February 9, 1999

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: D. C. COOK INSPECTION REPORT 50-315/98027(DRP); 50-316/98027(DRP) AND NOTICE OF VIOLATION

Dear Mr. Powers:

On January 13, 1999, the NRC completed an inspection at your D. C. Cook Units 1 and 2 reactor facilities. The inspection was an examination of activities conducted under your license as they relate to compliance with the Commission rules and regulations and with the conditions of your license. Areas reviewed included Operations, Maintenance, Engineering, and Plant Support. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations of activities in progress. The enclosed report presents the results of that inspection.

The inspectors observed examples of prompt involvement by you and members of your management staff when safety significant issues were raised by plant personnel and NRC inspectors. Specifically, your decision to delay the scheduled B engineered safety feature train outage until completion of the Train A affirmation was prudent and conservative. In addition, the inspectors observed that Performance Assurance personnel were performing pro-active assessments and identifying safety significant issues.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occured. The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. The inspectors also observed several examples of operability evaluations that were of poor quality. Contributing to the poor quality of the operability evaluations was the lack of clear procedural guidance on compensatory measures given to the operations staff, and the lack of procedural compliance by the engineering staff concerning operability evaluations which resulted in a violation identified by the NRC. In addition, the inspectors observed that the plant material condition was degrading. As plant material condition items continue to be identified, the need for good quality operability evaluations will become even greater and we encourage you to continue your efforts to correct the operability evaluation process. We recognize that the increase in the backlog of equipment problems is due, in part, to the impact of some of your major improvement initiatives. However, it is important that the material condition of the plant is maintained sufficiently to ensure that the design function of systems is not affected.

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence is already adequately addressed on the docket in this inspection report. Therefore, you are not required to respond to this letter unless the description therein does not accurately reflect you corrective actions or your position. In that case, or if you choose to provide additional information, you should follow the instructions specified in the enclosed Notice.

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

Sincerely,

/s/ J. A. Grobe

John A. Grobe, Director
Division of Reactor Safety

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

Enclosures:
1. Notice of Violation
2. Inspection Report 50-315/98027(DRP); 50-316/98027(DRP)

cc w/encls:
J. Sampson, Site Vice President
M. Rencheck, Vice President, Nuclear Engineering
D. Cooper, Plant Manager
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

See Previous Concurrences

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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 from December 4, 1998, through January 13, 1999, one violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> Title 10 CFR Part 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” required, in part, that activities affecting quality shall be prescribed by procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures.
>
> Procedure 12 EHP [Engineering Head Procedure] 7030.OPR [Operability Reviews] .001, “Supporting Analyses for Operability Determinations,” Revision 0, an activity affecting quality at Step 6.2.7, required, in part, “Any compensatory or corrective actions must be clearly documented and the appropriate tracking mechanism put in place to ensure their completion.”
>
> Contrary to the above, on October 8, 1998, the licensee failed to accomplish an operability evaluation in accordance with 12 EHP 7030.OPR.001. Specifically, operability evaluation 91-18-ENSM 990107TWF failed to document and implement appropriate tracking mechanisms to ensure completion of compensatory measures identified in Section 9 of the operability evaluation.
>
> This is a Severity Level IV violation (Supplement I).

The NRC has concluded that information regarding the reason for the violation, the corrective actions taken and planned to correct the violation and prevent recurrence and the date when full compliance will be achieved is already adequately addressed on the docket in this inspection report. However, you are required to submit a written statement or explanation pursuant to 10 CFR 2.201 if the description therein does not accurately reflect your corrective actions or position. In that case, or if you choose to respond, clearly mark your response as a “Response to a Notice of Violation,” and send it 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 of Violation (Notice).

If you choose to respond, your response will be placed in the NRC Public Document Room (PDR). Therefore, to the extent possible, the response should not contain any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction.

If you contest this enforcement action, you should provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555-001.

In accordance with 10 CFR 19.11 you may be required to post this Notice within 2 working days.

Dated this 9th day of February 1999

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/98027(DRP); 50-316/98027(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 4, 1998, through January 13, 1999
Inspectors:	B. L. Bartlett, Senior Resident Inspector B. J. Fuller, Resident Inspector J. D. Maynen, Resident Inspector
Approved by:	A. Vegel, Chief Reactor Projects Branch 6

EXECUTIVE SUMMARY

D. C. Cook Units 1 and 2
NRC Inspection Report 50-315/98027(DRP); 50-316/98027(DRP)

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 7-week period of resident inspection and includes the follow-up to issues identified during previous inspection reports.

Operations

! Operability evaluations and procedures reviewed by NRC inspectors and licensee personnel showed that shift managers, shift technical advisors, and engineers failed to adequately review operability evaluations. Procedural guidance on compensatory measures and administrative controls for those compensatory measures were weak or non-existent. Specifically, plant administrative procedures addressing operability evaluations did not contain requirements or adequate administrative controls for compensatory measures. (Section O1.2)

! The licensee did not have a process for ensuring that the plant winter preparations were integrated across organizational boundaries. The lack of integration contributed to the inadequate preparation for cold weather, especially considering the lack of process heat in the buildings due to the current shutdown condition. Cold weather preparations were untimely and severe cold weather caused multiple challenges to the plant operators. (Section O1.3)

! The licensee identified that there was a large backlog of unreviewed condition reports. The licensee's corrective action program is undergoing major assessment and corrective actions. The improvement plan includes actions to address problems with the condition report review process. (Section O7.1)

! Licensee Performance Assurance personnel demonstrated a critical questioning attitude in effectively assessing the circumstances related to a broken residual heat removal system pipe bracket which resulted in the identification of an inadequate operability evaluation. (Section O1.2)

Maintenance

! Observed maintenance activities were performed effectively in accordance with documented work instructions. (Section M1.1)

! Based upon an assessment of licensee corrective action requests, condition reports, contingency action logs, and in plant observations, the inspectors concluded that the material condition of several safety-related systems had declined during the forced outage. The delay in addressing equipment problems was due, in part, to the licensee's focus on corrective actions to address programmatic issues that are a part of the licensee's restart plan. (Section M2.1)

Engineering

- **!** Engineering procedure guidance concerning the conduct of operability evaluations was not being followed. One violation for failing to follow a procedure was identified. (Section O1.2)

Plant Support

- **!** During normal resident inspection activities, routine observations were conducted in the area of security and safeguards, fire protection, and health physics activities. No discrepancies were noted.

Report Details

Summary of Plant Status

Unit 1 remained in Mode 5, Cold Shutdown, during this inspection period. Work on Unit 1 took precedence over Unit 2.

Unit 2 remained in Mode 5, Cold Shutdown, during this inspection period. The restart schedule for Unit 2 was not yet complete.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using the referenced inspection procedure, the inspectors conducted frequent reviews of control room and in-plant operation of equipment during the extended outage of both reactor units. The inspectors found that, overall, the plant was operated in a safe manner and in accordance with procedures. However, based on issues identified by NRC inspectors and licensee personnel, on several occasions there was a failure by the licensee's staff to properly perform operability evaluations, appropriately control compensatory actions resulting from degraded conditions, and appropriately question operability evaluations. In addition, plant material condition was degrading.

O1.2 Operability Evaluations (Both Units)

a. Inspection Scope (71707)

During the course of routine observations of control room activities, the inspectors questioned an operability evaluation on charging system leakage between Unit 1 and Unit 2. The inspectors reviewed an additional sample of operability evaluations, reviewed the licensee's operability evaluation procedures, and interviewed Performance Assurance (PA) personnel concerning their observations on operability evaluations.

b. Observations and Findings

The licensee utilized an operability determination process to document the evaluation of degraded conditions to assess the impact of the condition on the ability of the system to meet its design basis. All operability evaluations, whether performed by the shift technical advisor (STA) or engineering, were reviewed and approved by the operations shift manager (OSM). During engineering performance of backup operability evaluations, engineering personnel would review the operability evaluation performed by operations personnel. The inspectors reviewed a sample of completed operability evaluations to assess the impact of changing plant conditions on previously performed operability evaluations, and to evaluate if the operability evaluation clearly stated the

bounding limits of applicability. The results of the completed reviews are discussed below.

b.1.1 CVCS Cross-Tie Leakage

On October 8, 1998, the licensee had noted Chemical and Volume Control System (CVCS) cross-tie leakage. The leakage was originally less than 1 gallon per minute (gpm) but steadily increased. The leakage resulted in Reactor Coolant System (RCS) water being transferred between Units. Due to system configurations and the resultant differences in pressures, water was transferred from Unit 1 to Unit 2. Since the boron concentration in Unit 2 was slightly less than Unit 1, fluid transfer from Unit 1 to Unit 2 resulted in slight increases in the Unit 2 RCS boron concentration. The licensee maintained system line-ups and configurations which ensured that the CVCS cross-tie leakage was from Unit 1 to Unit 2, thereby ensuring that the undesired dilution of boron concentration in either unit did not occur.

On January 5, 1999, the inspectors reviewed and questioned the operability evaluation performed by the STA. The licensee documented the concerns in Condition Report (CR) 98-5707. The initial operability evaluation was written on October 8, 1998, and another operability evaluation was written on December 12, 1998, when the cross-tie leakage was approximately 2.0 gpm. On January 5, 1999, the leakage was approximately 4.0 gpm and increasing.

The inspectors determined that the operability evaluations for the CVCS cross-tie leakage were weak and lacked supporting detail, as evidenced by:

- The use of the wrong value from Technical Specifications (TSs) for the amount of water from the refueling water storage tank (RWST) that was required to maintain shutdown requirements and core cooling. The amount used was conservative, but was incorrect.

- The failure to request a backup operability determination from engineering.

- The failure to specify that the prompt operability evaluation was appropriate for only short term usage and was narrowly focused on the then current leak rate.

In addition to the above problems, the operability evaluation did not clearly state the bounding limits or verify that the bounding limits were being tracked.

The inspectors reviewed Plant Managers Procedure (PMP) 7030.OPR.001, Revision 1, “Operability Determinations,” and had the following comments:

- The procedure lacked guidance concerning the plant conditions for when compensatory actions should be taken.

- The procedure lacked guidance concerning the identification of plant conditions necessary to ensure the operability evaluation remained bounded.

- The procedure lacked guidance concerning the tracking of plant conditions to ensure the operability evaluation remained valid.

- The procedure lacked guidance concerning the need to issue or revise procedures to control compensatory measures.

Following inspector questions regarding the maximum cross-tie leakage for which the operability evaluation would be acceptable, licensee personnel performed Engineering Supporting Operability Analysis Number 91-18-ENSM 990107WF. The inspectors' review of this "backup" operability evaluation determined that:

- There were fewer administrative controls contained within the new operability evaluation than the original operability evaluation;

- The engineering operability evaluation was a narrowly focused discussion of a calculation which determined the maximum allowable cross-tie leakage.

Discussions with engineering personnel determined that the backup operability evaluation had indeed been narrowly focused on determining the maximum allowable cross-tie leakage and was not intended to replace the original operability evaluation. This narrow focus of the operability evaluation did not follow the guidance contained within Engineering Procedure 12, Engineering Head Procedure (EHP) 7030.OPR.001, Revision 0. The procedure title was "Supporting Analyses for Operability Determinations," and the purpose, as stated in the procedure, was to provide supporting analyses, however, the procedure steps required a broader evaluation than just an analysis. It appeared that the engineering personnel were following the purpose of the procedure but were not following the procedure steps.

Licensee corrective actions included:

- Issuing revised operability evaluations with clearly defined bounding limits.

- Increasing the priority of the repair efforts to the leaking CVCS cross-tie valves.

- Providing management guidance to the STAs and the engineers regarding the quality of the operability evaluations.

- Bringing in a contractor to act as project manager to revise the operability evaluation process, review the previously completed operability evaluations, and provide assistance in the performance of operability evaluations during the programmatic improvement process.

Licensee Procedure 12 EHP 7030.OPR.001, Revision 0, required, in part, that "Any compensatory or corrective actions must be clearly documented and the appropriate tracking mechanism put in place to ensure their completion." 10 CFR Part 50, Appendix B, Criterion V, Instructions, Procedures and Drawings, states, in part, that activities affecting quality shall be accomplished in accordance with procedures. Step 6.2.7, of EHP 7030.OPR.001, required, in part, "Any compensatory or corrective actions must be clearly documented and the appropriate tracking mechanism put in place

to ensure their completion." The failure of the backup operability evaluation to document and implement appropriate tracking mechanisms compensatory measures was a violation. (NOV 50-315/98027-02).

b.1.2 Assessment of Compensatory Measures

As stated in NRC Generic Letter 91-18, Resolution of Degraded and Non-Conforming Conditions:

> "In its evaluation of the impact of a degraded or nonconforming condition on plant operation and on operability of SSCs [Structures, Systems, or Components], a licensee may decide to implement a compensatory measure as an interim step to restore operability or to otherwise enhance the capability of SSCs until the final corrective action is complete."

Licensee Condition Report 98-5707 written on December 12, 1998, stated:

> "In that it is not certain which valve contributes to the crosstie leakage, Shift Personnel have entered a contingency action in the "Contingency Action Log" to assure the proper RWST [Refueling Water Storage Tank] boration is met. The contingency is simply to assure that if the RWST is used for emergency borating that the charging flow-rate should be adjusted to exceed 120 gpm by a value that corresponds to the last known cross-tie leakage. This accounts for any crosstie leakage that would divert flow from the Unit 1 charging flowpath to Unit 2."

The inspectors reviewed the Contingency Action Log (CAL) in the control room. The log contained instructions which stated, "During emergency boration, either from BAST [Boric Acid Storage Tank] or RWST, the charging header cross-tie leakage must be considered when establishing the required flows." The inspectors considered the instructions vague and reliant on operator memory for implementation. The contingency actions for the CVCS cross-tie leakage did not clearly state the actions required of the operators.

The CAL was required to be reviewed by each of the licensed operators prior to assuming a shift. There were 13 open contingencies contained within the CAL, each requiring that the operator remember to perform some extra or modified action. Due to the need to have at least 120 gpm to comply with TSs, the importance of increasing the charging header flow rate to account for any leakage to the other unit, and the risk of relying upon operator memory to achieve these goals, the need to increase the boration flow should have been considered for addition to the appropriate licensee procedure.

With the units cooled down and de-pressurized, the Centrifugal Charging Pumps could inject approximately 550 gpm. Upon switching the Centrifugal Charging Pumps to the emergency boration mode, the flow rate to the RCS would be much greater than the required 120 gpm. The loss of the few gallons per minute to Unit 2 would not represent a safety significant problem.

b.2 Residual Heat Removal (RHR) Instrument Line Support Bracket Failure (Unit 1)

On October 22, 1998, a non-licensed operator on a routine plant tour identified a broken support bracket to RHR flow indicator sensing line 1-IFI-331-V1. The flow indication is for RHR flow to the containment spray header. While no flow was going to the containment spray header, the line experiences vibrational loads from flow through a near-by branch line to the reactor coolant system. A maintenance request was written to repair the broken bracket and an operability evaluation was performed.

The operability evaluation stated that the bracket was installed to address vibration issues; however, the evaluation was focused on seismic questions. Because the bracket was not installed for seismic loads, the operability evaluation stated the missing bracket did not affect operability. The evaluation failed to address the original purpose of the bracket which was to limit operational vibration loads. A supporting operability evaluation was subsequently performed by engineering personnel and also focused on the seismic aspects of the missing bracket and failed to address the operational vibration loads.

On January 8, 1999, a PA auditor re-identified the broken bracket and reviewed the operability evaluation. Performance Assurance personnel appropriately determined that the broken bracket increased the likelihood of an RHR leak and immediate action was taken to repair the bracket. The inspectors determined that the PA auditor demonstrated a critical questioning attitude and effectively performed the independent assessment function in reviewing the circumstances related to the broken support bracket.

b.3 Licensee Corrective Actions

In response to earlier operability evaluation issues raised by PA, licensee management implemented improvements to the operability procedure and operability review process. While the corrective actions had been ongoing for several months, the procedure had not been approved by the end of the report period. In addition, immediate corrective action had not been taken to ensure that operability evaluations performed during the corrective action implementation were of good quality. Following the inspectors and PA auditors' observations noted above, licensee senior management increased the priority of the planned revisions to the operability evaluation procedure and implemented a temporary process to improve the quality of operability evaluations.

The team performing the revision to the operability process was also performing reviews of the existing operability evaluations. The team had nearly completed compiling the evaluations that already existed and would then begin a detailed technical/quality review of the operability evaluations. It was estimated by licensee personnel that there were approximately 200 to 300 operability evaluations to be reviewed.

c. Conclusions

Operability evaluations and procedures reviewed by NRC inspectors and licensee personnel showed that shift managers, shift technical advisors, and engineers failed to adequately review operability evaluations. Procedural guidance on compensatory measures and administrative controls for those compensatory measures were weak or non-existent. Specifically, plant administrative procedures addressing operability

evaluations did not contain requirements or administrative controls for compensatory measures. In the case of engineering procedure guidance concerning the conduct of operability evaluations, the requirements were not being followed. One violation for failing to follow a procedure was identified.

Performance Assurance personnel demonstrated a critical questioning attitude in assessing the circumstances related to a broken RHR support bracket which resulted in the identification of an inadequate operability evaluation.

O1.3 Cold Weather Preparation (Both Units)

a. Inspection Scope (71714)

The inspectors verified that the licensee had carried out their routine tasks necessary to prepare the plant for cold weather. The inspectors reviewed the job order, verified that work documents were performed, and conducted a plant walkdown. The inspectors reviewed the following documents:

- R0014700, “Plant Winterization”
- Temporary Modification log

b. Observations and Findings

Plant winterization at D. C. Cook was ensured by completion of a recurring task job order that details mechanical and electrical alignments for cold weather. Ventilation fans are cautioned tagged, the fan openings are sealed, and electric heating units are placed in areas prone to cold air intrusion.

On January 4, 1999, the sensing lines for the traveling screen differential pressure detectors froze and caused erratic indications. The frozen sensing lines resulted in automatic initiation of screen wash cycles which required operator action to correct. The screen house area where the traveling screens are located had been winterized in accordance with the plant job order, but the combination of sub-zero outside temperatures and 32EF circulating water resulted in ice formation. The licensee investigated the formation of ice in the forebay area of the intake structure. The Unit 2 side of the screenhouse had three steam heaters out-of-service during the time when the sensing lines froze. The ambient temperature in the screenhouse increased when the three heaters were returned to service. Engineering and operations personnel determined that the low heat load of the plant did not contribute enough energy to the cooling water discharge stream to make it effective to use the normal anti-icing mode of operation of mixing discharge flow with the incoming water to raise the temperature. The procedure for operation of the Essential Service Water system was changed to permit returning Essential Service Water discharge flow to the forebay to raise the intake temperature. The licensee continued to monitor the forebay for the formation of ice during periods of freezing conditions.

On the evening of January 13, 1999, conditions favorable to the formation of frazil ice existed for the water intake structure. The licensee proactively determined actions

necessary to minimize the potential for frazil ice formation. These actions included securing the circulating water pumps to minimize flow rates in the intake structure, aligning the return of ESW to the forebay to add heat load, and aligning the plant heating boiler blowdown to the forebay to add additional heat. The forebay was continuously monitored during the period of favorable conditions. No frazil ice build-up was detected.

During a walkdown of areas adjacent to the plant exterior, the inspectors identified numerous areas where snow was being driven into the plant interior by high winds. The snow was not collecting on any surfaces but there was evidence of cold air infiltration into the plant. Exterior doors had evidence of snow being forced under the door sweep by the wind. In the Unit 1 and Unit 2 east steam generator stop valve enclosures, only some of the exhaust fans had been sealed with herculite material to stop cold air infiltration but other fans did not. Plant personnel performed walkdowns to identify areas that needed further attention.

The licensee continued to have trouble with the ability of the plant heating boiler to maintain physical plant temperatures above freezing with the reactors shut down. The plant heating boiler capacity had been reduced by boiler tube leaks, combustion air system back draft damper deficiencies, and a boiler level control system designed for much larger steam loadings than those experienced with both reactors shut down. In 1997 a commercial, oil-fired boiler was installed as a back-up to the plant heating boiler. The alternate heating boiler was not connected to the plant heating steam system at the optimum location which prevented the alternate boiler from functioning as a full back-up in extremely cold weather. Modifications to the tie-in point require shutdown of the plant heating boiler, and thus depended on the weather to allow the opportunity to perform the work when heating loads were lower.

On January 5, 1999, Unit 1 and 2 Control Room Pressurization fans were declared inoperable due to frost and snow accumulation on the recirculation inlet grating. Snow was driven into the duct by wind and migrated to the control room. The cold air/snow formed a frost on the ventilation grating which subsequently melted and dripped onto control room equipment. The intrusion of snow into the duct had been a long standing problem, requiring an annual installation of drip catches to protect equipment. The drip catches were not installed prior to the intrusion of snow because the Temporary Modification needed to install the drip catch had not been completed.

On January 6, 1999, an electrical ground was identified on a 600 volt AC bus in Unit 1. Licensee personnel conducted an investigation and determined that the most likely cause for the ground was moisture from melting snow. The snow was being drawn into the 4kV room via the ventilation supply fans intake. The snow buildup was removed from the roof in the vicinity of the air intakes. No prior provision had been made to prevent the intrusion of snow into the 4kV rooms.

Summary of observations and findings

The licensee failed to prevent snow intrusion into the site buildings, even though the areas susceptible were previously known. In addition, the inspectors observed that as each instance of snow/cold problems was identified, the licensee at first responded

narrowly to each specific instance vice addressing the cold weather preparation inadequacies from a more programmatic perspective.

c. Conclusions

The licensee did not have a process for ensuring that the plant winter preparations were integrated across organizational boundaries. The lack of integration contributed to the inadequate preparation for cold weather, especially considering the lack of process heat in the buildings due to the current shutdown condition. Cold weather preparations were untimely and severe cold weather caused multiple challenges to the plant operators.

O7 Quality Assurance in Operations

O7.1 Review Backlog of Condition Reports (CRs)

On January 11, 1999, the licensee declared all four emergency diesel generators inoperable (reference NRC Event Number 35242). The emergency diesel generators were declared inoperable following a determination by the licensee that there was inadequate evidence that safety-related relays had been properly tested following maintenance. Engineering personnel were unable to provide evidence of the testing in a reasonable period of time and the shift manager declared the components known to contain the relays inoperable. The licensee complied with the TS requirements to immediately suspend all operations involving core alterations or positive reactivity additions.

As part of the follow-up to this finding, licensee personnel determined that a condition report written on January 6, 1999, remained in the OSMs office without review until January 11, 1999. The OSMs normally reviewed CRs on the same day they entered the OSMs office. Due to a large increase in the number of condition reports being initiated, and the need to monitor the performance of their work crews, the OSMs had let the backlog of condition reports build up. The backlog was estimated to consist of approximately 100 condition reports.

Licensee management could not explain why neither the OSMs or STAs questioned the backlog of condition reports, nor why the independent shift mentors failed to identify the backlog of condition reports. The administrative duty to review condition reports for operability issues became so burdensome that the operators fell behind in the performance of the reviews.

The licensee added personnel to the OSM's office to assist in the review of the CRs and the process was modified so that the OSM was not required to review all CRs. The licensee also initiated a root cause analysis to identify and correct the cause of the sudden influx of CRs to the OSM's office.

The licensee's corrective action program (CAP) is undergoing major assessment and corrective actions. The CR process improvement needs, weakness, and corrective actions discussed in this section were incorporated in the CAP improvement action plan.

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:

- C0045960, ”Reactor Coolant Drain Tank Vent Valve2-WD-255, Disassemble and Replace Diaphragm”
- **01 Operations Head Procedure 4030.STP.027CD, “CD Diesel Generator Operability Test (Train A),” Revision 13
- A162057, Radiation monitor unexpected spiking
- A158930, Residual Heat Removal (RHR) Pump 2 East, discharge valve 2-IMO-314 leaks by
- A162866, Unit 2 West RHR heat exchanger outlet valve 2-IRV-320 leaks by

b. Observations and Findings

On December 2, 1999, licensee contract workers inadvertently started to disassemble the wrong valve on a reactor coolant drain tank vent line. The workers intended to work valve 2-WD-255; however, they had started to work on valve 2-WD-252, a physically identical valve located about 8 feet away. The licensee’s apparent cause determination found that the workers had improperly read the valve label and listed several environmental factors which contributed to the error. The inspectors discussed the event with a maintenance supervisor and reviewed the completed apparent cause determination. The inspectors concluded that the apparent cause determination was weak in that it did not include the earlier failures of a similar nature. The earlier wrong component errors were documented in Inspection Reports 50-315/98012; 50-316/98012 and 50-315/98016; 50-316/98016.

The licensee’s CAP is undergoing major assessment and corrective actions. The failure of the licensee to include previous similar events during root cause investigations was one of the weaknesses designed to be addressed in the CAP improvement action plan.

c. Conclusions

Overall, observed maintenance work activities were performed effectively. The inspectors observed that, overall, the workers followed procedures and appropriately documented the required information.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Material Condition Issues (Both Units)

a. Inspection Scope (62707 and 71707)

During the inspection report period, plant operation and maintenance activities were impacted by the material condition of the plant. The inspectors followed up on several activities which were affected.

b. Observations and Findings

Reactor Coolant Pump Seal Injection Flow (Unit 1)

On November 25, 1998, the licensee identified that the Unit 1 charging header flow control valve, 1-QRV-251, was leaking by. The leak by resulted in all four Unit 1 reactor coolant pump (RCP) seal injection flow rates rising from about 8 gpm to 15 gpm. Seal leak-off flow rates remained normal. After consulting with a vendor representative, the licensee determined that the excessive seal injection flow would not result in damage to the RCPs in Mode 5; however, the long term effects of the increased flow were unknown.

The licensee bypassed and isolated 1-QRV-251 and manually throttled the charging header flow. Valve 1-QRV-251 was subsequently repaired and satisfactorily tested.

Charging and Volume Control System (CVCS) Cross Tie Leakage (Both Units)

On October 8, 1998, the licensee had identified that the CVCS unit cross-tie valves were leaking by. Early in the inspection period, the licensee determined that the CVCS unit cross-tie valves were leaking by at about 2 gpm. The leakage resulted in a gradual lowering of level in the Unit 1 RWST and a gradual increase in the Unit 2 Volume Control Tank level. The licensee performed an operability evaluation (discussed above in Section O1.2) and determined that the plant could continue to operate with the CVCS cross-tie leakage.

Over the course of this inspection period, the cross-tie leak rate increased from about 2 gpm to 4 gpm. On December 22, 1998, the licensee voluntarily entered an orange shutdown risk condition in order to refill the Unit 1 RWST. The orange shutdown risk condition was a result of having no operable boration paths available for Unit 1 while the Unit 1 RWST was being filled. The licensee complied with the appropriate TS action statements to prevent inadvertent positive reactivity additions; therefore, the orange shutdown risk condition was not safety significant. As a result of the increased leakage and the impact on plant operations, licensee management made the CVCS cross-tie repairs a high priority item for the station. Over the weekend of January 9 - 10, 1999, both units were depressurized in preparation for this work. At the end of the inspection period, the CVCS cross-tie repairs were in progress.

Component Cooling Water (Unit 1)

On December 3, 1998, the licensee started to drain the east side of the Unit 1 Component Cooling Water (CCW) surge tank for maintenance. (There is one common surge tank for the east and west trains of Unit 1 CCW with a baffle plate dividing it in half.) The control room chart recorder indicated that the west side of the surge tank was draining rather than the east side. The operators stopped the draining, and the licensee investigated the abnormal indications.

The licensee determined that the west side of the surge tank was draining to the east CCW train through one or more leaking west train CCW cross-connect valves, and Action Requests were written to repair the valves. The inspectors reviewed the licensee's investigation results and questioned operations management on the need for CCW train separation. Operations management determined that train separation was not required in Mode 5; however, the ability to isolate the trains from one another during postulated abnormal operating events would need to be verified prior to entering Mode 4. The inspectors concluded that the licensee's original investigation into the CCW draining event narrowly focused on the specific effects of individual leaking CCW cross-connect valves. The investigation did not address the potential effect of multiple leaking cross-connect valves on the operation of the CCW system as a whole.

Residual Heat Removal Pipe Support (Unit 1)

On October 22, 1998, a non-licensed operator identified a broken support bracket to RHR flow indicator sensing line 1-IFI-331-V1. The flow indication is for RHR flow to the containment spray header. While no flow is going to the containment spray header, the line experiences vibrational loads from flow through a near-by branch to the reactor coolant system. An Action Request was written to repair the broken bracket and an operability evaluation was performed. (The quality of the operability evaluation is discussed in the Section O1.2 above.)

On January 8, 1999, while performing a walkdown and assessment of the material condition of Unit 1, Train B, a PA auditor identified that the bracket was not repaired. In 1982, the instrument line had developed cracks due to operation without adequate structural support. The failure to address the vibrational stress to the instrument line represented another material condition issue.

Other Material Condition Issues

The inspectors observed small boric acid leaks from the RHR system and the CVCS system. These leaks did not represent a threat to the operability of the systems but had existed for several months without being repaired.

Other material condition issues and the dates they were originally determined to be problems were:

Description of Problem	Date
CVCS Charging header flow indicator 1-QFI-200 reads approximately 20 gpm with no flow in the system	10/7/98
Unit 1 120 Volt Vital Inverter 1-CRID-1 unreliable due to frequent auto transfers to alternate source and immediate auto return to primary source	1/10/99
Normal and emergency boric acid flow paths are blocked with boric acid	11/9/98
Non-Essential Service Water expansion joint has a 2" long crack	7/31/98
The tempering dampers for the supply and exhaust fans to the Unit 2 AB D/G will not open in automatic	1/13/99
Alternate charging line shutoff valve 1-QRV-61 indicates intermediate when control switch is taken to close position	1/1/99
Unit 2 West charging pump inboard seal leaks at approximately 1 quart to ½ gallon per minute	12/10/98
There is an unexplained mis-match between normal charging flow. With indicator 2-QFI-200, reading about 26 gpm lower than letdown flow indicator 2-QFI-301, indicators calibrated successfully, suspect valve leak by	12/31/98
Unit 1 West component cooling water pump leaking oil from the inboard bearing. Approximately 6" diameter puddle per day	12/12/98
Base plate for spring support 1-MSH-4, main steam lead spring type pipe support, is pulled slightly from the concrete	10/31/98
Due to excessive leakby of 1-QRV-251, charging flow control valve causing reactor coolant pump seal injection to increase beyond procedural limits, valve 1-CS-302 was throttled. This will reduce the potential for damage to 2-QRV-200 as it would now take the full pressure drop.	11/27/98
Slow air leak on back up control air to pressurizer power operated relief valve 1-NRV-153	11/19/98
A stud/nut on the base plate of the Unit 1 West RHR pump is corroded	1/12/99
Minor oil leak on Unit 1 West RHR pump motor upper bearing assembly	11/17/98
Unit 2 West component cooling water radiation monitor 2-CRS-4401 high alarm does not work and the monitor sticks occasionally in check source	1/2/99

Description of Problem	Date
Due to periodic spikes on radiation monitor R17B the Unit 2 component cooling water surge tank vent could automatically closed. Need to perform periodic verifications to ensure it remains open. Spikes are occurring approximately 2 times per shift.	10/16/98
Unit 2 East RHR pump discharge cross-tie valve 2-IMO-314 leaks by	5/31/98
Unit 2 West RHR heat exchanger outlet valve 2-IRV-320 leaks by	5/31/98

The licensee had focused efforts on correcting issues on the ice condensers, recirculation sumps, fibrous material inside of containment and other issues referenced in the NRC Manual Chapter 0350 issues for D. C. Cook. The material condition of these components has improved dramatically. However, this re-direction of maintenance efforts has resulted in a decline in the normally good material condition of the units. The licensee expects to spend additional time identifying and correcting engineering and corrective action plan issues prior to resuming normal maintenance. The inspectors were concerned that the resultant delay in conducting corrective maintenance activities could potentially result in the continued degradation of plant material condition.

c. Conclusions

Based upon an assessment of licensee corrective action requests, condition reports, contingency action logs, and in plant observations, the inspectors concluded that the material condition of several safety-related systems had declined during the forced outage. The delay in addressing equipment problems was due, in part, to the licensee's focus on corrective actions to address programmatic issues that are a part of the licensee's restart plan.

III. Engineering

E1 Conduct of Engineering

E1.1 General Comments (37551)

On December 28, 1998, licensee management decided to defer the scheduled Train B outage. The decision was based on an engineer's finding that a component evaluation for a safety-related Train A motor operated valve (1-WMO-725, an essential service water supply valve to the Unit 1 CD emergency diesel generator) was based on an unverified calculation. The plant manager stated that, prior to proceeding with the Train B work, all of the Train A work would need to be affirmed as complete and correct. The inspectors concluded that the engineer's finding and plant manager's decision to delay the Train B restart work represented an example of conservative decision making. At the end of this inspection period, the Train A affirmation was not complete.

The inspectors periodically investigated engineering problems or incidents to assess the causes of the selected engineering problem. The inspectors observed that, overall, engineering activities were slowed by improvement initiatives resulting in issues not being resolved in a timely manner. Substantial efforts by the licensee's engineering organization, self-assessment organization, and by independent contractors had identified numerous problems and aggressive corrective actions were in progress.

E1.2 Plant Drawings (37551)

The licensee identified that following modifications not all drawings had been updated in a timely manner. There were approximately 6,000 drawings in the backlog awaiting revision. The licensee had initiated corrective actions to ensure that the drawings were updated in a more timely fashion and to ensure that the drawings needed for day-to-day operation and maintenance of the facility were accurate.

The licensee determined that the backlog of drawing revisions had not created an unsafe condition due to the separation of the plant drawings into those that needed prompt updating and those that could be delayed. Drawings used for operations and maintenance of the facility were included in the list of the ones requiring prompt updating and had been appropriately maintained. The licensee began using additional resources to update the remaining drawings.

The inspectors verified that the drawings being maintained up-to-date were those needed by plant personnel in the day-to-day operation of the facility. The inspectors reviewed the list of drawings to ensure that the appropriate drawings were being maintained. In addition, the inspectors reviewed CRs to determine whether plant personnel were encountering situations where out-of-date drawings were resulting in problems in the field. Lastly, the inspectors interviewed a sample of operators, mechanics, electricians, welders, Quality Control inspectors, I &C technicians, maintenance planners, and engineering support technicians to assess the impact of out-of-date drawings. All the personnel interviewed stated that only minor problems were being encountered and that the appropriate drawings for their use were being maintained up-to-date.

10 CFR Part 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, the failure of the licensee to update plant drawings following modifications was a violation. 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 (50-315/98027-01(DRP)).

E1.3 Safety Evaluation Issues

Recent NRC and licensee inspection activities have identified a breakdown in the licensee's program for performing safety evaluations in accordance with 10 CFR 50.59. As part of the plant restart effort docketed in the Restart Plan, the licensee has committed to performing a complete assessment of the safety evaluation program and

implementing actions to correct the identified deficiencies. In a letter dated July 30, 1998, and updated on October 13, 1998, the NRC informed the licensee that an oversight panel had been established in accordance with NRC MC 0350, and a checklist was enclosed which specified activities which the NRC considered necessary to be addressed prior to restart. Enclosure 1 to the NRC letter, the Case Specific Checklist, included the failure to perform safety evaluations and the performance of inadequate evaluations as an item to be addressed prior to restart. In accordance with MC 0350, an inspection plan was developed to evaluate the effectiveness of the licensee's actions to correct the items listed on the Case Specific Checklist.

Previous inspection activities have also identified specific discrepancies in the licensee's performance of safety evaluations. The inspectors reviewed these previously identified deficiencies and assessed the corrective actions specific to these issues. The programmatic safety evaluation weaknesses will be addressed in future inspections as delineated by the NRC MC 0350 process. Therefore, the following items are closed:

- **!** (Closed) Violation 50-316/96002-02: Inadequate safety evaluation for tarp installed over reactor cavity. On February 26, 1996, during a tour of containment, the inspectors noted a 8 ft by 8 ft yellow tarp secured over the refuel pool which was not installed in accordance with an approved safety evaluation. The licensee responded by removing the tarp and performing an analysis of the as found configuration. The licensee determined that the tarp did not impact the operability of the containment lower drains and the recirculation sump. A stop work order for all work in the Unit 2 containment was issued, and prior to restarting work, all workers were retrained on the potential consequences of loose debris in the containment. Additionally, the requirements for work inside containment in Modes 1 through 4 have been added to Attachment 6 of Plant Manager's Instruction 2293, "On Line Maintenance of Important Systems." The attachment requires that any work be evaluated for potential to prevent the proper operation of the recirculation sump following a loss of coolant accident. The inspectors reviewed the licensee's response and determined that the specific conditions identified in the violation had been adequately addressed.

- **!** (Closed) Violation 50-316/96004-01: Failure to perform safety review for procedure changes. In March 1996 the inspectors identified two examples of procedure changes where the intent of the procedure had been affected, but the temporary change process had been used. The temporary change process allowed delaying the safety screening for the procedure change for up to 14 days after the procedure change went into effect. Subsequently, the licensee performed a safety screening of the revised procedure prior to its use and found that no unreviewed safety question existed. Additionally, Plant Manager's Instruction 2010, "Instructions, Procedures, and Associated Indexes Policy," was revised to include the definition of the term "change of intent" and require a safety screening for any change to the intent of a procedure prior to the use of the procedure. The inspectors reviewed the licensee's response and determined that the specific conditions identified in the violation have been adequately addressed.

E7 Quality Assurance in Engineering

E7.1 (Closed) Apparent Violation (EEI) 50-315/98021-03: Engineered safeguards ventilation system may not be capable of meeting its design basis. In 1997, an NRC inspection team questioned the accuracy of the engineered safety features ventilation system heat gain calculation. Specifically, the calculation inputs for essential service water flow and post-LOCA containment sump temperature were not in accordance with the values described in the Updated Final Safety Analysis Report (UFSAR). A revised vendor calculation performed in response to the NRC finding showed that the engineered safety features (ESF) ventilation (AES) system may not be capable of meeting its design basis. Based on the calculation results, the licensee determined that some ESF room temperatures would exceed 125ºF under certain plant conditions. Condition Report 98-6364 was written to document this finding.

The inspectors concluded that the identification of the AES design discrepancy resulted from the earlier NRC identification that the AES heat gain calculation did not use the same plant parameter values documented in the UFSAR. The earlier finding was included as Item C.2.l in the letter dated October 13, 1998, which issued a Notice of Violation and Proposed Civil Penalty. Because the corrective actions for the earlier finding would necessarily involve revising the AES heat gain calculation with the correct input parameters specified in the UFSAR, the inspectors considered this issue to be an additional aspect of the original violation; no new violations were identified.

The licensee performed an environmental qualification (EQ) review of all safety-related equipment located in the rooms served by the AES system. The assessment concluded that all of the components required for safe shutdown of the respective units were considered operable with respect to the higher calculated ESF room temperatures. The inspectors reviewed the backup operability determination and corrective actions and had no questions. This EEI is closed.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Violation 50-315/96014-04; 50-316/96014-04: Inadequate justification for a safety evaluation. The inspectors identified that the safety evaluation for Design Change Package (DCP) 12-DCP-0049, Revision 1, “Spent Fuel Pool (AFX) Filtration System Bypass Damper Replacement,” was inadequate. Specifically, the safety evaluation relied on the lack of specific design and operation information in the UFSAR to determine that an unreviewed safety question (USQ) did not exist for the damper replacement. The licensee’s written records indicated that no USQ existed; however, the records did not provide an adequate basis for that conclusion.

In 1997, the Architectural and Engineering (AE) inspection (Inspection Report 50-315/97201; 50-316/97201) identified that the safety evaluation for 12-DCP-0049 did not address all potential failure modes for the instrument air system. After installing the damper modification, both trains of the engineered safety features ventilation (AES) system were vulnerable to a single failure of the nonsafety-related 85 psig instrument air header. As installed, the modification supplied the filter bypass dampers (normally open, fail closed) from the 85 psig instrument air system and the filter

inlet dampers (normally closed, fail open) from the 20 psig instrument air system. A failure of only the 85 psig instrument air system would result in the total isolation of the AES ventilation filters. If both sets of dampers were on the same header, then upon header loss, the dampers would go to their fail safe positions (bypass dampers closed and inlet dampers open). The failure of the design review to identify this single failure vulnerability was documented in the AE inspection and Inspection Report 50-315/98009; 50-316/98009 as Unresolved Item 50-315/98009-13; 50-316/98009-13.

Subsequently, the licensee revised the 12-DCP-0049 modification to supply both the inlet damper and bypass damper from the 85 psig instrument air header, eliminating the single failure vulnerability. The inspectors reviewed the vendor documentation for the replacement dampers and determined that no USQ existed. The inspectors concluded that the specific conditions which resulted in this violation have been corrected. However, the broader issue of inadequate safety reviews will be tracked under Unresolved Item 50-315/98009-13; 50-316/98009-13. This violation is closed.

E8.2 (Closed) Licensee Event Report 50-315/97023-00, -01: Design Change Introduces Possibility of Single Failure Which Could Result in Loss of Both Trains of ESF Ventilation Due to Failure to Identify Adverse Impact During Design Review. The issues raised by this licensee event report (LER) are discussed above in Section E8.1 and will also be tracked under Unresolved Item 50-315/98009-13; 50-316/98009-13. This LER is closed.

IV. Plant Support

R1 Conduct of Radiation Protection and Chemistry (71750)

During normal resident inspection activities, routine observations were conducted in area of radiation protection and chemistry 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 area 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 January 13, 1999.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

#G. Arent, Nuclear Licensing
#P. Barrett, Performance Assurance
#B. Bennett, Operations
#D. Cooper, Plant Manager
#MB. Depuydt, Nuclear Licensing Supervisor
#R. Eckstein, Engineering
#M. Finissi, Electrical and Auxiliary Systems Engineering
#R. Gillespie, Work Control Manager
#MB Greendonner, Protection
#D. Hafer, Plant Engineering Manager
#W. Kropp, Performance Assurance
#D. Kunsemiller, Director Regulatory Affairs
#D. Noble, Radiation Protection/Chemistry Superintendent
#T. O'Leary, Performance Assurance
#B. O'Rourke, Licensing
#F. Pisarsky, Performance Engineering
#R. Powers, Senior Vice President
#T. Quaka, Engineering Effectiveness
#J. Sampson, Site Vice President
#M. Skow, Performance Assurance
#J. Sankey, Engineering
#M. Stark, Maintenance
#L. Van Ginhoven, Materials Management
#W. Walschot, Corrective Action Program Manager
#B. Zemo, Engineering

# Denotes those present at the January 13, 1999, exit meeting.

INSPECTION PROCEDURES USED

IP 37551	Onsite Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observation
IP 71707	Plant Operations
IP 71714	Cold Weather Preparations
IP 71750	Plant Support Activities
IP 92700	Onsite Review of LERs

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/98027-01	NCV	Failure to update plant drawings in a timely manner
50-315/98027-02	NOV	Failure to follow procedure

Closed

50-316/96002-02	VIO	Inadequate safety evaluation for tarp installed over reactor cavity
50-316/96004-01	VIO	Failure to perform safety review for procedure changes
50-315/96014-04 50-316/96014-04	VIO	Inadequate justification for a safety evaluation
50-315/97023-01	LER	Design Change Introduces Possibility of Single Failure Which Could Result in Loss of Both Trains of ESF Ventilation Due to Failure to Identify Adverse Impact During Design Review
50-315/98021-03	EEI	Engineered safeguards ventilation system may not be capable of meeting its design basis
50-315/98027-01	NCV	Failure to update plant drawings in a timely manner

LIST OF ACRONYMS

AES	Engineered Safety Features Ventilation System
AR	Action Request
BAST	Boric Acid Storage Tank
bcc	blind carbon copy
CAL	Contingency Action Logs
CAP	Corrective Action Program
cc	carbon copy
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CVCS	Chemical and Volume Control System
DCC	Donald C. Cook
DCP	Design Change Package
DRP	Division of Reactor Projects
EEI	Apparent Violation
EHP	Engineering Head Procedure
ENPT	Engineering Performance Testing Procedure
ESF	Engineered Safety Feature
IFI	Inspector Followup Item
IR	Inspection Report
JO	Job Order
LCO	Limiting Condition for Operation
LER	Licensee Event Report
MI	Michigan
NCV	Non-cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OHP	Operations Head Procedure
OSM	Operations Shift Manager
PA	Performance Assurance
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PPA	Plant Performance Assurance
PDR	Public Document Room
QA	Quality Assurance
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RTD	Resistance Temperature Detector
RWST	Refueling Water Storage Tank
STA	Shift Technical Advisor
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation