonstrated adequate NPSH for the pumps in these systems. By letter dated January 7, 1999, the NRC informed the Licensee of the inconsistencies discovered during the review of the NPSH calculations. Further, the letter requested the Licensee to provide revised

NPSH calculations addressing the inconsistencies in the CTS and RHR systems NPSH calculations, and show that adequate NPSH is available for the safety-related pumps in these systems. In addition, the issue of adequate NPSH for safety-related pumps will be monitored by the MC 0350 restart panel. The Licensee's resolution of the issue will be reviewed and evaluated by the NRC.

In summary, the Petitioner stated that there were missing and inaccurate NPSH calculations for safety-related pumps at D. C. Cook. Upon request, the Licensee provided the NPSH calculation for all safety-related pumps at D. C. Cook. The Licensee's response demonstrated that there were NPSH calculations for all safety-related pumps at D. C. Cook. When the calculations were reviewed by the NRC, inconsistencies were discovered in values documented in the UFSAR and those used in the NPSH calculations. These concerns have been identified and transmitted to the Licensee. The Licensee's corrective actions will be monitored through the MC 0350 process to ensure appropriate actions are taken.

F. Licensee's Response to the CAL

In the Addendum, the Petitioner raised a concern about the credibility of the Licensee's response to the CAL. The Petitioner stated that since the Licensee's February 6, 1997, response to the NRC's October 9, 1996, 10 CFR 50.54(f) request for design-basis information was not accurate, based on the AE inspection finding, he could not see how the Licensee's response to the CAL could be accurate.

Following the Licensee's response to the CAL, the NRC performed additional inspections at D. C. Cook, documented in IR Nos. 50-315, 50-316/98004; 50-315, 50-316/98005; and 50-315, 50-316/98009. The findings of these inspections clearly showed that the Licensee's actions to bound the scope of engineering problems in response to the CAL were too narrowly focused and were not sufficient to address the broad array of problems

concerning the design-basis and licensing-basis issues that existed at D. C. Cook.

The Petitioner's concern in the Addendum (that the Licensee's response to the CAL failed to assure the NRC that corrective actions were adequate) has been substantiated. The inspection findings from early 1998 indicated that the CAL response did not bound the design-basis and licensing-basis issues at D. C. Cook. As indicated in Section II.A of the Director's Decision, the NRC took escalated enforcement action against the Licensee. In response to the violations and various programmatic breakdowns at D. C. Cook, the Licensee made a decision in early 1998 to perform a comprehensive assessment to provide reasonable assurance of plant system readiness, programmatic readiness, functional area readiness, and containment readiness before restart of either unit. The Licensee's primary mechanism to implement each of the plant assessment programs is the D. C. Cook Nuclear Plant Restart Plan. The Restart Plan was submitted in March 1998, and Revision 4 of the Restart Plan was docketed on December 16, 1998. As stated above, the NRC is using the guidelines in MC 0350 to oversee the Licensee's corrective actions and the readiness of a unit to restart. As additional problems or concerns are identified during the implementation of the Restart Plan, appropriate adjustments will be made to the Restart Plan and the Case Specific Checklist.

III. NRC RESPONSE TO REQUESTED ACTION

A. Request To Modify, Revoke, or Suspend the Operating Licenses for D. C. Cook, Units 1 and 2

The Petitioner requested that the operating licenses for D. C. Cook, Units 1 and 2 be modified, revoked, or suspended to prevent operation of the units until there is reasonable assurance that significant non-compliances have been identified and corrected so that systems

are in conformance with their design-basis and licensing-basis requirements. In addition, the Petition requested that the NRC broaden the inspection scope at D. C. Cook following the AE inspection. The NRC's regulatory oversight actions taken thus far at D. C. Cook, in part, fulfill the actions requested in the Petition. The regulatory oversight actions at D. C. Cook are broad and comprehensive and will ensure that there is reasonable assurance of safety prior to restart of either unit.

Inspection findings at D. C. Cook following the AE inspection verified that the corrective actions implemented by the Licensee as described in the CAL response were too narrowly focused and did not fully address the design-basis and licensing-basis issues. The NRC increased inspections at D. C. Cook identified a number of violations of NRC requirements, and as a result, took appropriate enforcement action against the Licensee as stated above. While the enforcement action did not modify, suspend, or revoke the operating licenses of the D. C. Cook facilities, it did emphasize the serious nature of the violations, the duration of the problems, and the Licensee's poor performance.

The Licensee has developed an integrated Restart Plan. The plan provides the frame-work to be used by the Licensee to identify, evaluate, and correct issues. The NRC regulatory oversight at D. C. Cook is following the guidelines of MC 0350 as discussed above. This approach focuses the correct level of management attention as well as resources on significant issues to be verified before restart of a unit at D. C. Cook. In addition, this approach allows the NRC the flexibility to change the focus of the oversight as different significant issues emerge. In the Licensee's effort to identify and correct issues, new issues will continue to emerge. As a result, the Licensee will be expected to modify the Restart Plan to ensure that corrective actions, to resolve the emergent issues, are implemented in a timely manner. The

MC 0350 restart panel will review these changes to the Restart Plan to ensure that the Licensee has taken appropriate corrective actions.

The Petitioner's request to suspend, modify, or revoke the licenses at D. C. Cook, Units 1 and 2 has not been granted at this time. The current regulatory oversight at D. C. Cook is sufficient, and provides reasonable assurance that before restart of a unit at D. C. Cook the Licensee will have identified and corrected issues so that the safety systems at D. C. Cook will be in compliance with their design-basis and licensing-basis requirements.

B. Request To Hold a Public Hearing on the Issues Raised in the Petition Before Restart of a Unit at D. C. Cook

The Petitioner requested that a public hearing into the issues raised in the Petition be held in the Washington, D.C. area before the first unit at D. C. Cook is authorized to restart. As discussed above, this request was granted. On August 19, 1998, an informal public hearing was held at the NRC headquarters in Rockville, Maryland. Both the Petitioner and the Licensee made presentations during the hearing. The hearing gave the Petitioner an opportunity to clarify the issues raised in the Petition and the Addendum.

C. Issues Raised in the Addendum

As discussed in Sections II. B. through II. E. of this Director's Decision, each of the actions requested by the Petitioner in the Addendum has been granted in that the Licensee is taking additional corrective actions to ensure that each issue raised in the Addendum will be resolved before restart of a unit at D. C. Cook, and the NRC will verify that the Licensee's corrective actions have been effective. Each of the issues raised in the Addendum will be reported on in a future inspection report.

IV. CONCLUSION

The NRC has determined, for the reasons given in the preceding discussion, that the request to prevent operation of the units at D. C. Cook until there is reasonable assurance that significant non-compliances have been identified and corrected so that systems are in conformance with their design-basis and licensing-basis requirements has been satisfied. The regulatory oversight actions being taken by the NRC as stated above will provide reasonable assurance that systems at D. C. Cook will be in conformance with their design-basis and licensing-bases, thus meeting the request made in the Petition and eliminates the need to modify, suspend, or revoke the licenses at D. C. Cook. The request to hold a public hearing into the issues raised in the Petition and Addendum in the Washington, D.C. area before the first unit at D. C. Cook is authorized to restart has been granted. Action has been taken on each concern identified in the Addendum, as stated above.

As provided for in 10 CFR 2.206(c), a copy of this Decision will be filed with the Secretary of the Commission for the Commission's review. This Decision will constitute the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes review of the Decision at that time.

Dated at Rockville, Maryland this 11th day of February 1999.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

7590-01-P

'99 FEB 16 P4:37

UNITED STATES NUCLEAR REGULATORY COMMISSION

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

ISSUANCE OF DIRECTOR'S DECISION UNDER 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation has issued a Director's Decision with regard to a Petition dated October 9, 1997, and an Addendum to the Petition dated January 12, 1998, filed by Mr. David Lochbaum on behalf of the Union of Concerned Scientists, hereafter referred to as the "Petitioner." The Petition pertains to the Donald C. Cook Nuclear Plant, Units 1 and 2 (D. C. Cook).

The Petitioner requested that the operating licenses for D. C. Cook be modified, revoked, or suspended to prevent operation of the units until there is reasonable assurance that significant non-compliances have been identified and corrected so that systems are in conformance with their design-basis and licensing-basis requirements. The Petition also requested that a public hearing into this matter be held in the Washington, D.C. area before the first unit at D. C. Cook is authorized to restart. As the basis for these requests, the Petitioner stated that the NRC completed an architect/engineering (AE) design inspection at D. C. Cook (NRC Inspection Report (IR) No. 50-315, 50-316/97201) on November 26, 1997. Findings by the NRC during the AE inspection led to the Licensee having to declare the emergency core cooling system (ECCS) inoperable at both units at D. C. Cook. As a result, the Licensee shut down both units in accordance with its Technical Specifications (TS). The systems reviewed during the AE inspection were the same systems that the Licensee had previously reviewed as part of its design-basis documentation reconstitution program, and the program did not identify any deficiencies concerning system operability. Therefore, the Petitioner asserted that the

Licensee's design-basis documentation reconstitution programs lacked the necessary rigor and focus to identify potential design-related operability issues. The Petitioner further asserted that deficiencies in the Licensee's design control programs may also be responsible for similar issues in safety systems that have not been examined by the NRC. On the basis of this potential, the Petitioner also requested that the NRC increase the inspection scope at D. C. Cook. By letter dated January 12, 1998, the Petitioner issued an Addendum to the Petition. The following six specific concerns were raised in the Addendum:

(1) ice condenser concerns

(2) 10 CFR 50.59 Safety Evaluation process

(3) engineering calculations

(4) net positive suction head (NPSH) calculations

(5) licensee's response to the CAL

(6) NRC inspection process

The Addendum also raised concerns about the 10 CFR 2.206 process, the NRC inspection process, and generic concerns with ice condenser containments. On February 23, 1998, the NRC acknowledged receipt of the additional information and informed the Petitioner that all specific concerns related to the D. C. Cook plant and the Petition would be considered in the Director's Decision. Further, the NRC informed the Petitioner that the concerns not directly applicable to the request in the Petition would be evaluated and transmitted to the Petitioner in separate correspondence. By letters dated July 10 and December 28, 1998, the NRC sent the Petitioner the status of the review of these issues.

On August 19, 1998, an informal public hearing was held at the NRC headquarters in Rockville, Maryland. Both the Petitioner and the Licensee made presentations during the hearing. The hearing gave the Petitioner an opportunity to clarify the issues raised in the Petition and the Addendum.

The Director of the Office of Nuclear Reactor Regulation has determined that the request to prevent operation of the units at D. C. Cook until there is reasonable assurance that significant non-compliances have been identified and corrected so that systems are in conformance with their design-basis and licensing-basis requirements has been satisfied. The regulatory oversight actions being taken by the NRC will provide reasonable assurance that systems at D. C. Cook will be in conformance with their design bases and licensing bases, thus meeting the request made in the Petition and eliminates the need to modify, suspend or revoke the licenses at D. C. Cook. The reasons for this decision are explained in the Director's Decision Pursuant to 10 CFR 2.206 (DD-99-03), the complete text of which follows this notice and is available for public inspection at the Commission's Public Document Room, Gelman Building, 2120 L Street, NW., Washington, D. C., and at the local public document room located at Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, MI 49085.

A copy of the Director's Decision will be filed with the Secretary of the Commission for the Commission's review in accordance with 10 CFR 2.206 of the Commission's regulations. As provided for by this regulation, the Decision will constitute the final action of the Commission 25 days after the date of issuance, unless the Commission on its own motion institutes a review of the Decision in that time.

Dated at Rockville, Maryland, this 11th day of February 1999.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 9, 1997

Mr. David A. Lochbaum
Union of Concerned Scientists
1616 P Street, NW., Suite 310
Washington, DC 20036-1495

Dear Mr. Lochbaum:

This letter acknowledges receipt of a Petition dated October 9, 1997, submitted by you on behalf of the Union of Concerned Scientists (Petitioner) addressed to the Executive Director for Operations of the U.S. Nuclear Regulatory Commission (NRC). The Petition requests that the operating licenses for Donald C. Cook Nuclear Plant, Units 1 and 2, operated by American Electric Power Company (the Licensee) be modified, revoked, or suspended until there is reasonable assurance that the Licensee's systems are in conformance with design- and licensing-bases requirements. The Petition also requests that a public hearing into this matter be held in the Washington, DC area before the first unit at D.C. Cook is authorized to restart.

As the basis for these requests, the Petition states that the NRC recently completed an architect/engineer design inspection at D.C. Cook. The Licensee had previously reviewed the same systems as part of its design-basis documentation reconstitution program. Findings by the NRC inspection team led to a shutdown of both units and have necessitated changes to the plant's physical configuration. The NRC issued a Confirmatory Action Letter (CAL) with regard to these matters on September 19, 1997. Therefore, the Petition asserts that the Licensee's design-basis documentation reconstitution and updated final safety analysis report validation programs lack the necessary rigor and focus. The Petition further asserts that deficiencies in the Licensee's design control programs may also be responsible for similar problems in its safety systems which were not examined by the NRC. The Petition requests that systems with a safety function at D.C. Cook must be certified to be capable of performing their required function under all design conditions before restart.

Your Petition has been referred to me for preparation of a response pursuant to 10 CFR 2.206 of the Commission's regulations. As provided by 10 CFR 2.206, action will be taken on your request within a reasonable time. With regard to your issue of review of safety systems at D.C. Cook before restart, the CAL recognized the importance and need for an assessment of this issue and requested that the Licensee perform such an assessment. The Licensee agreed to provide a written response to the CAL, including an assessment of safety system operability, before restart. The NRC will consider the Licensee's submittal in making a restart decision and will inform you of our finding.

The Licensee responded to the CAL on December 2, 1997, and my staff informed you of where to obtain a copy on the Licensee's website. With regard to your request for a public hearing, the NRC staff will determine whether an informal public hearing as described in NRC Management Directive 8.11, "Review Process for 10 CFR 2.206 Petitions", copy enclosed, is appropriate after evaluation of the Licensee's submittal referred to above.

I have enclosed for your information a copy of the notice that is being filed with the Office of the Federal Register for publication. I have also enclosed for your information a pamphlet on the public petition process.

Sincerely,

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures: 1. NRC Management Directive 8.11
2. Federal Register Notice
3. Public Petition Process Pamphlet

cc: See next page

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI 49106

Al Blind, Site Vice President
Donald C. Cook Nuclear Plant
1 Cook Place
Bridgman, MI 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Steve J. Brewer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

w/copy of incoming 2.206 request:
E.E. Fitzpatrick, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

UNITED STATES NUCLEAR REGULATORY COMMISSION

AMERICAN ELECTRIC POWER COMPANY

DOCKET NOS. 50-315 AND 50-316

RECEIPT OF PETITION FOR DIRECTOR'S DECISION UNDER 10 CFR 2.206

Notice is hereby given that by Petition dated October 9, 1997, David A. Lochbaum, on behalf of the Union of Concerned Scientists, has requested that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to Donald C. Cook Nuclear Plant, Units 1 and 2, operated by American Electric Power Company (the Licensee). The Petition requests that the operating licenses for D.C. Cook, Units 1 and 2, be modified, revoked, or suspended until there is reasonable assurance that the Licensee's systems are in conformance with design- and licensing-bases requirements. The Petition requests that systems with a safety function at D.C. Cook be qualified and capable of performing their required function under all design conditions before restart. The Petition also requests that a public hearing into this matter be held in the Washington, DC, area before the first unit at D.C. Cook is authorized to restart.

As the basis for these requests, the Petition states that the NRC recently completed an architect/engineer design inspection at D.C. Cook. The Licensee had previously reviewed the same systems as part of its design-basis documentation reconstitution program. Findings by the NRC inspection team led to a shutdown of both units and has necessitated changes to the plant's physical configuration. Therefore, the Petition asserts that the Licensee's design-basis documentation reconstitution and updated final safety analysis report validation programs lack the necessary rigor and focus. The Petition further asserts that deficiencies in

the Licensee's design control programs may also be responsible for similar problems in its safety systems, which were not examined by the NRC.

The request is being treated pursuant to 10 CFR 2.206 of the Commission's regulations. The request has been referred to the Director of the Office of Nuclear Reactor Regulation. As provided by 10 CFR 2.206, appropriate action will be taken on this Petition within a reasonable time. A copy of the Petition is available for public inspection at the Commission's Public Document Room, located at the Gelman Building, 2120 L Street, NW., Washington, DC 20555-0001.

Dated at Rockville, Maryland, this 9th day of December 1997.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 23, 1998

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

Dear Mr. Lochbaum:

This letter acknowledges receipt of the January 12, 1998, Addendum to your Petition dated October 9, 1997, regarding the Donald C. Cook Nuclear Plant, Units 1 and 2, operated by American Electric Power Company (the Licensee) and a January 15, 1998, request for a copy of a special team inspection report. The Addendum discusses the 2.206 process and six concerns covering a broad range of issues. Your January 15, 1998, supplement requests the opportunity to review the report of a special team inspection conducted at the D.C. Cook facility in January 1998 before restart of the D.C. Cook facility. The NRC will address the actions to be taken with respect to each item separately.

A portion of your January 12, 1998, letter addresses the 2.206 petition process itself. This issue, whether the 2.206 process is functioning effectively, is not directly applicable to the request in your Petition for certain actions preceding restart of the first unit at D.C. Cook. I will correspond to you separately regarding this issue.

The first of your technical concerns in the January 12, 1998, letter relates to D.C. Cook's ice condenser containment. You question if the ice condenser problems identified at Watts Bar also apply to D.C. Cook. Specifically, you discuss problems with the bay doors and components of the ice baskets. Also, you discuss problems in the configuration and testing of the ice condenser at Watts Bar. You also state that the problems were known, but were not properly reported to NRC by several licensees and the vendor. The specific concerns regarding the applicability of the ice condenser problems to Cook will be addressed in my response to your Petition. The concern that the problems were known, but not properly reported by several licensees and the vendor, will be handled separate from the 2.206 process.

Your second concern addresses the licensee's 50.59 process and questions whether the licensee's safety evaluations had been assessed. Your third concern involves the scope of the licensee's review of engineering calculations and the NRC's assessment of that review. Your fourth concern pertains to missing and inaccurate net positive suction head calculations at D.C. Cook. Given the specific applicability of these issues to D.C. Cook, they will be addressed as part of your Petition.

You also state that the NRC's architect-engineering (AE) inspection at D.C. Cook found significant issues in systems that were previously examined by the licensee. In view of the licensee's inability to identify the issues previously, you question why the evaluation performed

in response to the confirmatory action letter (CAL) should be considered adequate. As this pertains directly to the issue in your original Petition, this issue will also be addressed as part of your Petition and is considered your fifth concern.

Your sixth concern involves the adequacy of the NRC's inspection process. Noting that an inspection finding represents both a nonconforming condition and a programmatic failure that allowed the problem to go undetected, you question if the NRC inspection process should have defined criteria for expanding the inspection scope based on findings. This concern is not directly applicable to the request in your Petition for certain actions to be taken before restart of the first unit at D.C. Cook. Therefore, I will correspond with you separately regarding this issue.

Your January 15, 1998, letter also notes that a special inspection team had been sent to D.C. Cook to review a sample of the licensee's corrective actions in response to the CAL, to validate the licensee's root-cause analysis, and to assess the adequacy of the Licensee's conclusions. You also state that this inspection will be a primary factor in the staff's decision on whether the licensee's actions pursuant to the CAL are adequate and whether the licensee's decision to restart D.C. Cook Unit 1 is proper and will form a basis for responding to the Petition. Therefore, you request that the NRC send you a copy of the special inspection report, even if it is a preliminary version subject to revision, at least one business day before lifting of the CAL. With regard to your request for the draft inspection report, it is NRC policy not to release draft information. This policy is intended to prevent improper influences and assure that predecisional information, or contemplated enforcement actions, are not compromised by a premature release. Based on this, I must deny your request for a draft copy of the report of the special inspection team. However, my staff will ensure that you receive a copy of the final inspection report when it is issued. In addition, as previously discussed with you, my staff will contact you prior to the lifting of the CAL to inform you of that decision.

You should be aware that the special team inspection results will only be one of several inputs into NRC's review of the licensee's restart readiness. Other items that will be considered are the licensee's responses to the CAL, the public meetings held with the Licensee to discuss its actions, oversight of licensee activities by the NRC resident inspector staff, and inspections conducted on particular issues, such as the licensee's response to the ice condenser issues raised at Watts Bar.

Please contact the NRC project manager assigned to oversee activities at D.C. Cook, John Hickman, at (301) 415-3017, if you have any questions.

Sincerely,

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

cc: See next page

Indiana Michigan Power Company

Donald C. Cook Nuclear Plant
Units 1 and 2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI 49106

Al Blind, Site Vice President
Donald C. Cook Nuclear Plant
1 Cook Place
Bridgman, MI 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Steve J. Brewer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

w/copy of incoming 2.206 request:
E.E. Fitzpatrick, Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

ACTION

EDO Principal Correspondence Control

FROM: DUE: 11/12/97 EDO CONTROL: G970724
DOC DT: 10/09/97
FINAL REPLY:

David A. Lochbaum
Union of Concerned Scientists

TO:

Callan, EDO

FOR SIGNATURE OF : ** GRN ** CRC NO:

Collins, NRR

DESC: ROUTING:

PETITION PURSUANT TO 10 CFR 2.206, DONALD C. COOK NUCLEAR PLANTS UNITS 1 AND 2, DOCKET NOS. 50-315 AND 50-316

Callan
Thadani
Thompson
Norry
Blaha
Burns
Beach, RIII
Cyr, OGC
JGoldberg, OGC
JKennedy, NRR

DATE: 10/14/97

ASSIGNED TO: CONTACT:

NRR Collins

SPECIAL INSTRUCTIONS OR REMARKS:

NRR RECEIVED: OCTOBER 15, 1997
NRR ACTION: DRPW:ADENSAM

NRR ROUTING:
COLLINS
MIRAGLIA
ZIMMERMAN
SHERON
ROE
TRAVERS
BOHRER

ACTION
DUE TO NRR DIRECTOR'S OFFICE
BY Nov 6, '97

UNION OF
CONCERNED
SCIENTISTS

DOCKETED
USNRC

'99 FEB 16 P4:38

OF
RUL
ADJU

October 9, 1997

Mr. L. Joseph Callan
Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PETITION PURSUANT TO 10 CFR 2.206, DONALD C. COOK NUCLEAR PLANTS UNITS 1 AND 2, DOCKET NOS. 50-315 AND 50-316

Dear Mr. Callan:

The Union of Concerned Scientists submits this petition pursuant to 10 CFR 2.206 requesting that the operating licenses for Donald C. Cook Units 1 and 2 be modified, revoked, or suspended until there is reasonable assurance that their systems are in conformance with design and licensing bases requirements. A process comparable to the system certifications recently used by the Salem and Millstone licensees would provide this necessary level of assurance. UCS additionally requests that a public hearing into this matter be held in the Washington, DC area prior to the first unit at D C Cook being authorized to restart. At this hearing, we will present information supporting the contentions in this petition.

Background

On October 9, 1996, the NRC requested that its power reactor licensees provide information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design bases information. The NRC's issued this request as a result of its investigations at the Millstone Power Station. The licensee for the D C Cook plant responded with a letter dated February 6, 1997, describing the administrative controls it uses to provide assurance that the Cook Nuclear Plant is operated and maintained within the established design bases.

An NRC team recently conducted an architect/engineer design inspection at D C Cook. According to the NRC's Project Manager for D C Cook, this NRC team examined two safety systems and their supporting systems. The team's findings forced the licensee to shut down both units on September 10, 1997.

The NRC issued a confirmatory action letter to the licensee dated September 19, 1997, specifying issues arising from the design inspection that must be resolved prior to restarting the units. These issues (listed in Attachment 1) include physical modifications to the plants and revisions to the plants' operating licenses. Numerous NRC Daily Event Reports (listed in Attachment 2) described the findings from design inspection as reported by the licensee. The NRC has not yet released the design inspection report and we have been told that it will not be issued until next week at the earliest.

EDO -- G970724

Washington Office: 1616 P Street NW Suite 310 • Washington, DC 20036-1495 • 202-332-0900 • FAX: 202-332-0905
Cambridge Headquarters: Two Brattle Square • Cambridge, MA 02238-9105 • 617-547-5552 • FAX: 617-864-9405
California Office: 2397 Shattuck Avenue Suite 203 • Berkeley, CA 94704-1567 • 510-843-1872 • FAX: 510-843-3785

PRINTED ON RECYCLED PAPER

Basis for Requested Action

The NRC conducted architect/engineer design inspections at only six of its nearly 70 operating power reactor licensee sites. These design inspections examined only one or two safety systems along with their supporting systems at each site. The NRC Project Manager reported that the design inspection at D C Cook examined the residual heat removal and component cooling water systems along with their supporting systems. These design inspections focused on the facilities' original design and the licensees' conformance with the safety analysis reports.

The systems examined by the NRC at D C Cook had already been covered by the licensee's design basis documentation reconstitution program. Design basis documents (DBDs) for the containment, containment structure, containment spray, emergency core cooling, component cooling water, and residual heat removal systems had been approved by the licensee prior to the NRC team's arrival. The licensee informed the NRC that its DBD program had not identified any deficiencies involving equipment operability.

The findings by the NRC design inspection team prompted the licensee to declare both trains of the emergency core cooling systems and the containment spray system inoperable. The units were shut down on September 8 and 9, 1997. The licensee reported making physical changes to the plant to correct some of the problems and indicated that additional physical changes may be required.

The licensee has proposed fixing the specific operability issues identified during the NRC design inspection and then restarting the units. Confining the scope of the restart activities in this way would be treating the symptoms rather than the cause of the problems. The NRC design inspection revealed serious deficiencies in the licensee's design control programs. These deficiencies created the specific problems that forced the plants to be shut down. These deficiencies may also be responsible for similar problems in other safety systems which were not examined by the NRC.

It is important to note that the NRC identified significant operability problems in systems that the licensee had covered in recently approved DBDs. The licensee stated in its February 6, 1997, submittal that it verifies and validates the information in its DBDs via reviews and physical plant walkdowns prior to their approval. Thus, the NRC discovered significant problems in systems which had been closely scrutinized by the licensee. Had the NRC's findings involved systems which have not yet been covered under the licensees' DBD program, it might be reasonable to assume that the licensee would have identified them at that later date. However, there is little reason to believe that these problems would have been resolved unless the NRC had identified them.

Attachment 2 lists NRC Daily Event Reports (DERs) involving issues identified by the NRC design inspection at D C Cook. DER Nos. 32740, 32806, 32822, 32839, 32843, 32875, 32890, 32904, 32914, 32915, 32921, 32948, and 32988 describe potential deficiencies that appear to have existed at D C Cook prior to the initiation of its design basis documentation reconstitution effort in 1992. That effort was therefore apparently unable to detect these potential deficiencies. DER Nos. 32823, 32824, 32903, 32939, and 32948 describe potential deficiencies that appear to have been introduced since 1992. Thus, the licensee's design control and quality assurance programs are apparently unable to ensure that the facility is maintained within its design bases.

UCS feels that the design basis documentation reconstitution and Updated Final Safety Analysis Report (UFSAR) validation programs as described in the licensee's response to the NRC's 50.54(f) letter lack the rigor and focus necessary to identify potential design-related operability issues. Our conviction is supported by the findings from the NRC design inspection. Since the corrections to the NRC's findings were not limited to mere paperwork fixes but included actual changes to the plant's physical configuration, the safety significance of these and potentially other undetected problems cannot be understated.

The flaws in the licensee's design control programs must be corrected. The systems at D C Cook, at least those with a safety function, must be certified to be capable of performing their required actions under all design conditions. Then, and only then, can the units be restarted with reasonable assurance that public safety will be adequately protected. It would be irresponsible to restart these units knowing that the programmatic failures that caused the safety problems identified by the NRC team may have produced comparable problems affecting the operability of other safety systems.

The legal precedent for our position is stated by the NRC's Atomic Safety and Licensing Appeal Board in the Matter of Vermont Yankee Nuclear Power Corporation , Memorandum and Order (ALAB-138), dated July 31, 1973:

> "As a general rule, the Commission's regulations preclude a challenge to applicable regulations in an individual licensing proceeding. 10 CFR 2.758. This rule has been frequently applied in such proceedings to preclude challenges by intervenors to Commission regulations. Generally, then, an intervenor cannot validly argue on safety grounds that a reactor which meets applicable standards should not be licensed. By the same token, neither the applicant nor the staff should be permitted to challenge applicable regulations, either directly or indirectly. Thus, those parties should not generally be permitted to seek or justify the licensing of a reactor which does not comply with applicable standards. Nor can they avoid compliance by arguing that, although an applicable regulation is not met, the public health and safety will still be protected. **For, once a regulation is adopted, the standards it embodies represent the Commission's definition of what is required to protect the public health and safety.**" [emphasis added]
>
> "In short, in order for a facility to be licensed to operate, the applicant must establish that the facility complies with all applicable regulations. **If the facility does not comply, or if there has been no showing that it does comply, it may not be licensed.**" [emphasis added]

The NRC design inspection at D C Cook identified significant issues which caused both units to be shut down. These issues were caused by programmatic deficiencies in the licensee's design control programs. A contributing factor for these issues is the failure of the licensee's quality assurance and self-assessment programs to detect these problems. Nothing in the reported findings from the design inspection supports a conclusion that these findings are isolated consequences. The NRC's design inspection invalidates any showing that this facility complies with all applicable regulations. Therefore, the design control deficiencies must be corrected to prevent future non-compliances with safety regulations. And just as importantly, a thorough review of all systems with safety functions must be completed prior to restart to detect and correct past non-compliances.

UCS is not advocating that the NRC apply a higher standard at D C Cook. Instead, we are requesting that the NRC ensure that the D C Cook facility is in accordance with the minimum safety standards which constitute the legal grounds for allowing the units to operate. Our request is consistent with the measures required by the NRC when other sampling inspections find problems. We ask the NRC to expand the inspection scope based upon the identified problems just as would be required when snubber (e.g., pipe restraint) and reactor vessel internals inspections found problems.

Requested Actions

UCS petitions the NRC to protect public health and safety by preventing the units at D C Cook from operating until such time that there is reasonable assurance that all significant non-compliances have been identified and corrected. The system certification process recently used at the Salem Generating Station and the Millstone Power Station would provide such reasonable assurance. We request a public hearing on this matter be held in the Washington, DC area before any unit at D C Cook is authorized to restart.

Sincerely,

David A. Lochbaum
Nuclear Safety Engineer

cc: Chairman Shirley Ann Jackson
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Mr. A. B. Beach, Regional Administrator
United States Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Honorable Fred Upton
United States House of Representatives
Washington, DC 20515-2206

Honorable Spencer Abraham
United States Senate
Washington, DC 20510-2203

Honorable Carl Levin
United States Senate
Washington, DC 20510-2202

Attachments:

1) Design Inspection Issues That Will Be Resolved Prior to D C Cook Restart
2) NRC Daily Event Reports on D C Cook Design Inspection Findings

Attachment 1
Design Inspection Issues That Will Be Resolved Prior to D C Cook Restart

The following issues, quoted verbatim, were specified on the NRC's Confirmatory Action Letter dated September 19, 1997, as requiring resolution prior to restart of any D C Cook unit:

1. **Recirculation Sump Inventory/Containment Dead Ended Compartments Issue**

   Analyses will be performed to demonstrate that the recirculation sump level is adequate to prevent vortexing, or appropriate modifications will be made. [See also Attachment 2 - Power Reactor Event Number 32890]

2. **Recirculation Sump Venting Issue**

   Venting will be re-installed in the recirculation sump cover. The design will incorporate foreign material exclusion requirements for the sump. [See also Attachment 2 – Power Reactor Event Numbers 32875 and 32903]

3. **Thirty-six Hour Cooldown, with One Train of Cooling**

   Analyses will be performed that will demonstrate the capability to cool down the units consistent with design basis requirements and necessary changes to procedures will be completed.

4. **ES-1.3 (Switchover to Recirculation Sump) Procedure**

   Changes to the emergency procedure used for switchover of the emergency core cooling and containment spray pumps to the recirculation sump will be implemented. These changes will provide assurance there will be adequate sump volume, with proper consideration of instrument bias and single failure criteria. [See also Att. 2 – Power Reactor Event Numbers 32806 and 32904]

5. **Compressed Air Overpressure Issue**

   Overpressure protection will be provided downstream of the 20 psig, 50 psig, and 85 psig control air regulators to mitigate the effects of a postulated failed regulator. [See also Attachment 2 – Power Reactor Event Numbers 32939 and 32988]

6. **Residual Heat Removal (RHR) Suction Valve Interlock Issue**

   A technical specification change to allow operation in mode 4 with the RHR suction valves open and power removed is being processed. Approval of this change by the NRC will be required prior to restart. [See also Attachment 2 – Power Reactor Event Numbers 32914 and 32921]

7. **Fibrous Material in Containment**

   Removal of fibrous material from containment that could clog the recirculation sump will be completed. [See also Attachment 2 – Power Reactor Event Number 32948]

Attachment 2
NRC Daily Event Reports on D C Cook Design Inspection Findings

The following summaries were taken from the daily event reports available on the NRC's website (www.nrc.gov). The only editing involved deletion of unnecessary detail, such as who was notified about the events, and the addition of clarification for acronyms. Otherwise, these narratives are verbatim.

POWER REACTOR EVENT NUMBER: 32890
UNUSUAL EVENT DECLARED & TECHNICAL SPECIFICAITON REQUIRED SHUTDOWN ON BOTH UNITS DUE TO INOPERABLE CONTAINMENTS

As a result of issues raised during the ongoing architect/engineer design inspection, the licensee was reviewing the design aspects of the containments (both units have similar containments). After consulting with the nuclear steam supply system supplier (Westinghouse) the licensee determined that concerns existed about whether adequate communication (flow paths) exists between the active and inactive portions of the containment sump.

During certain scenarios, the volume of water flow back to the containment recirculation sump may not be adequate to support long-term emergency core cooling (ECC) systems (RHR [residual heat removal] system, safety injection system, charging system) or containment spray pump operation during the recirculation phase of a large or small break LOCA. The containment drainage system is designed to ensure that water entering the containment from the breach in the reactor coolant system, ECC systems injection, and ice condenser melt flows back into the containment recirculation sump via drains. Licensee analysis was unable to confirm that sufficient communication existed between inactive and active volumes of the containment to ensure adequate drainage to the recirculation sump. Without adequate drainage into the sump, a low sump level will result, which jeopardizes long term operation of the ECC Systems and containment spray pumps due to vortexing and air entrainment.

As a conservative measure because of these concerns, the licensee declared both trains of the ECC Systems and the containment spray system inoperable for both units and entered Technical Specification limiting condition for operation action statement 3.0.3 to shut down both units. The licensee commenced shutting Unit 1 down from 100% power at 1655 and Unit 2 down from 100% power at 1728. At 2000, the licensee declared an unusual event on both units due to the potential loss of containment barrier on both units.

The licensee plans to perform further analysis to determine the extent of the existing communication between the portions of the sumps and whether plant modifications will be necessary.

Update @ 0311 EDT on 09/10/97 by Tilly taken by MacKinnon

The unusual event was terminated and exited at 0303 EDT when Unit 1 entered mode 5 (cold shutdown). Unit 2 entered mode 5 at 0015 EDT (cold shutdown).

POWER REACTOR EVENT NUMBER: 32875
FAILURE TO MAINTAIN THE CONTAINMENT RECIRCULATION SUMP 1/4" PARTICULATE RETENTION REQUIREMENT (HISTORICAL ISSUE)

A 1/4" particulate retention requirement for the containment recirculation sump was not properly established in 1979 following sump modifications. The containment recirculation sump requirement to retain 1/4" particles is to ensure that containment spray nozzles do not become plugged. The containment spray system takes suction from the containment recirculation sump following injection of the refueling water storage tank supply during a loss of coolant accident.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

In 1979, modifications were performed on the containment recirculation sump. One of the modifications involved moving a 1/4" retention element from inside the recirculation sump to the entrance of the sump. When the retention element was moved, the 1/4" retention requirement was not fully addressed, and pathways exceeding the 1/4" requirement were inadvertently established. The inadvertent pathways established included: 3/4" vents in the roof of the recirculation sump entrance, the containment sump drain line from the recirculation sump, and small gaps around the sump entrance. These pathways have since been eliminated or the 1/4" requirement has been established.

The licensee is reporting the fact that since 1979 until the 1/4" requirement was established or the pathway was eliminated, the containment recirculation sump did not meet its design requirement.

The containment recirculation sump currently meets the 1/4" requirement. A condition report has been written to initiate investigation into this event and determine appropriate preventive actions.

This event was determined to be reportable at 0856 on September 5, 1997.

*** Update at 1905 on 09/10/97 by Randy Ptacek entered by Jolliffe ***

After further review of the above condition, the licensee concluded that the emergency core cooling (ECC) system was outside its design basis as a result of the 1/4" requirement not being met following the 1979 plant modifications. By not adequately covering the 1/4" particulate retention requirement, larger particles had the potential to enter the recirculation sump. The ECC System has not been analyzed for these larger particles nor is it within the design of the ECC System to handle these larger particles.

The licensee has concluded that this event is also reportable to the NRC in accordance with the requirements of 10CFR50.72(b)(1)(ii)(a) unanalyzed condition, and 10CF50.72(b)(2)(iii)(d)accident mitigation.

POWER REACTOR EVENT NUMBER: 32903
CONTAINMENT RECIRCULATION SUMP VENT HOLES HAVE BEEN FILLED WITH CONCRETE

As a result of questions posed by the NRC architect/engineer design inspection team, the licensee determined that the inlet venting requirement for the containment recirculation sumps was not properly maintained following modifications to the Unit 2 sump in 1996 and the Unit 1 sump in 1997 (both units have similar containments).

The containment recirculation sump venting requirement was established in 1979 as part of the original sump design to reduce the potential for air entrainment through the sump. The venting requirement was met through the addition of five 3/4-inch diameter holes drilled in the roof of the sump inlet. (The holes did not meet the 1/4-inch diameter requirement as reported in Event #32875.) When these holes were discovered during the Unit 2 1996 refueling outage and the Unit 1 1997 refueling outage, they were classified as abandoned equipment holes that exceeded the 1/4-inch particulate retention requirement for the sumps and they were filled with concrete.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

POWER REACTOR EVENT NUMBER: 32806
INSTRUMENTATION INDICATIONS USED TO DETERMINE WHEN REFUELING WATER STORAGE TANK TO CONTAINMENT SWITCHOVER IS REQUIRED MAY NOT HAVE BEEN CORRECT TO PREVENT VORTEXING IN THE CONTAINMENT RECIRCULATION SUMP.

During the evaluation of a proposed procedure change that affects switchover from the refueling water storage tank (RWST) to the containment sump during a loss-of-coolant accident (LOCA), it was determined that the instrumentation indications used to determine when the switchover is required may not have been correct to prevent vortexing in the containment recirculation sump.

To address this situation, procedures associated with the switchover (on both units) have been conservatively changed to accommodate the related instrument inaccuracies. These changes assure adequate RWST water is in containment before switchover to eliminate concerns that vortexing would occur in the containment sump after switchover.

The problem is that the RWST water level indicators are connected to the suction line that goes to the residual heat removal (RHR) pumps. Due to the flow in these lines, the indicated water level at which the switchover would be initiated would be less than the actual water level of the RWST (the licensee would be putting less water into the containment than expected). Also, the licensee said that they had some inaccuracies associated with their containment sump instrumentation. The licensee adjusted the containment sump indication to assure that they have an adequate volume in the containment to prevent vortexing. The licensee relies upon two indications for switchover; RWST water level and containment water level.

POWER REACTOR EVENT NUMBER: 32904
SINGLE FAILURE DURING RECIRC SUMP SWITCHOVER COULD BE UNANALYZED CONDITION

As a result of questions posed by the NRC architect/engineer design inspection team, the licensee determined that the possibility of a single failure during an accident while performing switchover of the emergency core cooling system pumps from the refueling water storage tank (RWST) suction to the recirculation sump suction could have resulted in the plant being in an unanalyzed condition. This condition is outside the plant design basis, and it potentially could have prevented the fulfillment of a safety function of structures or systems.

The plant emergency operating procedures (EOPs) as currently written require that the west residual heat removal (RHR) pump be the first pump switched from the RWST suction to the recirc sump suction. Once this is accomplished, the centrifugal charging (CC) pumps' suctions and the safety injection (SI) pumps' suctions are then swapped from the RWST supply to the discharge of the west RHR pump. If the west RHR pump were to fail at this point when all CC and SI pumps were being supplied from its discharge, prior to the east RHR pump suction being transferred from the RWST to the recirc sump, all CC and SI pumps could also fail due to the loss of suction flow. This would result in the loss of all high and medium head injection with only the flow from the east RHR pump available for injection into the reactor coolant system. The licensee is currently reviewing the EOPs to determine an alternate switchover sequence that would eliminate the condition as described above.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

POWER REACTOR EVENT NUMBER: 32939
INSTALLED PLANT MODIFICATION INTRODUCED THE POSSIBILITY OF A SINGLE FAILURE WHICH COULD RESULT IN THE LOSS OF BOTH TRAINS OF THE ESF VENTILATION SYSTEM.

At 1620 on 09/16/97, the licensee determined that a plant modification installed between December 1996 and August 1997 introduced the possibility of a single failure which could result in the loss of both trains of the engineered safety features (ESF) ventilation system if the 85-psi air header was to be lost. Prior to the installation of the plant modification, the ESF ventilation system charcoal inlet and bypass dampers both utilized a 20-psi air header and were positioned such that the charcoal bypass dampers were normally open and would fail closed; and the charcoal inlet dampers were normally closed and would fail open. The plant modification installed new bypass dampers which required higher air pressure to operate and were, therefore, transferred to the 85-psi header. If the 85-psi air header was lost, it would result in the repositioning of the normally open bypass dampers without the opening of the charcoal inlet dampers on both trains. This would result in dead heading of the filter train fans and loss of cooling to emergency core cooling system (ECCS) equipment.

POWER REACTOR EVENT NUMBER: 32988
NON-SAFETY RELATED AIR HEADERS LACK OVERPRESSURE PROTECTION

During an architectural engineering inspection a question was raised regarding the lack of overpressure protection on the 20, 50 and 85 psig control air headers. The specific concern is the potential for common mode failure of both trains of safety related equipment served by the air headers. The overpressure condition is postulated to be caused by regulator failure.

Although system reviews have found no component failure mode which would result in the devices being incapable of going to their fail-safe position, a design change package has been prepared to provide overpressure protection on the 20, 50 and 85 psig headers.

POWER REACTOR EVENT NUMBER: 32914
LICENSEE IDENTIFIED THAT BOTH UNITS HAD OPERATED THEIR RHR SYSTEM CONTRARY TO THE DESCRIPTION IN THE FSAR.

At 1615 EDT, with Units 1 and 2 shutdown in mode 5, it was determined that both units have operated contrary to the design basis for the residual heat removal (RHR) system as described in the Final Safety Analysis report (FSAR). FSAR Chapter 9, Section 9.3, describes the interlocks associated with the residual heat removal (RHR) suction valves from the reactor coolant system (RCS). The suction line valves are interlocked through separate channels of the RCS system pressure signals to provide automatic closure of both valves whenever RCS pressure exceeds RHR design pressure. The FSAR states that the interlock may be defeated when the RCS is open to atmosphere. However, for a number of years this interlock has been procedurally defeated on both units to prevent inadvertent closure and loss of RHR suction during shutdown cooling operation by opening the valves and racking out their breakers in mode 4.

The overpressure protection afforded by the automatic closure function described in the FSAR was defeated without a safety evaluation being performed. This loss of automatic closure function represents an unanalyzed condition and is, therefore, reportable.

Plans are to degas, depressurize, and open the RCS on both units to atmosphere. Degas will start on Unit 1, and when completed, the unit will proceed to depressurize while Unit 2 starts degas procedures. When the RCS is open to atmosphere on both units, the plant will be in compliance with the FSAR.

This condition was identified by the licensee during an ongoing NRC architect/engineer inspection.

*** Update at 2130 EDT on 9/13/97 from Robert Blyth to S. Sandin ***

The licensee has completed its safety evaluation for mode 5 operation and concluded that there was no unreviewed safety question or change of operation as described in the FSAR. Consequently, degas of Unit 1 has been terminated, and neither unit will be vented to atmosphere.

POWER REACTOR EVENT NUMBER: 32921
THE LICENSEE IDENTIFIED THAT BOTH RHR PUMPS HAD BEEN OPERATED WHEN THE RCS WAS DEPRESSURIZED, WHICH IS CONTRARY TO THE DESCRIPTION IN THE FSAR.

Chapter 9 of the Final Safety Analysis Report (FSAR) states: 'Only one residual heat removal (RHR) pump will be operated when the reactor coolant system is open to atmosphere to prevent damaging both pumps in the unlikely event that suction should be lost.' Operating procedures for the RHR system do not prevent operation of both RHR pumps when the reactor coolant system (RCS) is open to atmosphere, and in the past, both RHR pumps have been run when the RCS was vented to atmosphere.

Plant operating procedures are being reviewed to determine the impact. Procedure changes will be implemented as necessary to address the FSAR requirement. A condition report has been initiated to investigate and determine appropriate preventative actions.

POWER REACTOR EVENT NUMBER: 32948
IT WAS DETERMINED THAT FIBROUS MATERIAL IS PRESENT IN BOTH UNIT 1 AND UNIT 2 CONTAINMENT IN ENOUGH QUANTITY TO POTENTIALLY CAUSE EXCESSIVE BLOCKAGE OF THE CONTAINMENT RECIRCULATION SUMP SCREEN DURING THE RECIRCULATION PHASE OF A LOSS OF COOLANT ACCIDENT.

In 1985, 1986, and 1995 "Fiberfrax" refractory insulation materials in bulk, blanket or board form were used as damming material when installing fire stops in cable trays in both containments. The specification governing installation of the fire stops did not require removal of the material, only stating that it should be removed "if necessary." The material was not removed. The material is present in 12 cable trays in Unit 1 and 15 cable trays in Unit 2.

When the Fiberfrax is exposed to water or steam/water environment it could potentially break into small pieces, which could be transported to the recirculation sump by the water flow in containment during a loss of coolant accident. Once it reaches the recirculation sump it has the potential to clog the screens in excess of the design value. Excessive screen blockage could result in ECCS inoperability during the recirculation mode.

The Fiberfrax material is currently being removed from the containments, and removal will be completed prior to restart of the units. The possibility that the licensee's work control process allowed unencapsulated fibrous material to be installed in other locations inside containment is being investigated.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

POWER REACTOR EVENT NUMBER: 32740
UNITS 1 & 2 OPERATED OUTSIDE THE DESIGN BASIS FOR SERVICE WATER INLET TEMP

As a result of questions posed by members of the ongoing NRC design inspection team, the licensee has determined that Units 1 & 2 have operated outside the plant design basis for service water inlet temperature.

The Updated Final Safety Analysis Report (UFSAR), Table 9.5-3, lists service water inlet temperature design value as 76°F. This value is used as input to analyses such as containment peak pressure and control room habitability. Although engineering analyses were performed in 1988 raising the temperature to 87.5°F as listed in the plant Technical Specifications, a 10CFR50.59 safety evaluation was never performed, nor was the UFSAR properly revised.

Plant service water inlet temperature is the same as Lake Michigan water temperature. A review of historical data indicates that during July and August of any year, Lake Michigan water temperature is likely to exceed the 76°F value. Specific data for 1997 shows that Lake Michigan water temperature, and therefore plant service water inlet temperature, was greater than 76°F on July 17, July 18, and August 4, 1997. All plant systems which utilize service water as a cooling medium have been determined to be operable. A 10CFR50.59 safety evaluation will be performed and appropriate changes will be incorporated into the UFSAR.

This report is intended to cover any temperature exclusions above 76°F and below the 87.5°F value listed in the plant Technical Specifications that may occur prior to the completion of the 10CFR50.59 safety evaluation.

POWER REACTOR EVENT NUMBER: 32822
DISCOVERY THAT A NORMAL OPERATING PROCEDURE ALLOWED PLANT OPERATION WITH COMPONENT COOLING WATER HEAT EXCHANGER OUTLET TEMPERATURES GREATER THAN THE DESIGN LIMIT SPECIFIED IN THE FINAL SAFETY ANALYSIS REPORT

During the ongoing NRC architect/engineer design inspection, a question was asked relative to a statement used in the normal operating procedure for the component cooling water (CCW) system. The statement allows for a heat exchanger outlet temperature for CCW to reach 120°F for a period of 3 hours during normal cooldown on the residual heat removal system. Investigation revealed that this statement was in the original issue of the procedure in 1976. However, no 10 CFR 50.59 unreviewed safety evaluation determination documentation could be found to support this design parameter.

The licensee's Final Safety Analysis Report (FSAR) states that the CCW heat exchanger outlet design temperature is 95°F. Based on the FSAR requiring the 95°F outlet temperature and the lack of an unreviewed safety question determination to justify operation exceeding 95°F, the units were in a condition that allowed operation outside the design basis because the procedure allowed operation up to 120°F for a period of 3 hours during normal cooldown on the residual heat removal system. The units are not currently in a Technical Specification limiting condition for operation as a result of this issue.

Procedure changes have been made to remove the inappropriate statement. A condition report has also been written to initiate an investigation into this event and determine appropriate preventive actions.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

POWER REACTOR EVENT NUMBER: 32823
FAILURE OF A SAFETY REVIEW TO ADDRESS FINAL SAFETY ANALYSIS ATTRIBUTES ON ASSOCIATED COMPONENT COOLING WATER COOLING REQUIREMENTS

During the ongoing NRC architect/engineer design inspection, a question was asked relative to dual train component cooling water (CCW) system outages. During dual train CCW outages, CCW cooling is supplied to the spent fuel pool (SFP) heat exchanger only from the opposite unit. If that unit has a loss of coolant accident (LOCA), CCW to the SFP heat exchanger will isolate. Final Safety Analysis Report (FSAR) Table 9.5-2, footnote 3, indicates that the SFP heat exchanger is assumed to be on the non-accident unit.

The licensee reported the following inspection questions:

1) Does a dual train CCW outage represent a condition outside the plant design basis?

2) Was this reviewed as part of the process of allowing a dual train CCW outage?

Based on a review of FSAR Table 9.5-2, it was concluded that footnote 3 was established to clarify why no values for SFP heat exchanger flow for the unit undergoing the LOCA are listed in the table. Footnote 3 reflects normal SFP cooling system design and operation.

A review was performed of the safety evaluation performed for the Unit 2 full core offload with one train of spent fuel cooling. This safety review covered the Unit 2 refueling outage schedule which included a dual train CCW outage.

Footnote 3 of Table 9.5-2 represents the normal design of the SFP cooling system, that is, the SFP cooling system is designed to remove the heat generated by stored spent fuel elements in the [SFP]. The system incorporates two separate trains.

The safety review for the Unit 2 full core offload with one train of spent fuel cooling addressed the FSAR section 9.4 attribute of the SFP cooling dealing with time to boil events and bulk pool temperature requirements; however, the safety review failed to address FSAR section 9.5 attributes associated CCW cooling requirements as given in Table 9.5-2.

This issue impacts both units. However, the units are not currently in a Technical Specification limiting condition for operation as a result of this issue.

POWER REACTOR EVENT NUMBER: 32824
FAILURE TO PERFORM A 10 CFR 50.59 EVALUATION FOR A PROCEDURE CHANGE INVOLVING COMPONENT COOLING WATER HEAT EXCHANGER OUTLET TEMPERATURE LIMITS

During the ongoing NRC architect/engineer design inspection, a question was asked relative to the fact that during the last Unit 2 refueling outage, an administrative limit of 90°F was placed on the component cooling water (CCW) system. The thermal analysis indicated that a maximum CCW temperature of 90°F would eliminate all margin associated with the spent fuel pool (SFP) design assuming a design flow of 3,000 gpm.

Attachment 2 (continued)
NRC Daily Event Reports on D C Cook Design Inspection Findings

The following inspection question was asked: Since a change in CCW temperature was required to meet the Final Safety Analysis Report (FSAR) value of 160°F for the SFP, was a 10 CFR 50.59 unreviewed safety evaluation performed?

The licensee reviewed the change to the procedure to limit CCW temperature to 90°F. The licensee considered this change to be an administrative change only to lower the allowable temperature to the SFP cooling heat exchanger. A 10 CFR 50.59 evaluation was not performed because it was not recognized that the 95°F requirement was essentially being changed.

Without the completion of an unreviewed safety question determination, the plant was in a condition outside the design basis. The units are not currently in a technical specification limiting condition for operation as a result of this issue.

A condition report has been written to initiate actions to investigate this event and provide preventive actions. The 90°F limit is no longer in the operating procedures.

POWER REACTOR EVENT NUMBER: 32839
AVAILABLE WATER VOLUME IN RWST NOT ADEQUATE IN MODES 5 AND 6

During the ongoing NRC architect/engineer design inspection, NRC inspectors asked a question about the reactor coolant makeup required after a 10CFR50, Appendix R fire. To respond to the question, the licensee reviewed two associated design calculations. The more restrictive calculation was determined to be the calculation of record to meet the requirement. This calculation requires 87,000 gallons of water to be available in the refueling water storage tank (RWST). The value of 87,000 gallons was approved on 02/20/90. During modes 1 through 4, plant procedures adequately ensure that this requirement is met. During modes 5 and 6, plant procedures are not adequate to ensure that this requirement is met.

The plant has been in modes 5 and 6 many times since this requirement became effective on 02/20/90. Based on this, the plant has been in an unanalyzed condition several times since 02/20/90.

Currently both units are in mode 1. The licensee is reviewing plant operating procedures to determine impact and will implement procedure chances as needed prior to either unit entering modes 5 or 6. The licensee is continuing to evaluate the subject calculations and plans to submit a licensee event report to the NRC on this subject.

POWER REACTOR EVENT NUMBER: 32843
LAKE MICHIGAN TEMPERATURE EXCEEDED PLANT DESIGN BASIS LIMIT IN AUGUST 1988

As a result of questions posed by members of the ongoing NRC architect/engineer design inspection team, the licensee has determined that the water temperature of Lake Michigan, the plant's ultimate heat sink, exceeded the plant design basis lake temperature limit of 76°F for 22 days during August 1988.

The control room is normally cooled by an air conditioning system which utilizes non-safety related chillers. The safety related portion of the control room air conditioning system utilizes water from Lake Michigan as the cooling medium. This water would be supplied directly to the cooling coils following manual realignment. At an average lake temperature of 81°F that existed during the 22 day period in August 1988, the temperature inside the control room could have reached 110.4°F had the non-safety related chillers not functioned. At a temperature of 110.4°F, the lifetime of some instrumentation inside the control room, the solid state protection system, and the nuclear instrumentation, is estimated to be at 150 hours or 6.25 days. The impact of this shortened instrument life span on plant operation had not been evaluated.

At the time of this event, the plant Technical Specifications allowed continuous operation with control room temperatures up to 120°F. The Technical Specifications have since been revised such that continued operation with control room temperatures in excess of 95°F is not permitted.

Operation of the plant during the time period when lake temperature exceeded the design basis limit, without analysis indicating acceptable control room cooling could be maintained above this temperature limit, and without procedures to alert personnel of the situation, is considered as operation in an unanalyzed condition. The instrumentation was not adversely impacted by the high lake temperatures as the non-safety related chillers continued to function and maintain acceptable control room temperatures.

POWER REACTOR EVENT NUMBER: 32915
OVERPRESSURE PROTECTION OF THE COMPONENT COOLING WATER SYSTEM PIPING NOT IN ACCORDANCE WITH THE ANSI CODE REQUIREMENTS

Chapter 9.5 of the FSAR states: 'The relief valve on the component [cooling water] surge tank is sized to relieve the maximum flow rate of water that would enter the surge tank following a rupture of a reactor coolant thermal barrier cooling coil. The set pressure assures that the design pressure of the component cooling system is not exceeded.'

The piping design code at the Cook plant is B31.1. B31.1 states that an intercepting stop valve cannot be located between the source of pressure and the pressure relief device credited for protecting the pipe. In this instance, the pressure source is the ruptured thermal barrier; the pressure relief device is a safety relief valve on the surge tank. Contrary to the code requirement, there are manual valves maintained open between the two. These valves were not controlled in accordance with or exempted from B31.1.

An evaluation is being performed to determine the most effective method of establishing and maintaining the code requirement. A condition report has been written to initiate an investigation into this event and determine the appropriate preventative actions."

This condition was identified in response to an ongoing NRC architect/engineer design inspection.

UNION OF
CONCERNED
SCIENTISTS

DOCKETED
USNRC

'99 FEB 16 P4:39

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

January 12, 1998

Mr. L. Joseph Callan
Executive Director for Operations
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: ADDENDUM TO PETITION PURSUANT TO 10 CFR 2.206, DONALD C. COOK NUCLEAR PLANTS UNITS 1 AND 2, DOCKET NOS. 50-315 AND 50-316

Dear Mr. Callan:

The Union of Concerned Scientists submits this addendum to the petition pursuant to 10 CFR 2.206 we submitted on October 9, 1997 regarding Donald C. Cook Units 1 and 2. This addendum was requested by Ms. Elinor Adensam of your staff following my oral presentation this morning of our safety concerns. Enclosed is the prepared statement which I read during that presentation.

Sincerely,

David A. Lochbaum
Nuclear Safety Engineer

cc: Chairman Shirley Ann Jackson
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Mr. A. B. Beach, Regional Administrator
United States Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Honorable Fred Upton
United States House of Representatives
Washington, DC 20515-2206

Honorable Spencer Abraham
United States Senate
Washington, DC 20510-2203

Honorable Carl Levin
United States Senate
Washington, DC 20510-2202

Washington Office: 1616 P Street NW Suite 310 • Washington DC 20036-1495 • 202-332-0900 • FAX: 202-332-0905
Cambridge Headquarters: Two Brattle Square • Cambridge MA 02238-9105 • 617-547-5552 • FAX: 617-864-9405
California Office: 2397 Shattuck Avenue Suite 203 • Berkeley CA 94704-1567 • 510-843-1872 • FAX: 510-843-3785

This is a public meeting, not the public hearing that we requested when we submitted our 2.206 petition over three months ago. There have not been many public hearings held for 2.206 petitions. In fact, it is my understanding that I have attended every 2.206 public hearing ever held. One. That public hearing was held on the Millstone petition filed by We The People and Mr. George Galatis. Mr. Galatis was featured on the cover of TIME in March 1996. The first, and only, public hearing for a 2.206 petition was held the following month. Coincidence? I honestly doubt it. But I will get into statistics and how they are used by the NRC later.

You agreed to this meeting to see if I have "new" information about D C Cook. Before I present my information, I will briefly discuss some "old" information. You have heard this information before, but maybe not yet in 1998 – the 2.206 petition process is seriously and fundamentally broken. It isn't bent, it's broke.

You revised the 2.206 process 3 or 4 years ago and think it is fixed. The process was indeed changed, but it is not fixed. The old 2.206 process was broken. The new 2.206 process is broken. It needs to be fixed, or eliminated.

I suspect that the NRC's difficulty in stemming declining performance by its licensees offers a close parallel with the history of the 2.206 petition process. Your inspectors detect a performance problem at a plant. Its owner implements corrective actions. You conduct a followup inspection. If you find that things are the same, you correctly assume that the problem has not been fixed. If you find that things are different, you assume that the problem has been fixed. However, things can be different but still not fixed. That's your trouble with the 2.206 process and may have been the trouble you had preventing performance declines during the early stages of Salem and Millstone.

UCS submitted its 2.206 petition on October 9, 1997. We asked for two things: specific actions regarding D C Cook and a public hearing to present our concerns. To date, UCS has received one piece of paper from you concerning our petition – a letter dated December 9, 1997, acknowledging its receipt. All of the few telephone discussions we've had regarding the petition have been originated by me.

But enough on the 2.206 process. Today's meeting is for UCS to convey its concerns regarding D C Cook to you. Normally, I distribute copies of the slides or handouts to accompany my oral remarks. Since I thought, in good faith, that we would be granted a public hearing and assumed that I'd have at least 10 days to prepare for it, and since that did not happen, I am unable to provide any written documentation to you.

There are six concerns that I would like to discuss with you today.

My first concern involves D C Cook's ice condenser containment. The NRC Inspector General's office was informed last summer about alleged problems in the configuration and testing of the ice condenser at Watts Bar. Problems with the bay doors and components of the ice baskets were specifically identified. The allegations also suggested that many of the problems were generic and therefore affected the other ice condenser plants, including D C Cook. Finally, it was alleged that the problems were known, but not properly reported, by the Watts Bar licensee, the D C Cook licensee, the McQuire licensee, and even Westinghouse.

I refer you to Mr. George Mulley in the IG's office for the technical issues. I don't want to compromise IG's investigation, but these allegations exist and they may affect D C Cook. You recently issued an amendment to D C Cook's technical specifications involving the amount of ice in the ice condenser. The ice condenser licensing bases were changed, albeit to a limited extent. It provided another opportunity for the licensee to identify and report any ice condenser problems. I did not see any such report. Are the Watts Bar ice condenser problems valid? Do they apply to D C Cook? I can't answer that – at this time. Can you?

My second concern involves the licensee's 50.59 safety evaluation process. From the material I've reviewed, it appeared that you felt the licensee's 50.59 safety evaluation process needed improvements. I understand that the licensee made changes to its process. I am concerned that it is not evident that the licensee made any attempt to determine if safety evaluations prepared under the old process led to inappropriate conclusions. In other words, did the bad process cause bad products?

Before joining UCS in 1996, I was a consultant on a UFSAR vertical slice project for Salem Unit 2. We looked at <u>every</u> safety evaluation written for <u>every</u> modification to the systems we examined. Prior to that assignment, I was a consultant on the power update project for Susquehanna. Although that licensee did not have a suspect 50.59 process, the effect of increasing the plant's licensed power level might have invalidated the conclusions from prior safety evaluations. Therefore, we reviewed the summary for <u>every</u> safety evaluation written. Prior to that assignment, I was a consultant on the Browns Ferry Restart Project. TVA did have a configuration management problem. We reviewed <u>every</u> safety evaluation written for <u>every</u> modification to the systems we examined.

So, based on industry experience and common sense, I expected to see at least some screening of safety evaluations written at D C Cook using the bad process. Has an assessment of D C Cook's safety evaluations been performed? If not, could "bad" safety evaluations prepared using the "bad" 50.59 process mean that unidentified safety problems remain at D C Cook?

My third concern involves engineering calculations. From the material I've reviewed, it appears that the quality of the licensee's calculations was suspect. In fact, the licensee's response to the confirmatory action letter (CAL) dated December 2, 1997, stated that a root cause for its problems was that "Some analyses were found to contain errors and incorrect assumptions." The licensee said a peer review process was used to spot check its calculations. According to the licensee's response, a total of 191 calculations were peer reviewed. Sounds like a broad review. But it's not, for the following reason.

171 calculations were reviewed to resolve the concerns you raised during the design inspection. The remaining 20 calculations covered the auxiliary feedwater, component cooling water, chemical volume and control, containment spray, essential service water, residual heat removal, and electrical distribution systems. 20 calculations for 7 safety systems. That's an average of fewer than 3 calculations reviewed per safety system. Even given this tiny sample, the licensee reported that "some administrative and minor technical concerns were identified."

Is the NRC satisfied that a review of merely 20 calculations is an adequate extent of condition assessment? If so, why?

My fourth concern also involves engineering calculations. Between the time we submitted our petition and the time the licensee responded to the CAL, I received allegations involving net positive suction head (NPSH) calculations performed for D C Cook. The individual making the allegations was at D C Cook and told me there were problems with more than one NPSH calculation. The alleged problems involved both "missing" and inaccurate calculations. I do not know which pumps were affected, but it should not be too difficult for you to check. I am unable to check myself since these documents are not publicly available. Do the safety-related pumps at D C Cook have adequate NPSH as shown by quality calculations?

My fifth concern involves the credibility of the licensee's response to your CAL. By letter dated February 6, 1997, the licensee submitted, under oath, its response to the NRC's 50.54(f) request dated October 9, 1996. I think it is fair to state that the licensee, in that response, told you that there were no major problems with the two safety systems you examined in the subsequent design inspection. Each of these safety systems had been the subject of a design bases document recently issued by the licensee. Essentially, the licensee gave both of these safety systems a clean bill of health. Your subsequent design inspection clearly showed otherwise. Both units have been shut down for over three months to fix the problems you identified in the allegedly "clean" systems.

Since the shutdown, the licensee has expended considerable effort fixing the many problems you identified. Numerous physical plant changes were necessary. However, the licensee has expended less effort examining whether the programmatic problems you found affected other systems as well. The licensee was unable to identify the problems in the two systems you examined during a thorough design bases documentation program. It appears that the licensee applied less effort, per system, on the recent extent of condition assessment than it applied during the design bases document process. Since the larger effort failed, can you be sure that the smaller effort succeeded?

My sixth concern involves the NRC's own inspection process. You came in, looked at two safety systems, and found enough problems to force both units to shut down. The licensee maintains that these problems were confined to these two systems and everything else is well. Sound familiar? In 1996, you examined 4 systems at Maine Yankee and documented over 70 pages of problems. That licensee claimed the problems were limited to just those systems. Last year, you examined 2 systems at Vermont Yankee and found a serious problem affecting 1 system and lesser problems affecting the other. That licensee claimed the problems were limited to just those systems. If these licensees are correct, then you are the best regulator on the planet. You consistently find the needles in the haystacks. You find the only significant system problems that exist at the plants.

Were these licensees correct? I don't know. More importantly, you don't know either. You've never expanded the scope for system sampling inspections. If you had examined another system or two, then you'd really know whether you found the only problems or not.

You make sure that the licensees fix the problems you find in the few systems. That obviously needs to be done. But much more needs to be done. The true purpose of the your inspection of sample systems is <u>not</u> to ensure the operability of these few systems. Your inspections are intended to assess the licensee's programs and controls for maintaining all safety systems. Your findings tell you something about the material condition of the plant, but they also provide you information on the licensee's general safety management ability. Theoretically, you should not find anything during an inspection. Thus, any finding actually represents <u>two</u> problems – a nonconforming condition as well as a failure of the licensee's Quality Assurance (QA) process. Too often, you allow licensees to simply fix half of the problem – the nonconforming

condition. For example, when you find a broken widget, you make sure that the licensee changes the widget. You also need to find out why the licensee did not identify the broken widget and if they have any other broken widgets. The licensee's programmatic failures must be fixed. Otherwise, problems in other systems will remain undetected and future problems may be introduced.

What would it take for you to expand the sample size? This should not be a rhetorical question. You should develop and issue clearly defined criteria on when you will require additional system assessments based on findings from your system inspections.

These are my concerns.

I think UCS asked for very reasonable actions in our petition. The significant problems you found raise valid questions about the other safety systems at D C Cook. To date, I do not think those questions have been adequately answered. It is clearly the licensee's burden to answer these questions. It is your burden not to permit D C Cook to restart until these questions are answered and the answers indicate the plant will be operated safely.

To be perfectly candid, I never expected our petition to be granted. The NRC's record is such that a public petition has very little chance of being granted. My fallback position is to monitor daily event reports, LERs, and inspection reports after the plants restart. When I see a significant problem reported that might have been identified and corrected before restart had the NRC granted our petition, you can be sure I'll let you know.