July 7, 2006

Mr. M. Nazar
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: D.C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2, NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000315/2006009(DRS) 316/2006009(DRS)

Dear Mr. Nazar:

On May 26, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the D. C. Cook Nuclear Power Plant. Inspectors conducted subsequent in-office review prior to report issuance to resolve several issues identified during the onsite inspection. The enclosed report documents the results of the inspection, which were discussed with Mr. J. Jensen, and others of your staff at the completion of the inspection on May 26, 2006 and on July 6, 2006.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of the inspection, three NRC identified findings of very low safety significance were identified, which involved violations of NRC requirements. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the D. C. Cook Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000315/2006009(DRS); 05000316/2006009(DRS)

cc w/encl: M. Peifer, Site Vice President
L. Weber, Plant Manager
G. White, Michigan Public Service Commission
L. Brandon, Michigan Department of Environmental Quality - Waste and Hazardous Materials Division
Emergency Management Division MI Department of State Police
State Liaison Officer, State of Michigan
D. Lochbaum, Union of Concerned Scientists

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000315/2006009(DRS); 05000316/2006009(DRS)

cc w/encl: M. Peifer, Site Vice President
L. Weber, Plant Manager
G. White, Michigan Public Service Commission
L. Brandon, Michigan Department of Environmental Quality - Waste and Hazardous Materials Division
Emergency Management Division MI Department of State Police
State Liaison Officer, State of Michigan
D. Lochbaum, Union of Concerned Scientists

DOCUMENT NAME:E:\Filenet\ML061920589.wpd
☐ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☐ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII	DRS	RIII	DRS	RIII		RIII	
NAME	DHills for RDaley: ls		CLipa		DHills			
DATE	07/07/06		07/07/06		07/07/06			

OFFICIAL RECORD COPY

ADAMS Distribution:
JLD
PST
RidsNrrDirsIrib
GEG
KGO
BJK1
CAA1
LSL
C. Pederson, DRS
DRPIII
DRSIII
PLB1
JRK1
ROPreports@nrc.gov

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-315, 50-316
License Nos.:	DRP-58; DRP-74
Report No:	05000315/2006009(DRS); 05000316/2006009(DRS)
Licensee:	American Electric Power Company
Facility:	D.C. Cook Nuclear Power Plant, Units 1 and 2
Location:	7700 Red Arrow Hwy Stevensville, MI 49127-9726
Dates:	May 8, 2006 through May 26, 2006
Inspectors:	R. Daley, Senior Reactor Inspector C. Acosta, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000315/2006009(DRS); 05000316/2006009(DRS); 05/08/2006 through 05/26/2006; D.C. Cook Nuclear Power Plant, Units 1 and 2; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a two week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by two regional based engineering inspectors. Three Green Non-Cited Violations (NCV) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply, may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3; dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for the deletion of four sections of the Technical Requirements Manual (TRM). Specifically, the licensee deleted Sections 8.4.7, Tavg Lower Limit, 8.6.1, Ice Bed Temperature Monitoring System, and 8.6.2, Inlet Door Position Monitoring System, and 8.3.7, Post Accident Monitoring (PAM) Instrumentation, Table 8.3.7-1 without evaluating these changes per the requirements of 10 CFR 50.59.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors, at the time of the inspection, could not reasonably determine that the Updated Final Safety Analysis Report (UFSAR) change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, even though these TRM sections along with their associated surveillance requirements were deleted, the licensee was able to show that all deleted surveillance requirements had been performed satisfactorily and within their prescribed frequency in spite of the deletion. This issue was entered into the licensee's corrective action program. (Section 1R02.1.b.1)

Green. The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for the modification of the 2-East Centrifugal Charging Pump (CCP). Specifically, the licensee performed modifications to the 2-East Centrifugal Charging Pump that required more restrictive frequency requirements to be established than were already in the Technical Specifications. Had a 10 CFR 50.59 evaluation been performed, as required, the

evaluation should have shown that a change to the Technical Specifications (TS) was required so that the new required frequency value could be incorporated into the applicable TS Surveillance Requirements. This issue was entered into the licensee's corrective action program.

Because the issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the modification of the 2-East Centrifugal Charging Pump would not have ultimately required NRC approval. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening for the mitigating systems cornerstone and determined that the finding was of very low safety significance because they were able to answer "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, while the 10 CFR 50.59 evaluation, and ultimately the required license amendment, were not performed as required, administrative controls were put into place after the modification was performed such that the CCP would always be able to perform its function. (Section 1R17.1.b.1)

Green. The inspections identified a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance. Specifically, verification of containment lower compartment average temperature per Surveillance Requirement (SR) 3.6.5.2 was being performed using temperature readings that were not representative (and non-conservative) of the true average temperature in the lower containment. The issue was entered into the licensee's corrective action program.

The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of "Design Control," and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the methodology for determining lower containment average temperature was non-conservative and did not account for the heightened temperatures that were experienced in the Steam Generator (SG) Enclosure Rooms. Had average temperature been above the TS limits, temperatures during a Design Basis Accident could have exceeded the ratings of safety related mitigating equipment thereby challenging the functionality of the equipment. This finding was of very low safety significance, because the inspectors answered "no" to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after performing a calculation that included the SG Enclosure Rooms, the licensee determined that under worst case historical conditions, average air temperature was 119.5 degrees which was still less than the TS requirement of 120 degrees F. (Section 1R17.1.b.2)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From May 8 through May 26, 2006, the inspectors reviewed six evaluations performed pursuant to 10 CFR 50.59. The inspectors confirmed that the evaluations were thorough and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 14 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

b. Findings

b.1 Inappropriate Deletion of Technical Requirements Manual Sections

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR 50.59, "Changes, Tests, and Experiments," having very low safety significance (Green) for the deletion of portions of the TRM without performing an evaluation in accordance with 10 CFR 50.59. Specifically, the licensee deleted four TRM sections. While portions of these sections may have been able to be removed under the provisions of 10 CFR 50.59, certain portions, particularly the surveillance requirements, would have needed NRC approval prior to being removed.

Description: During implementation of the Improved Technical Specifications (ITS) at D.C. Cook Nuclear Power Plant, certain requirements that were originally a part of the Current Technical Specifications (CTS) were removed from the Technical Specifications and relocated to the Technical Requirements Manual. After relocation, these requirements were subject to the controls of 10 CFR 50.59. For purposes of future changes to the TRM, these requirements were considered to be "incorporated by reference" into the UFSAR.

During this process in June 2005, 10 CFR 50.59 Screens 2005-0139-00, 2005-0334-00, and 2005-0421-00 were performed to delete certain of these requirements from the TRM prior to implementation of TRM Revision 0. These Screens deleted TRM 8.3.7, Post Accident Monitoring (PAM) Instrumentation, 8.4.7, Tavg Lower Limit, 8.6.1, Ice Bed Temperature Monitoring System, and 8.6.2, Inlet Door Position Monitoring System. The licensee treated the changes as simple procedure changes rather than an actual deletion of UFSAR requirements. Consequently, the licensee incorrectly determined that the changes did not adversely affect the UFSAR. As a result, the required 10 CFR 50.59 evaluation was not performed for these changes. Additionally, since the Screens also deleted the surveillance requirements in those TRM sections, portions of the deletions would not have successfully passed a 10 CFR 50.59 evaluation. Deletion of portions of these testing requirements for equipment important to safety would result in more than a minimal increase in the likelihood of a malfunction of a structure, system, or component important to safety since surveillance testing minimizes the probability of failure of equipment.

Based upon the inspectors' findings, the licensee issued Condition Report (CR) 06129041. As a result of the CR, the licensee verified that all deleted surveillance requirements were performed within their previous required frequencies. Additionally, the licensee planned to re-evaluate the TRM sections that were deleted and put the portions that could not be justified for removal under 10 CFR 50.59 back into the TRM.

Analysis: The inspectors determined that this issue was a performance deficiency since, in June 2005, the licensee failed to perform a required safety evaluation in accordance with 10 CFR 50.59. Specifically, the licensee deleted TRM Sections 8.4.7, Tavg Lower Limit, 8.6.1, Ice Bed Temperature Monitoring System, and 8.6.2, Inlet Door Position Monitoring System, and 8.3.7, Post Accident Monitoring (PAM) Instrumentation without evaluating this change per the requirements of 10 CFR 50.59. The finding was determined to be more than minor because the inspectors could not reasonably determine that these TRM deletions would not have ultimately required NRC prior approval.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, while the TRM deletions may not have been able to be done without a license amendment, the licensee was able to show that all deleted surveillance requirements had been performed satisfactorily and within their prescribed frequency in spite of the deletion. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments as described in the UFSAR. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, the licensee, in June 2005, deleted portions (Sections 8.3.7, 8.4.7, 8.6.2, and 8.6.1) of the TRM, constituting a change in procedures as described in the UFSAR, without providing the required written evaluation. In addition, the inspection team could not reasonably determine that these changes would not have required a license amendment, because the deletion of the associated surveillance requirements could have resulted in more than a minimal increase in the likelihood of a malfunction of a structure, system, or component important to safety. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program (CR 06129041), it is considered a Non-Cited Violation consistent with VI.A.1 of the NRC Enforcement Policy (NCV). (NCV 05000315/2006009-01; 05000316/2006009-01 (DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From May 8 through May 26, 2006, the inspectors reviewed six permanent plant modifications that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, one modification was chosen that affected the barrier integrity cornerstone. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

b.1 Failure to Perform 10 CFR 50.59 Evaluation for Modification to the 2-East Centrifugal Charging Pump

Introduction: The inspectors identified a Non-Cited Violation of 10 CFR 50.59, "Changes, Tests, and Experiments," having very low safety significance (Green) for the modification of the 2-East Centrifugal Charging Pump. Specifically, the licensee implemented this

modification with a 10 CFR 50.59 Screen even though it required a Technical Specification change to implement.

Description: In the first quarter of 2006, D.C. Cook Nuclear Plant staff initiated and approved EE-2005-0436 allowing for the replacement of the CCP impeller and shaft. Because of the different operating characteristics of the new impeller, the modified pump would draw a higher Brake Horsepower (BHP) than the original pump. While there was a recognition that this higher BHP would exist, the licensee never addressed the effects of this higher horsepower on the pump motor.

Prior to implementation of the modification, the licensee discovered that with the higher BHP, under the worst case high frequency of 61.2 Hz, the CCP motor would exceed the motor nameplate rating of 690 horsepower (HP) (600 HP with a 15 percent service factor). Because of this discovery, the licensee issued CR 06089052 on March 30, 2006. At this time, the design modification had still not been field implemented.

Because of the loading problems with the motor, the licensee performed an evaluation to show that the motor would still be able to operate within its design rating as long as the frequency was held below 60.5 Hz. As stated in the CR, "On this bases, several Technical Specification Surveillance Requirements, which have an acceptance criteria of 61.2 Hz on the high end, are non-conservative." At this point, instead of delaying the modification, the licensee decided to perform the change; however, more stringent administrative controls were to be established to address the non-conservative TS issues. The licensee believed that this approach was allowable because of the guidance provided in NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety." Based upon this decision, the licensee implemented changes to the plant TRM to ensure that Emergency Diesel Generators (EDG) frequency would always be below 60.5 Hz. Additionally, the licensee ensured that all TS surveillances performed that required an upper band of 61.2 Hz met the more stringent criteria of 60.5 Hz. This showed that the EDGs would always start and load at a frequency lower than 60.5 Hz.

While the licensee had adequately addressed any operability concerns by imposing these more restrictive administrative controls, the licensee's use of Administrative Letter 98-10 as a tool to knowingly enter a condition (by performing the modification) involving non-conservative Technical Specification Surveillance Requirements was not appropriate. Specifically, Administrative Letter 98-10 was to be used if plant staff found non-conservative Technical Specification values. It was not a justification for creating a non-conservative Technical Specification condition. Additionally, the team further noted that when it was discovered that the CCP motor would exceed the motor nameplate rating, the licensee should have realized that the scope and the effects of the modification had changed. At this point, had the licensee re-evaluated the modification and performed a 10 CFR 50.59 review and evaluation, the results should have shown that a change to the Technical Specifications was required so that the new required frequency value of 60.5 Hz could be incorporated into the applicable TS Surveillance Requirements. The inspectors determined that this change would have required a license amendment prior to being performed.

Based upon the inspectors' concerns, the licensee issued CR 06143098. Because, the

licensee had already adequately addressed administrative controls for this condition, there were no immediate operability concerns.

Analysis: The inspectors determined that this failure to perform a safety evaluation in accordance with 10 CFR 50.59 for changes made to the UFSAR was a performance deficiency warranting a significance determination. Specifically, the licensee performed modifications to the 2-East Centrifugal Charging Pump that required more restrictive frequency requirements to be established than were already in the Technical Specifications. Had a 10 CFR 50.59 evaluation been performed, as required, the evaluation should have shown that a change to the Technical Specifications was required so that the new required frequency value could be incorporated into the applicable TS Surveillance Requirements. The finding was determined to be more than minor because the required technical specification change for the CCP modification would have required NRC prior approval.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the significance determination process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors completed a significance determination of the underlying technical issue using NRC's inspection manual chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and answered "no" to the Mitigating Systems screening questions in the Phase 1 Screening Worksheet. Specifically, while the 10 CFR 50.59 evaluation, and ultimately the required license amendment, were not performed as required, administrative controls were put into place prior to plant operations such that the CCP would always be able to perform its function. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV Violation.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment. Additionally, 10 CFR 50.59(c)(I) states, in part, that the licensee may make changes to the facility provided the change does not require a change to the technical specifications incorporated in the license.

Contrary to the above, the licensee, in 2006, performed a modification to the 2-East CCP without performing the required safety evaluation. Additionally, because the change caused the high frequency band of 61.2 Hz to no longer be adhered to, the change should have involved a TS amendment to change the frequency bands for the applicable TS surveillance requirements prior to implementation of the modification. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program

(CR 06143098), it is considered a Non-Cited Violation consistent with VI.A.1 of the NRC Enforcement Policy (NCV). (NCV 05000315/2006009-02; 05000316/2006009-02 (DRS))

b.2 Non-Conservative Verification of Containment Average Air Temperature

Introduction: The inspectors identified a Non-Cited Violation having a very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, the licensee was verifying containment lower compartment average temperature per Surveillance Requirement (SR) 3.6.5.2 by using temperature readings that were not representative of the true average temperature in the lower containment.

Description: In ICP-01045, Revision 00, “Unit 1 and Unit 2 Steam Generator Compartment High Temperature Alarm Change,” the licensee documented a setpoint change to the temperature alarm from 140 degrees to 180 degrees Fahrenheit (F). This change was necessary since the Steam Generator Enclosure areas, which were located in the containment lower compartment, were operating at temperatures in excess of 140 degrees Fahrenheit.

During the inspection, the team questioned how TS 3.6.5, “Containment Air Temperature,” was addressed considering these higher temperatures. Specifically, Limiting Condition for Operability (LCO) 3.6.5 requires that average lower containment air temperature be less than or equal to 120 degrees F. Additionally, Surveillance Requirement 3.6.5.2 requires that the containment lower compartment air temperature be verified within limits (120 degrees F). The Bases for SR 3.6.5.2 states, “Verifying that containment average air temperature is within the LCO limits ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated using measurement taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere . . . In the lower compartment, the locations at nominal elevations 626 ft 6 inches, 624 ft 10 ½ inches, and 624 ft 0 inches are used.” While the location of these temperature instruments were representative of the majority of the lower containment, they were not representative of the SG Enclosure Rooms which were essentially isolated from the other areas in the lower containment. Because the SG Enclosure Rooms contained a significant amount of the air volume for the lower containment (Preliminarily calculated at approximately 12 percent of total volume), the inspectors were concerned that SR 3.6.5.2 was no longer conservative. The SG Enclosure Rooms were historically operating at temperatures in excess of 140 degrees F; however, the temperatures used in determining acceptability of SR 3.6.5.2 did not account for these higher temperatures at all.

Based upon the inspectors concerns, the licensee initiated CR 06145114. Because of the present outside ambient and lake temperatures, the licensee was able to determine that the lower containment was operable. Additionally, the licensee performed a past operability evaluation. In that evaluation, the licensee determined that temperatures in the SG Enclosure Rooms had historically reached as high as 164.3 degrees F. However, after performing a lower containment temperature calculation for the worst case days in the recent past, the licensee determined that the highest average air temperature was still within TS limits. After including the SG Enclosure Rooms, the

licensee determined a worst case average air temperature of 119.5 degrees. While this temperature was within the TS limit of 120 degrees F, it was significantly closer to the TS limit than the original method had determined.

Analysis: The inspectors determined that this failure to provide conservative acceptance criteria for SR 3.6.5.2 was a performance deficiency warranting a significance determination. The issue was more than minor because it was associated with the Mitigating System Cornerstone attribute of “Design Control,” and affected the cornerstone objective of ensuring the capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the methodology for determining lower containment average temperature was non-conservative and did not account for the heightened temperatures that were experienced in the SG Enclosure Rooms. Had average temperature been above the TS limits, temperatures during a Design Basis Accident could have exceeded the ratings of safety related mitigating equipment thereby challenging the functionality of the equipment.

The finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. Specifically, after performing a calculation that included the SG Enclosure Rooms, the licensee determined that under worst case historical conditions, average air temperature was 119.5 degrees which was still less than the TS requirement of 120 degrees F.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control” states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, no established design basis existed to support the use of the temperature measurements that were being used as a representative sample for the average temperature for the lower containment that was delineated in SR 3.6.5.2.

Because this failure to provide conservative temperature measurements for the lower containment average air temperature was determined to be of very low safety significance and because it was entered in the licensee’s corrective action program as CR 06145114, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000315/2006009-03; 05000316/2006009-03 (DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From May 8 through May 26, 2006, the inspectors reviewed ten Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate

the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. Jensen and others of the licensee's staff, on May 26, 2006 and on July 6, 2006. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Baker, Configuration Programs Supervisor
D. Fadel, Vice President Engineering
R. Kohrt, Safety Analysis Supervisor
R. Meister, NRA Senior Specialist
R. Neuendorf, Configurations Programs
M. Scarpello, NRA Supervisor

Nuclear Regulatory Commission

B. Kemker, Senior Resident Inspector
J. Lennartz, Resident Inspector
D. Hills, EB1 Branch Chief

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000315/2006006-01; 05000316/2006006-01	NCV	Inappropriate Deletion of Technical Requirements Manual Sections
05000315/2006009-02; 05000316/2006009-02	NCV	Failure to Perform 10 CFR 50.59 Evaluation for Modification to the 2-East Centrifugal Charging Pump
05000315/2006009-03; 05000316/2006009-03	NCV	Non-Conservative Verification of Containment Average Air Temperature

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather, that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

10 CFR 50.59 Screenings

Screen 2004-0382-00; Operation of the CVCS Holdup Tanks; Revision 00

Screen 2004-0480-00; Revise UFSAR Table 9.5.2 to Reflect a Change in the CCPs Heat Exchangers and PASS Minimum CCW Flow Requirements; Revision 0

Screen 2005-0039-00; Incorporation of Administrative Changes into the TRM Revision 0; Revision 00

Screen 2005-0045-00; Incorporation of Less Restrictive Change into TRM Section 8.3.6; Revision 00

Screen 2005-0210-00; ESW Flow Verification; Revision 00

Screen 2005-0316-00; Deletion of Technical Requirements Manual Section 8.3.7; June 18, 2005

Screen 2005-0317-00; Incorporation of Five Less Restrictive Changes into TRM Section 8.3.7 - L.1, 8.6.1 -L.1, and 8.6.2 - L.1 and L.2; Revision 00

Screen 2005-0340-00; ESW Flow Verification; Revision 00

Screen 2005-0453-00; Revision of ITS 3.8 Bases Section; Revision 00

Screen 2005-0506-00; Auxiliary Feed Pump Operation; Revision 00

Screen 2006-0017-01; Proceduralized Temporary Modification for Electrical Maintenance Procedure 2-IHP-5040-EMP-008, “Reactor Containment Building MOV Temporary; Revision 01

Screen 2006-0032-00; Interchanging the Spare Component Cooling Water Pump with the East or West Component Cooling Water Pump; January 25, 2006

Screen 2006-0032-01; Interchanging the Spare Component Cooling Water Pump with the East or West Component Cooling Water Pump; May 17, 2006

Screen 2006-0081-00; Revise UFSAR Section 8.7.2.1 to Clarify AFW Pump Operation During a Station Blackout (SBO) Event; Revision 00

10 CFR 50.59 Evaluations

2004-0367-00; Change to UFSAR Design Basis Value for Ice Condenser Bypass in Containment Divider Barrier; Revision 00

2004-0575-00; Revision to Spray System Information Described in Table 14.3.1-12; Revision 0

2005-0187-01; Addition of Supplemental Diesel Generators (SDGs) for EDG Allowed Outage Time Extension; Revision 01

2005-0420-00; SPV Unit 2 Reactor Trip Switches that Could Eliminate an Unexpected Trip; Revision 00

2005-0469-00; Revision to Cook Nuclear Plant Unit 1 Emergency Operating Procedure OHP 4023 ECA 0.0, Step 4; Revision 00

2006-0090-00; Unit 2 Cycle 16 Core Reload: Changes to Unit 2 TRM 8.1.2 in Regard to the Required Minimum RWST Volume; Changes to Unit 1 and Unit 2 TS Bases in Regard to the Power Distribution Surveillance Exclusion Zone; Changes to Unit 1 and Unit 2 TRM App A in Regard to the OTdT Trip Setpoint Equation; Revision 0

IR17 Permanent Plant Modifications (71111.17B)

Modifications

1-CMM-30059; Removal of Inner Screens From Unit 1AB EDG Intake Ventilation System Ductwork; Revision 0

2-CMM-40775; SPV Unit 2 Rewire Reactor Trip Switches that Could Eliminate an Unexpected Trip; Revision 00

ICP-01045; Unit 1 and Unit 2 Steam Generator Compartment High Temperature Alarm Change; Revision 0

2-LDCP-4925; Restoration of the Design Basis for Line 2-CCW-131; Revision 0

2-LDCP-5205; Add East and West RHR Pump Minimum Flow Line High Point Vents; Revision 0

2-MOD-45677-R0; 2-CPN-2-CCW Inner Cooling Coil Replacement; Revision 0

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CR 06129041; During the Modification-50.59 inspection, it was identified that a 50.59 Screen did not identify that a 50.59 Evaluation was required; May 9, 2006

CR 06131037; NRC inspector Questioned Safety Classification of Valve 12-CCW-168 in Screen 2006-0032-00; May 11, 2006

CR 06132017; Adequacy of Design Inputs for 1-CMM-30059; May 12, 2006

CR 06132019; Adequacy of Design Inputs for 2-LDCP-4925; May 12, 2006

CR 06132022; Adequacy of Design Inputs for 2-LDCP-5205; May 12, 2006

CR 06139070; In the evaluation of CR 04259057, the CCW pumps were accidentally omitted from the list of safety-related 4KV pumps to which a particular purchase code (SU-11) should be applied; May 23, 2006

CR 06142069; NRC questioned adequacy of the 50.59 evaluation for 2-CMM-40775, Rewire Reactor Trip Switches that Could Eliminate an Unexpected Trip; May 22, 2006

CR 06143098; NRC identified that the 50.59 evaluation associated with the 2-East Centrifugal Charging Pump (CCP) rotating assembly was inadequate; May 23, 2006

CR 06144042; Lack of adequate independent verification for the inlet and outlet of the spare CCW pumps; May 24, 2006

CR 06144055; NRC identified that 2-CMM-40775 and its associated 50.59 Evaluation did not receive a PORC review; May 24, 2006

CR 06144069; The air intake/rain hood for the unit 1 AB diesel is supported by three columns. The base plate for two of these columns does not have 100 percent bearing support area; May 24, 2006

CR 06144081; NRC inspector identiifed that on the anchors for support 1-GRH-R26A, there are a variety of washer configurations (from one washer to 10 washers) and requested to show its acceptability; May 24, 2006

CR 06145114; Lower containment air temperature may not be representative of the entire lower volume; May 25, 2006

Corrective Action Program Documents Reviewed During the Inspection

P-99-00993; Excessive Vibration in RHR System; January, 15, 1999

P-99-09724; The Spare CCW Pump (12-P-10) Has No Internal Corrosion Protection; dated April 28, 1999

P-99-25560; Seismic Impact Review Is Not Contained in the Modification Closeout Package 12-RFC-1070; dated October 18, 1999

CR 03234063; Request Verification That Plant Commitment No. 6576, is Met in 2-EHP-4030-216-248; dated August 22, 2003

CR 04223098; Unit 1 and Unit 2 Steam Generator Enclosure temperatures are above the temperatures assumed in the bounding analysis for short term Containment Subcompartment (SG Enclosure) presure as stated in UFSAR Section 14.3.4.2.4; August 10, 2004

CR 04232026; Procedures that Inject Cold AFW Into SGs at power Do Not Provide Information on How Much Power Should Be Reduced to Prevent Exceeding RTP Limits; dated August 19, 2004

CR 05266036; During the NRC SSDPC, NRC identified two issues with UFSAR Section 7.2.1 concerning limitations for control of AFW flow during an SBO event; September 23, 2005

CR06040062; Defective Fuel Pellets Were Detected in Some Fuel Rods; dated February 9, 2006

CR 06055065; A design input error for Calculation 1-E-N-ELCP-4KV-001 and 2-E-N-ELCP-4KV-001 was verified regarding EDG loading; February 24, 2006

CR 06089052; Effects of EDG Technical Specification SR 3.8.1.2 of 61.2 Hz on the Unit 2 CCP motor; March 30, 2006

<u>Calculations</u>

TH-98-09; Bounding Containment Spray Flow Rates; Revision 2

TH-99-13; CST Volume Calculation; Revision 00

TH-05-03; AFW Injection Impact Assessment; Revision 0

<u>Drawings</u>

12-5728-27; Heating and Ventilation Elect. SW. GR and Diesel Gen. Area Section and Details; Revision 27

12-5728B; Heating and Ventilating Elect. Switchgear and Diesel Generator Area Section and Details; Revision 0

OP-1-5135-41; CCW Pumps and CCW Heat Exchangers; dated May 5, 2003

OP_2-5143-63; Emerg. Core Cooling (RHR) Unit No.2; dated April 12, 2006

OP-1-12001-74; Main Auxiliary One-Line Diagram Bus “A” & “B” Engineered Safety System; Revision 70

OP-1-12010-23; MCC Aux One-Line 600V Bus 11A, 11B Engineered Safety System; Revision 23

Procedures

01-OHP-4023-ECA-0.0; Loss of all AC Power; Revision 18

PMI-1040; Plant Operations Review Committee; Revision 25

Miscellaneous Documents

AEP:NRC:0896L; Expedited Technical Specification Change to Delete Overload Test for Diesel Generators; October 31, 1988

DIT-B-00802-10; CCW Flow Balance Criteria for Procedures 01-EHP-4030-116-248 and 02-EHP-4030-216-248; dated September 23, 2003

DIT-B-00856-01; Effects of Reduced Temperature (55 degrees F) on the Capacity of the Battery 1-BATT-AB; September 30, 2000

DIT-B-01977-00; Battery Room Temperature Requirements to Support SBO Analysis; March 7, 2001

DIT-B-02743-00; Design Basis Configurations of Penetration CPN-2; dated June 8, 2003

DIT-B-03000-00; Provide Justification for CNP Continued Operation Despite Increased Temperature Above 120 degrees F; July 23, 2005

DIT-S-01491-00; Maximum Nitrogen Pressure to be Applied to the CRAC North and South Liquid Chiller Condenser Control Valve During the Performance of the ESW Flow Verification Procedure; dated April 15, 2005

E-Mod 12-MOD-45617; Addition of Supplemental Diesel Generators (SDGs) for EDG Allowed Outage Time Extension; Revision 1

EE-2005-0436; Complete Centrifugal Charging Pump Assembly, Ingersoll-Dresser (Flowserve) Model 2-1/2 RLIJ Modified; Revision 2

ES-VALVE-1430-QCN; Gate and Glove Valves, Sizes 2" and Smaller; dated July 22, 1996

JO 05230023-01; Rewire/Install Jumpers on Reactor Trip Switch; April 17, 2006

JO R0276416; Inspect and Clean Intake Screens for 1-HV-DGS-1 AB Emergency Diesel Generator Room Ventilation Supply Fan; January 20, 2006

Letter to AEP Nuclear Generation Group DC Cook Nuclear Plant Units 1 and 2; SECL-97-189 Revision 2, FSAR Change to Support Increased CCW Temperatures; dated November 12, 1997

Letter to AEP Nuclear Generation Group DC Cook Nuclear Plant Units 1 and 2; Unit 2 Large Break LOCA Analysis; dated June 30, 1999

Letter to AEP Nuclear Generation Group DC Cook Nuclear Plant Units 1 and 2Response to AEP Request For Evaluation for DC Cook Unit 2 Spray Flow Increase; dated July 21, 1999

Letter to AEP Nuclear Generation Group DC Cook Nuclear Plant Units 1 and 2; Input to Operability Determination/Functionality Assessment for Damaged Annular Axial Blanket Pellet; dated February 9, 2006

Letter to AEP Nuclear Generation Group DC Cook Nuclear Plant Units 1 and 2; Impact of Damaged Fuel Pellets on Current Reload Evaluations; dated February 10, 2006

NRC SER; Station Blackout Analysis, Donald C. Cook Nuclear Plant, Units 1 and 2; October 31, 1991

Screen 2005-0583-00; Change to Annunciator Response Procedures 1-OHP-4024-121 and 1-OHP-4024-221; December 29, 2005

SD-011120-001; Design Basis Analysis of Containment Penetrations CPN-2, CPN-3, CPN-4 and CPN-5; Revision 0

TB-04-22; Westinghouse Technical Bulletin, Reactor Coolant Pump Seal Performance - Appendix R Compliance and Loss of all Seal Cooling; Revision 1

VTD-JOYT-0001; Joy Technologies Installation and Manual Series 800/1000/2000/3000 Axivane Fan; Revision 0

VTD-JOYT-0010; Joy Technologies Specifications, Drawings, and Bill of Material for SF-27402; Revision 0

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
ASME	American Society of Mechanical Engineers
BHP	Brake Horsepower
BTP	Branch Technical Position
CFR	Code of Federal Regulations
CCP	Centrifugal Charging Pump
CR	Condition Report
CTS	Current Technical Specifications
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
F	Fahrenheit
HP	Horsepower
IMC	Inspection Manual Chapter
IR	Inspection Report
ITS	Improved Technical Specifications
LCO	Limiting Condition for Operability
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PAM	Post Accident Monitoring
PARS	Publicly Available Records
PRA	Probabilistic Risk Assessment
SDP	Significance Determination Process
SRA	Senior Risk Analyst
SG	Steam Generator
SR	Surveillance Requirement
TRM	Technical Requirements Manual
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report