February 20, 1998

Mr. E. E. Fitzpatrick
Executive Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: NOTICE OF VIOLATION AND NRC INSPECTION REPORT NO. 50-315/97025(DRP); 50-316/97025(DRP)

Dear Mr. Fitzpatrick:

On January 31, 1998, the NRC completed an inspection at your D. C. Cook 1 and 2 reactor facilities. The enclosed report presents the results of that inspection. During the five-week period covered by this inspection report, the inspectors observed that the plant was operated in a safe manner, and work procedures were properly followed.

Based on the results of this inspection, two violations of NRC requirements were identified by the NRC. The violations are cited in the enclosed Notice of Violation (Notice), and the circumstances surrounding them are described in detail in the subject inspection report. The violations both involved a failure to make a timely report for two occurrences required to be reported under 10 CFR 50.72 and 50.73. The violations are of concern because of the importance of the timely notification of reportable events.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be placed in the NRC Public Document Room.

Sincerely,

Original signed by
Marc L. Dapas for

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket Nos.: 50-315, 50-316
License Nos.: DPR-58, DPR-74

Enclosures:
1. Notice of Violation
2. Inspection Report No. 50-315/97025(DRP); 50-316/97025(DRP)

cc w/encls:
John Sampson, Site Vice President
A. A. Blind, Vice President Nuclear Engineering
Douglas Cooper, Plant Manager
Richard Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality

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/s/Marc L. Dapas for

Geoffrey E. Grant, Director
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NOTICE OF VIOLATION

Indiana Michigan Power Company
Donald C. Cook Nuclear Power Plant

Docket No. 50-315; 50-316
License No. DPR-58; DPR-74

During an NRC inspection conducted on December 28, 1997, through January 31, 1998, two violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

1. 10 CFR 50.72(b)(2)(I) requires, in part, that the licensee report within four hours any event found while shutdown, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded or in an unanalyzed condition that significantly compromised plant safety.

   Contrary to the above, on January 11, 1998, 38 Solid State Protection System relay covers failed a pull test to determine seismic acceptability, resulting in an unanalyzed condition that significantly compromised plant safety. The licensee did not report this condition to the NRC until January 15, 1998, more than four hours after the event.

   This is a Severity Level IV violation (Supplement I).

2. 10 CFR 50.73 (a)(1) requires, in part, that the licensee shall submit a License Event Report (LER) for any event of the type described in this paragraph within 30 days after the discovery of the event. 10 CFR 50.73(a)(2)(ii)(B) describes, in part, any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant.

   Contrary to the above, on September 11, 1997, an NRC inspector identified fibrous material in the Unit 2 lower containment, a condition outside the design basis of the plant. LER 50-315/97024 was not submitted until October 17, 1997, 36 days after the date of discovery.

   This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Indiana Michigan Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Lisle, Illinois
this 20th day of February 1998

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-315, 50-316
License No:	DPR-58, DPR-74
Report No:	50-315/97025(DRP); 50-316/97025(DRP)
Licensee:	Indiana and Michigan Power 500 Circle Drive Buchanan, MI 49107-1395
Facility:	Donald C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	December 28, 1997, through January 31, 1998
Inspectors:	B. L. Bartlett, Senior Resident Inspector B. J. Fuller, Resident Inspector E. R. Schweibinz, Project Engineer
Approved by:	Bruce L. Burgess, Chief Reactor Projects Branch 6

EXECUTIVE SUMMARY

D. C. Cook Units 1 and 2
NRC Inspection Report No. 50-315/97025(DRP); 50-316/97025(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a five-week period of resident inspection and includes the followup to issues identified during previous inspection reports.

Operations

- The inspectors concluded that cold weather preparations had been properly implemented. The inspectors also concluded that it was prudent to procure a stand-by boiler in order to ensure that safety-related equipment and other plant spaces were not adversely affected by cold weather in the event the plant heating boiler was unable to operate (Section O1.2).

Maintenance

- The inspectors concluded that the work activities observed were performed in a quality manner with procedures present and in use. The high quality of the Instrumentation and Control (I&C) technician work on the Solid State Protection System (SSPS) relay repairs was especially noteworthy. An exception to this good performance was a personnel error by an I&C technician on another activity which resulted in an invalid reactor trip signal. In addition, the inspectors determined that the required report regarding the inadvertent trip signal was not going to be made to the NRC until after the inspectors questioned the lack of a report (Section M1.1).

- The questionable material condition of the SSPS master relay dust covers resulted in both trains of SSPS being declared inoperable for operability under seismic conditions. A violation was identified when the licensee failed to make a timely report to the NRC concerning an unanalyzed condition that significantly compromised plant safety. An Unresolved Item was opened to track the review of additional data to determine when licensee personnel were aware of the degraded condition of the SSPS relays (Section M2.1).

Engineering

- Performance testing of the containment hydrogen skimming system continued during this inspection report period. Computer modeling and engineering assessments were being performed in an effort to determine the as-found operability and to ensure the system would be returned to an operable condition. An unresolved item on the as-found condition of the hydrogen skimming system remained open. The inspectors concluded that the Licensee Event Report (LER) issued contained inappropriate statements (Section E1.1).

- During a review of selected LERs, the inspectors identified one event that was improperly retracted and another event that was not reported in a timely manner. A third LER had an insufficient basis for retraction until the licensee performed additional calculations in response to inspector questions. One violation for failure to submit a timely LER was issued (Section E8.1).

Plant Support

- No discrepancies were noted.

Report Details

Summary of Plant Status

Unit 1 remained in Mode 5, Cold Shutdown, during this inspection period. The unplanned outage was in response to NRC and licensee concerns with the operability of the containment recirculation sump and other engineering issues.

Unit 2 remained in Mode 5, Cold Shutdown, during this inspection period. The unplanned outage was in response to NRC and licensee concerns with the operability of the containment recirculation sump and other engineering issues.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using the referenced inspection procedure, the inspectors conducted frequent reviews of ongoing plant operations. The inspectors observed licensed operators closely monitoring the panels and maintaining appropriate awareness of plant conditions, although both units were in Mode 5 for an extended period. Specific events and noteworthy observations are detailed in the sections below.

O1.2 Cold Weather Preparation (Both Units)

a. Inspection Scope (71714)

The inspectors verified that the licensee had implemented the procedures necessary to prepare the plant for cold weather. The inspectors reviewed logs and Work Procedure R0014700, “Plant Winterization,” and conducted plant walkdowns.

b. Observations and Findings

With both units shutdown, the licensee had to rely on the single non-safety-related plant heating boiler (PHB) for facility heating. Licensee personnel obtained a commercial oil-fired steam boiler (referred to as the alternate heating boiler) as a backup source of heat to ensure availability of steam for plant heating requirements. The alternate heating boiler was maintained in hot standby, but isolated from the plant heating system, as a ready source of heating steam for the facility.

The alternate heating boiler was placed into service on January 15, 1998, and again on January 30, 1998, when the PHB was removed from service to repair system leaks. The alternate heating boiler maintained the facility at an acceptable temperature. The inspectors performed walk downs of the turbine and auxiliary buildings during the period when the alternate heating boiler was in service to verify the absence of cold weather problems.

During a walkdown of the auxiliary building, the inspectors identified an opening to the refueling water storage tank yard in the Unit 2 exterior pipeway near the Nos. 1 and 4 feed regulating valves. The opening appeared to be for outage related services and allowed a significant inflow of cold air, but it did not pose an immediate threat of cold weather damage to any equipment. The inspectors informed the operations shift manager, and the opening was sealed.

c. Conclusions

The inspectors concluded that cold weather preparations had been properly implemented. The inspectors also concluded that it was prudent to procure a stand-by boiler to ensure that safety-related equipment and other plant spaces were not adversely affected by cold weather in the event the plant heating boiler was unable to operate.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns (Both Units)

In addition to routine plant inspections, the inspectors used Inspection Procedure 71707 to walkdown selected portions of the essential service water system. No operability concerns were identified.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707 and 61726)

Portions of the following maintenance job orders, action requests, and surveillance activities were observed or reviewed by the inspectors:

- C043418, Perform airflow tests on containment hydrogen skimmer system 2-HV-CEQ-1
- C043419, Perform airflow tests on containment hydrogen skimmer system 2-HV-CEQ-2
- C043563, Test and repair relays on Unit 1 Train A and B Solid State Protection System (SSPS)
- C043565, Test and repair relays on Unit 2 Train A and B SSPS
- C043697, Inspect diverter valves in the Unit 1 West Essential Service Water pump strainer

- **1 I&C Head Procedure (IHP).4030.Surveillance Test Procedure (STP).410, Revision 5, Train “A” Reactor Protection System (RPS) and Engineered Safety Features (ESF) Reactor Trip Breaker and SSPS Automatic Trip/Actuation Logic Functional Test

b. Observations and Findings

Overall, the inspectors noted that work was performed in a quality manner with the work packages present and in active use. The high quality of the I&C technician work on the SSPS relay repairs was especially noteworthy. The technicians were noted to carefully review the procedure steps and equipment before performing each step and to closely observe each other to avoid errors. Additional detail concerning this job evolution is contained in Section M2.1.

During the performance of 4030.STP.410, an I&C technician inadvertently missed a step which resulted in a reactor trip signal. Licensee personnel performed a followup investigation and took appropriate corrective action. The failure to follow a ** procedure step-by-step, as required by plant procedures, was a violation of 10 CFR Part 50, Appendix B, Criterion 5, “Instructions, Procedures, and Drawings.” This non-repetitive, licensee-identified and corrected violation is being treated as a Non-Cited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy (50-316/97025-04).

On January 8, 1998, licensee personnel called the NRC operations center to report the inadvertent reactor protection system trip signal (reference NRC Event No. 33505). On January 21, 1998, licensee personnel called the operations center to retract Event No. 33505 based upon additional review of the event. On February 2, 1998, the inspectors informed the licensee that after an NRC review of the circumstances surrounding the event that the event was required to be reported to the NRC. Licensee personnel stated that an LER would be issued within the 30-day requirement of 10 CFR 50.73.

c. Conclusions

The inspectors concluded that the work activities observed were performed in a quality manner with procedures present and in use. The high quality of the I&C technician work on the SSPS relay repairs was especially noteworthy. An exception to this good performance was a personnel error by an I&C technician on another activity which resulted in an invalid reactor trip signal. In addition, the inspectors determined that the required report regarding the inadvertent trip signal was not going to be made to the NRC until after the inspectors questioned the lack of a report.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Degraded Covers on Solid State Protection System Master Relays (Both Units)

a. Inspection Scope (62707)

On January 8, 1998, the licensee identified degraded covers on some master relays in the SSPS. The inspectors observed the licensee's investigation and short-term corrective actions. Licensee documents reviewed included:

- **1 IHP.Special Procedure (SP).SSPSA, Revision 0, “Unit 1 Train “A”, SSPS Master Relay Replacement”
- **1 IHP.SP.SSPSB, Revision 0, “Unit 1 Train “B”, SSPS Master Relay Replacement”
- **2 IHP.SP.SSPSA, Revision 0, “Unit 2 Train “A”, SSPS Master Relay Replacement”
- CR 98-0069, Covers for SSPS master relays are damaged
- Engineering Technical Direction Memo (ENTDM) 98-011, “Wiring Direction for The Replacement of SSPS Master Relays”
- Letter from Westinghouse to American Electric Power (AEP-88-248), dated May 10, 1988, Evaluation of master relay cover removal

b. Observations and Findings

The SSPS master relays were installed with dust covers to keep dirt and dust from accumulating inside the relay. During a routine surveillance of the SSPS on January 8, 1998, an I&C technician noted that some SSPS master relay dust cover ears were damaged or broken off. Past practice during surveillance testing of this portion of the SSPS involved removal of the relay dust covers to allow manual actuation of the relay contacts. Removal of the cover required that retaining ears on the base of the relay be pried open to release the cover. Repeated prying of the ears resulted in some ears being damaged or torn off.

Condition Report (CR) 98-0069 was written to document the degraded covers. The prompt operability determination in the CR stated that normal operation of the relays was not affected. The operability determination also stated that a seismic event would likely dislodge the covers from the relay assembly. The operability determination concluded that due to the light weight of the covers, it was unlikely for a dislodged cover to spuriously actuate relays in the vicinity or to prevent relays from actuating. Based on the prompt operability determination, the SSPS was initially declared operable with the damaged relay covers. The licensee’s CR preliminary reportability review also determined that the degradation of the covers was not reportable under 10 CFR 50.72.

On January 11, 1998, the licensee performed a pull test to determine the seismic acceptability of the covers. A total of 38 covers (15 of 48 in Unit 1, and 23 of 48 in Unit 2) failed the pull test. The inspectors questioned the operability status of the SSPS. Management and I&C personnel had determined that the relays were inoperable; however, the relays were not logged as inoperable in the operations department equipment out-of-service logs.

After pull testing was completed, operations department personnel declared the SSPS operable pending engineering review of the test data. The inspectors determined that the job order used to test the relay covers included appropriate acceptance criteria and that the operations department should have declared the relays which did not meet the acceptance criteria inoperable. After the inspectors questioned operations management about the pull

test results, the licensee concluded that the prompt operability determination of the original condition report was inadequate. Both trains of SSPS in each unit were subsequently declared inoperable.

Based on the results of the pull testing, on January 15, 1998, the licensee determined that the degraded SSPS relay covers were an unanalyzed condition that significantly compromised plant safety, and a four-hour non-emergency report was made to the NRC. 10 CFR 50.72(b)(2)(i) required, in part, that the licensee report within four hours any event found while shutdown, that, had it been found while the reactor was in operation, would have resulted in the nuclear power plant being in an unanalyzed condition that significantly compromised plant safety. Because the pull test job order contained appropriate acceptance criteria, the inspectors determined that the delay from discovering the condition, on January 11, 1998, to reporting the condition, on January 15, 1998, constituted a violation of 10 CFR 50.72. (50-315/97025-01(DRP); 50-316/97025-01(DRP)).

Replacement of the unacceptable covers began on January 19, 1998. Extensive planning and coordination between the engineering, maintenance and operations departments were evident. The inspectors observed portions of the planning process and replacement of selected relays. The I&C technicians were observed to be careful, methodical, and thorough in the performance of the relay replacement and subsequent testing.

Preliminary data identified by the licensee showed that the SSPS relays were known to be degraded since at least May 10, 1988, and adequate corrective actions were not taken in a timely manner. In May 1988, the SSPS vendor recommended that the dust covers be replaced or epoxied in place to prevent plastic fragments from the cover from falling into the relay or another relay below it. The recommendation also stated that if the cover began backing off the relay during a seismic event, chattering of the normally closed contacts could be induced. The licensee did not follow the recommendations of the vendor to replace or epoxy the covers although the seismic operability of the relays was in doubt. Licensee and NRC personnel are continuing to identify and review documentation related to this issue. Pending the results of the additional review this will remain an Unresolved Item (50-315/97025-03(DRP); 50-316/97025-03(DRP)).

c. Conclusions

The questionable material condition of the SSPS master relay dust covers resulted in both trains of SSPS being declared inoperable for operability under seismic conditions. A violation was identified when the licensee failed to make a timely report to the NRC concerning an unanalyzed condition that significantly compromised plant safety. An Unresolved Item was opened to track the review of additional data to determine when licensee personnel were aware of the degraded condition of the SSPS relays.

III. Engineering

E1 Conduct of Engineering

E1.1 (Open) Unresolved Item (50-316/97024-03(DRP)) Inadvertently Plugged Hydrogen Skimmer Suction Line (Unit 2)

a. Inspection Scope (62707)

On November 26, 1997, licensee maintenance personnel identified a blockage of the Steam Generator (S/G) No. 3 enclosure Train B hydrogen skimmer suction piping. Subsequently, engineering personnel initiated an assessment of the as-found operability and began efforts to restore the system to an operable status. Licensee procedures and documentation reviewed included:

- Job Order C0043514, Investigate and repair motor operated damper 1-VMO-101 cycling at 45 degrees
- LER 50-316/97-009-00, Blockage of containment air recirculation inlet line
- CR 98-0033, During containment hydrogen skimmer tests on Unit 1, flow rates were found below the acceptance criteria.

b. Observations and Findings

As part of the corrective action for a blocked hydrogen skimmer line (referred to as the CEQ system) the licensee committed to perform flow testing of both trains in both Unit 1 and Unit 2. On January 4, 1998, the licensee tested the Unit 1 CEQ fans and identified that flows from various compartments in both trains were lower than required by the Updated Final Safety Analysis Report (UFSAR). Subsequently, licensee personnel determined that the low flow on the No. 1 CEQ system was due to a mis-positioned shutoff damper, 1-VMO-101.

Damper 1-VMO-101 was improperly re-installed during maintenance such that instead of going from full closed to full open on an actuation signal it was partially open at all times. In addition to reducing flow through the skimmer line, the mis-positioned damper also created an ice condenser bypass flow path.

Unresolved Item (50-316/97024-03(DRP)) was initially opened pending the results of a proposed surveillance test. During this inspection period, the licensee determined that flows through the safety system were less than stated in the UFSAR. However, the CEQ flows were discussed twice in the UFSAR. In Chapter 5, “Containment,” higher, normal values were listed and in Chapter 14, “Accident Analysis,” lower values were listed. The lower values reflected more realistic assumptions. The values identified during the surveillance test were between the two values listed in the UFSAR.

The licensee determined that while the flow was degraded, constituting a potentially non-conforming condition, a specific operability decision could yet be made without further analysis. The initial flow tests performed during pre-operational testing were performed

with the ice condenser empty of ice. This test condition eliminated any concerns with ice melt during testing. Additional flow tests and computer modeling were required because with the ice condenser filled with ice, test conditions identical to the preoperational tests could not be established.

c. Conclusions

Performance testing of the containment hydrogen skimming system continued during this inspection report period. Computer modeling and engineering assessments were being performed to determine the as-found operability and to ensure the system would be returned to an operable condition. An unresolved item on the as-found condition of the hydrogen skimming system remained open.

E8 Miscellaneous Engineering Issues

E8.1 Review of Selected LERs (Both Units)

a. Inspection Scope (92700)

The inspectors reviewed selected LERs to verify that the reports were timely, complete, and appropriate. The LERs reviewed were:

- LER 50-316/97003, Revision 0, Revision 1, and Revision 2
- LER 50-316/97004, Revision 0, and Revision 1
- LER 50-315/97024, Revision 0

b. Observations and Findings

b.1. (Open) LER 50-315/97024: Material Discovered in Containment Degrades Containment Recirculation Sump and Results in Condition Outside Design Basis

On September 11, 1997, with the unit shutdown, an NRC inspector identified a fibrous material, known as Fiberfrax, in the Unit 2 lower containment, a condition that was outside the design basis of the plant. Subsequent inspections by the licensee confirmed this finding and resulted in the identification of additional fibrous material. On September 17, 1997, the licensee reported this finding to the NRC in accordance with 10 CFR 50.72(b)(2)(i). However, the licensee did not send in the LER until October 17, 1997.

Licensee personnel submitted the LER 30 days after the date this issue was determined to be reportable instead of within 30 days of the date of discovery. 10 CFR 50.73 (a)(1) requires, in part, that the licensee shall submit a LER for any event of the type described in this paragraph within 30 days after the discovery of the event. 10 CFR 50.73(a)(2)(ii)(B) describes, in part, any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant. The failure to submit an LER to the NRC within 30 days of the date of discovery was a violation of 10 CFR 50.73. (50-315/97025-02(DRP); 50-316/97025-02(DRP)).

b.2. (Open) LER 50-316/97003-03: Performance of Dual Train Component Cooling Water (CCW) Outage During Unit 2 1996 Refueling Outage Resulted in Condition Outside The Plant's Design Basis

The licensee reported in Revisions 0, and 1 of the LER that during the Unit 2, 1996 refueling outage a dual train CCW train outage was planned and performed. However, certain sections of the UFSAR had not been properly considered and thus the plant had possibly operated outside the design basis. The licensee's failure to comply with the UFSAR had been identified by NRC inspectors and documented in Inspection Report No. 50-315/97201.

Subsequently in Revision 2, the licensee retracted the LER based upon the conclusion that sufficient controls were in place to ensure the plant remained within the design basis. The inspectors reviewed the licensee's basis for retracting the LER and concluded that the basis was insufficient. The licensee had taken credit for the ability of maintenance and operations department personnel to restore the CCW components within the spent fuel pool (SFP) calculated time to boil upon a postulated loss of cooling.

The inspectors determined that taking credit for manual action lacked a sufficient analyzed basis to support the conclusion that the CCW system could be restored in time for the SFP to remain below the bulk temperature limit. The inspectors informed licensee personnel of their conclusions and the licensee subsequently resubmitted LER 50-316/97003, as Revision 3.

10 CFR 50.73(a)(2)(ii), required that the licensee report to the NRC conditions outside the design basis. The inspectors concluded that the licensee's withdrawal of the LER in Revision 2 was not appropriate.

b.3. (Closed) LER 50-316/97004-01: Change to CCW Temperature Without Revision to UFSAR

On August 26, 1997, with Unit 2 at 100 percent rated thermal power, the licensee determined that the unit had operated outside its design basis during the Unit 2 1996 refueling outage. It was determined that this event was reportable under 10 CFR 50.73(a)(2)(ii) as a condition outside the design basis. During the refueling outage the licensee had administratively lowered the maximum allowable CCW temperature to 90°F. This had been done to support the thermal analysis of the SFP during the Unit 2, 1996 refueling outage. The UFSAR stated that the CCW system was capable of safely handling the required safety loads at temperatures up to 95°F.

Subsequently in Revision 1, the licensee withdrew the LER based upon a determination that the 90°F limit and associated administrative controls had been reviewed as required by 10 CFR 50.59, prior to the Unit 2 refueling outage, and that this condition did not represent an Unreviewed Safety Question.

The inspectors reviewed the LER and determined that the design basis of the CCW system included the capability of the CCW system to safely shutdown/maintain shutdown the unit at a maximum initial temperature of 95°F. The licensee's analysis for the Unit 2

refueling outage showed that the CCW temperature was required to be maintained lower than the UFSAR upper temperature limit in order for the SFP to be maintained within the design basis.

This issue was reported in Inspection Report No. 50-315/97021; 50-316/97201 as Unresolved Item 50-315/97201-23; 50-316/97201-23. The NRC Architect Engineering inspection team was concerned that the licensee's 10 CFR 50.59 evaluation failed to properly recognize the safety consequence that the CCW system could not perform its design function to remove the required SFP heat load at its maximum operating system temperature of 95°F, as stated in the UFSAR. Thus, the licensee was operating outside the design basis of the CCW system and the issue was reportable under 10 CFR 50.73(a)(2)(ii) as a condition outside the design basis.

Following discussions with NRC inspectors on the 90°F limit versus the 95°F design basis, the licensee reanalyzed the Unit 2 with more realistic assumptions. The new analysis demonstrated that the SFP would be maintained within the design basis at a CCW temperature of 95°F.

The inspectors concluded that the licensee's retraction of LER 50-316/97004 was appropriate; however, the supporting evidence was insufficient until NRC inspectors questioned the basis for the retraction. The issues identified during the review of the LER regarding the licensee's 10 CFR 50.59 evaluation of the CCW system will be tracked under Unresolved Item 50-315/97201-23; 50-316/97201-23; therefore, this LER is closed.

c. Conclusions

During a review of selected LERs the inspectors identified one event that was improperly retracted and another event that was not reported in a timely manner. A third LER had an insufficient basis for retraction until the licensee performed additional calculations in response to inspector questions. One violation for an untimely report was issued.

E8.2 (Closed) LER 50-316/97009: Blockage of containment air recirculation inlet line results in a condition outside the design bases. This issue is discussed in Inspection Report No. 50-315/97024; 50-316/97024 and above in Section E1.1 as Unresolved Item 50-316/97024-03(DRP). The LER did not present any new information, nor did it include any additional licensee commitments; therefore, this LER is closed. However, the inspectors identified several portions of the LER which were not fully justified and appeared to be contrary to the design bases.

- As part of the assessment of the safety significance, the LER stated, "An evaluation of the impact of the blockage has concluded that there would have been adequate flow through the steam generator (S/G) enclosure to preclude excessive accumulation of hydrogen." To support this assessment of the safety significance, the LER included a statement that hydrogen would not be generated inside the S/G compartments and that the hydrogen generated in the lower compartment would be well mixed, thus having little potential for significant amounts of hydrogen to accumulate in the S/G compartment. The LER also stated that the opposite train of the hydrogen skimmer system was available except for short periods of maintenance and would have been capable of providing the required flow.

The design basis of the hydrogen skimming system as stated in the UFSAR assumed that hydrogen would be generated inside the S/G compartments. Licensee personnel informed the inspectors that the assumptions contained in the UFSAR were overly conservative. By using more realistic assumptions and engineering judgement, the licensee's engineering staff determined that lower flows through the skimmer system would be adequate.

The inspectors considered that using different assumptions than those contained within the UFSAR was inappropriate for evaluating the safety significance of the lower hydrogen skimmer flow. The LER did not clearly identify that this evaluation was based on engineering judgement and the use of assumptions different from those contained in the design basis. Also, the planned flow testing of the two hydrogen skimmer trains had not been completed when the LER was issued. Thus, the capability of either train to meet the required flow had not been verified. These comments were provided to the appropriate licensee staff for consideration and corrective actions, as necessary.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls (71750)

During the resident inspection activities, routine observations were conducted in the areas of radiological protection and chemistry controls using Inspection Procedure 71750. No discrepancies were noted.

S1 Conduct of Security and Safeguards Activities (71750)

During normal resident inspection activities, routine observations were conducted in the areas of security and safeguards activities using Inspection Procedure 71750. No discrepancies were noted.

F1 Control of Fire Protection Activities (71750)

During normal resident inspection activities, routine observations were conducted in the area of fire protection activities using Inspection Procedure 71750. No discrepancies were noted.

X1 Exit Meeting

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on February 2, 1998. The licensee had additional comments on some of the findings presented. No proprietary information was identified by the licensee.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

#K. Baker, Manager, Production Engineering
#T. Beilman, Scheduling Superintendent
#A. Blind, Vice-President Engineering
#J. Boesch, Maintenance Superintendent
#D. Cooper, Plant Manager
#S. Delong, Management Information
#MB. Depuydt, Nuclear Licensing
#S. Farlow, Supervisor I&C Engineering
#M. Finissi, Supervisor, Electrical Systems
#R. Gillespie, Operations Superintendent
#D. Hafer, Manager, Plant Engineering
#D. Landot, Plant Performance
#D. Morey, Chemistry Superintendent
#D. Noble, Radiation Protection Superintendent
#R. Ptacek, Nuclear Licensing
#F. Pisarsky, Supervisor, Mechanical Component Engineering
#T. Postlewait, Manager, Design Engineering
#P. Russell, Supervisor, Plant Protection
#J. Sampson, Site Vice-President
#P. Schoepf, Supervisor, Safety-related Mechanical Systems
#R. Stevens, Nuclear Licensing
#J. Tyler, Manager, Plant Protection and Emergency Preparedness
#L. VanGinhoven, Materials Management
#A. Verteramo, Supervisor Reactor Engineering
#S. Wolf, Performance Assurance

USNRC

#B. Burgess, Branch Chief, Region III

#Denotes those present at the February 2, 1998, exit meeting.

INSPECTION PROCEDURES USED

IP 37551	On-site Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observation
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 92700	Onsite Review of LERs

ITEMS OPENED, CLOSED, AND DISCUSSED

ITEMS OPENED

50-315/97025-01 50-316/97025-01	VIO	Failure to report an event outside the design basis in a timely manner in accordance with 10 CFR 50.72
50-315/97025-02 50-316/97025-02	VIO	Failure to report an event outside the design basis in a timely manner in accordance with 10 CFR 50.73
50-315/97025-03 50-316/97025-03	URI	SSPS Relay Operability History
50-316/97025-04	NCV	Failure to follow procedures

ITEMS CLOSED

50-316/97004-01	LER	Change to CCW temperature without revision to UFSAR
50-316/97009-00	LER	Blockage of containment air recirculation inlet line results in a condition outside the design bases
50-315/97025-03	NCV	Failure to follow procedures

ITEMS UPDATED

50-316/97024-03	URI	Inadvertently Plugged Hydrogen Skimmer Suction Line

LIST OF ACRONYMS

AEP	American Electric Power
bcc	blind carbon copy
cc	carbon copy
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
DCC	Donald C. Cook
DRP	Division of Reactor Projects
DPR	Demonstration Power Reactor
EDT	Eastern Daylight Time
ENTDM	Engineering Technical Direction Memo
ESF	Engineered Safety Feature
ESW	Essential Service Water
I&C	Instrumentation and Control
IHP	I&C Head Procedure
IR	Inspection Report
JO	Job Order
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MI	Michigan
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulator
PHB	Plant Heating Boiler
PDR	Public Document Room
RPS	Reactor Protection System
SCFM	Standard Cubic Feet per Minute
SE	Safety Evaluation
SFP	Spent Fuel Pool
SM	Shift Manager
SP	Special Procedure
SRO	Senior Reactor Operator
SSPS	Solid State Protection System
STP	Surveillance Test Procedure
S/G	Steam Generator
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item