December 22, 2005

Mr. Gary Van Middlesworth
Site Vice-President
Duane Arnold Energy Center
Nuclear Management Company, LLC
3277 DAEC Road
Palo, IA 52324

SUBJECT: DUANE ARNOLD NUCLEAR POWER STATION, NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000331/2005013 (DRS)

Dear Mr. Van Middlesworth:

On November 18, 2005, 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 Duane Arnold Nuclear Power Station. The enclosed report documents the results of the inspection, which were discussed with Mr. J. Bjorseth and others of your staff at the completion of the inspection on November 18, 2005 and by telephone on December 21, 2005.

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, two NRC-identified findings of very low safety significance were identified, both of 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.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public

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Sincerely,

/RA/

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

Docket Nos. 50-331
License Nos. DPR-49

Enclosure: Inspection Report 05000263/2005013(DRS)

cc w/encl: E. Protsch, Executive Vice President -
Energy Delivery, Alliant;
President, IES Utilities, Inc.
C. Anderson, Senior Vice President, Group Operations
J. Cowan, Executive Vice President and Chief Nuclear Officer
J. Bjorseth, Site Director
D. Curtland, Plant Manager
S. Catron, Manager, Regulatory Affairs
J. Rogoff, Vice President, Counsel, & Secretary
B. Lacy, Nuclear Asset Manager
Chairman, Linn County Board of Supervisors
Chairperson, Iowa Utilities Board
The Honorable Charles W. Larson, Jr.
Iowa State Senator
D. Flater, Chief, Iowa Department of Public Health
D. McGhee, Iowa Department of Public Health

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The Honorable Charles W. Larson, Jr.
Iowa State Senator
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D. McGhee, Iowa Department of Public Health

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	05000331/2005013 (DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Duane Arnold Nuclear Power Station
Location:	3277 DAEC Road Palo, IA 52324
Dates:	November 14 through 18, 2005
Inspectors:	R. Daley, Senior Reactor Inspector, Team Leader A. Dahbur, Reactor Inspector M. Garza, Reactor Inspector M. Munir, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

Enclosure

SUMMARY OF FINDINGS

IR 05000331/2005013 (DRS); 11/14/2005 - 11/18/2005; Duane Arnold Nuclear Power Station; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a one-week announced baseline inspection on evaluations of changes, tests or experiments and permanent plant modifications. The inspection was conducted by four regional based engineering inspectors. Two 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

Green. The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59 in that the licensee failed to perform an adequate safety evaluation review for changes made to the facility as described in the Updated Final Safety Analysis Report (UFSAR). Specifically, the licensee adversely changed the description in the UFSAR of the license basis function of the recirculation pump runback in that the recirculation runback feature could no longer prevent a reactor scram if a feedwater pump tripped. Within the 10 CFR 50.59 evaluation, the licensee failed to provide a basis for why this malfunction of the recirculation pumps' runback logic (equipment important to safety) did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a Structures, Systems, and Components (SSC) important to safety.

Because the issue affected 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 UFSAR change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The finding was determined to be of very low safety significance (Green) because the recirculation runback feature was not a mitigating function. (Section 1R02.1.b.1).

Green. A finding of very low safety significance was identified by the inspectors associated with a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," where the licensee had not evaluated and updated the plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher temperatures in the Condenser and Heater Bays. After identification by the team, the licensee was able to demonstrate that even though the higher temperatures decreased the ampacity margins for the effected cabling, it did not decrease the margins to the limit where the cabling would fail if called upon to provide power to equipment important to safety.

The finding was more than minor because it affected the mitigating system cornerstone attribute of “Design Control.” Specifically, the licensee did not account for high temperature conditions in the Condenser and Heater Bay room that adversely affected the ampacity of cabling supplying power to equipment important to safety. This finding was of very low safety significance because it screened out using the Phase 1 worksheet. Specifically, the licensee’s preliminary evaluation determined that the higher temperatures would not prevent pertinent equipment from functioning.
(Section 1R17.1.b.1)

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 November 14 through 18, 2005, the inspectors reviewed three 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 team could not review the minimum sample size of five evaluations, because the licensees only performed three evaluations during the biennial sample period. The inspectors also reviewed 12 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 Updated Final Safety Analysis Report (UFSAR) Change Reducing Capability of the Automatic Runback of the Recirculation Pumps on a Feedwater Pump Trip

Introduction: The inspectors identified that the licensee did not perform an adequate safety evaluation in accordance with 10 CFR 50.59, when the licensee made changes to the UFSAR. Specifically, the licensee failed to provide adequate bases when they determined that changes to UFSAR Section 1.3.2.8.1 "Runback of Recirculation Pump on Feedwater Pump Trip" did not require a licensee amendment. The licensee failed to address the malfunction of the recirculation pumps' runback logic which was designed to prevent a reactor scram in the event of one feedwater pump trip. The issue was considered to be of very low safety significance, (Green) and was dispositioned as a Severity Level IV Non-Cited Violation (NCV).

Description: During review of Duane Arnold 10 CFR 50.59 Screening Number 3409, the team questioned changes, referenced in the screening, to the UFSAR by safety evaluation SE-98-011. The inspectors were concerned that the licensee did not provide adequate bases when they determined that changes to UFSAR Section 1.3.2.8.1 "Runback of Recirculation Pump on Feedwater Pump Trip" did not require a licensee amendment.

Specifically, prior to the implementation of SE-98-001, UFSAR Section 1.3.2.8.1 previously stated, "automatic runback of the recirculation pumps on a feedwater pump trip results in a reactor power reduction that is within the capabilities of the feedwater system with only one pump. The correction for the loss of one feedwater pump is designed to be fast enough to prevent a reactor scram. See section 7.9.4.3 of the initial FSAR." Based on plant response and experience, the licensee found out that this was not necessarily true. The automatic runback of the recirculation pumps on a feedwater pump trip did not allow the feedwater system to respond fast enough to prevent a reactor scram. Therefore, the licensee revised Section 1.3.2.8.1 to state, "automatic runback of the recirculation pumps on a feedwater pump trip results in a reactor power reduction which may not be within the capabilities of the feedwater system with only one pump. The correction for the loss of one feedwater pump may not be fast enough to prevent the reactor scram." The licensee also deleted the reference to section 7.9.4.3 of the UFSAR which stated that a scram will not occur with a single feedwater pump trip.

In the safety evaluation for this change, the licensee answered "no" to the following question, "May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety." The licensee justification for this answer stated, "The activity changed the wording in the UFSAR to describe more accurately how the plant responds on a feedwater pump trip. The Loss of Feedwater Flow transient is already described in the UFSAR Section 15.6.3 and has already been evaluated. This section analyzes the total loss of feedwater and is concluded that this transient is a non-limiting event and bounds one feedwater pump trip. No physical changes occurred in the plant as a result of the change." The inspectors questioned the correctness and the adequacy of the bases for the licensee's justification, because the malfunction of the recirculation pump runback logic, the equipment important to safety, was not addressed by the safety evaluation. The inspectors noted that this change to the UFSAR, together with the inability of the recirculation pump runback to prevent a scram, may have resulted in a change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of Structures, Systems, and Components (SSC) important to safety.

Following identification of this issue, the licensee entered the issue into their corrective action program as Action Request (AR) CAP038955.

Analysis: The inspectors determined that this issue was a performance deficiency since, in 1998, the licensee failed to provide adequate basis for changes made to the UFSAR in accordance with 10 CFR 50.59. Specifically, the licensee failed to provide a basis for why this malfunction of the recirculation pumps' runback logic (equipment important to safety) did not present more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. 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). The finding was determined to be more than minor because the inspectors could not reasonably determine that the changes to UFSAR Section 1.3.2.8.1 would not have ultimately required NRC prior 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." The inspectors answered "no" to the Transient Initiator screening question in the Phase 1 Screening Worksheet which states, "Does the finding contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions will not be available," because the recirculation runback feature is not a mitigating 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.

Contrary to the above, in their safety evaluation, SE 98-011, the licensee failed to provide an adequate basis for the determination that the revision to UFSAR Section 1.3.2.8.1 was acceptable without a license amendment. Specifically, the licensee adversely changed the description in the UFSAR of the license basis function of the recirculation pump runback in that the recirculation runback feature could no longer prevent a reactor scram if a feedwater pump tripped. Within the 10 CFR 50.59 evaluation, the licensee failed to provide a basis for why this malfunction of the recirculation pumps' runback logic (equipment important to safety) did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a Structure, System and Component (SSC) 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 (CAP038955), it is considered a Non-Cited Violation consistent with VI.A.1 of the NRC Enforcement Policy (NCV).
(NCV 05000331/2005013-01 (DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From November 14 through 18, 2005, the inspectors reviewed eight 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 Consider Adverse Ampacity Effects of High Temperature Conditions in the Condenser and Heater Bay Room

Introduction: The inspectors identified a Non-Cited Violation (NCV) having very low safety significance (Green) of 10 CFR 50, Appendix B Criterion III, "Design Control." Specifically, the inspectors identified that the licensee had not evaluated and updated the plant cable ampacity calculation to determine the potential consequences of adverse effects to cabling due to higher temperatures in the Condenser and Heater Bays.

Description: Engineered Maintenance Action (EMA) A69614 raised the high temperature alarm for the Condenser and Heater Bay room from 127 degrees F to 140 degrees F. This change was performed because higher temperatures were being experienced in this area after the plant power uprate and because there was a certain amount of damaged or missing piping insulation in the area. The setpoint change was made to prevent the alarm from coming in, since the temperatures were frequently hitting or exceeding the setpoint.

The modification increased the alarm setpoint, but it did not address the effects of these heightened temperatures on the ampacity of electrical cables in the area. Since higher temperatures adversely affect the ampacity of electrical cables, the higher temperatures in the Condenser and Heaters Bay room had the potential to adversely affect the functionality and/or operability of equipment important to safety fed by cabling in these areas. The inspectors were concerned that the possibility existed that some of the equipment that were fed by cables in the area may not function due to possible faulting of the supply cables.

The licensee determined that Duane Arnold Ampacity Calculation 434-E001 assumed temperatures of 104 degrees and 122 degrees F. This was clearly non-conservative for the Condenser and Heater Bay room. As a result of the inspectors' concern, the licensee issued corrective action document CAP038933.

After performing a preliminary evaluation that assessed cabling in the area and equipment fed from that cabling, the licensee determined that there was no evidence that safety related Structures, Systems, and Components (SSCs) would not function as required. While the higher temperatures decreased the ampacity margins for the effected cabling, the licensee preliminarily determined that it did not decrease the margins to the limit where the cabling would fail if called upon to provide power to equipment important to safety.

Analysis: The inspectors determined that this issue was a performance deficiency since the licensee failed to meet the requirements of 10 CFR Part 50 Appendix B, Criterion III, "Design Control." Specifically, the licensee did not account for high temperature conditions in the Condenser and Heater Bay room that adversely affected the ampacity of cabling supplying power to equipment important to safety. The issue was more than minor because it affected the mitigating system cornerstone attribute of "Design Control." 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. In particular, the licensee's preliminary evaluation determined that the higher temperatures would not prevent pertinent equipment from functioning.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that measures shall be established to assure that applicable design basis are correctly translated into specifications, drawings, procedures and, instructions. Contrary to the above, the licensee did not have a design basis calculation for cable ampacity that supported the actual high temperatures that were being experienced in the Condenser and Heater Bay room. The Duane Arnold calculation that did address ampacity was significantly less conservative, since temperatures of 104 degrees and 122 degrees F were assumed while actual temperatures in the area were exceeding 127 degrees and were being allowed to go as high as 140 degrees F before alarms actuated.

Because the failure to address the adverse ampacity effects of heightened temperatures in this room was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CAP038933, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000331/2005013-02 (DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From November 14 through 18, 2005, the inspectors reviewed twelve Corrective Action Process documents (CAPs) that identified or were related to 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.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. Bjorseth and others of the licensee's staff on November 18, 2005 and by telephone on December 21, 2005. 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

<u>Licensee</u>

S. Catron, Nuclear Safety Assurance Manager
S. Haller, Site Engineering Director
R. Murrell, Regulatory Assurance Specialist
J. Swales, Engineer
L. Swenzinski, Regulatory Assurance Specialist

<u>Nuclear Regulatory Commission</u>
B. Burgess, Reactor Projects Branch 2
D. Hills, Chief, Engineering Branch 1
D. Spaulding, NRR Project Manager
G. Wilson, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000331/2005013-01	NCV	UFSAR Change Reducing Capability of the Automatic Runback of the Recirculation Pumps on a Feedwater Pump Trip
05000331/2005013-02	NCV	Failure to Consider Adverse Ampacity Effects of of High Temperature Conditions in the Condenser and Heater Bay Room

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

3281, Revision 1; UFSAR Change Request 03-024 is Removing the Specific Computer Program name "STRESS" From UFSAR Table 3.8-7; dated December 1, 2003

3313; Establish Required Airflow After Charcoal Addition to Ensure That the SBGT System Meets Design Function; dated December 2, 2003

3551; Clarify Cable Separation Requirements for New Installations; dated February 3, 2004

3415; Add steps to install/remove jumpers on the 45% Recirc. Runback signal; dated May 1, 2005

3678; Perform Condition Evaluation to Disposition Nonconformance of "Proper Clearance" Measurement of 4160 Circuit Breakers; dated March 5, 2004

3726; Verify Breaker Springs Charged for 1BR91/92 Crosstie; dated March 16, 2004

4101; TRMCR-016 (TSCR-067, Amendment #254); dated June 29, 2004

4371; UFSAR Change #04-003, CAP #30681 and 31021; dated September 21, 2005

4453; Revision to EMA A58898 Defines the Maximum Full Load Current as Calculated by CAL-E02-003, Revision 1 and UFSAR, Table 8.3-2; dated October 11, 2004

4895; TRMCR-015 (TSCR-064, Amendment #255); dated April 11, 2005

5132; Changing the pressure setpoint of PRV1634 based on CAL-M01-068; dated April 25, 2005

5386; Add Clarifying Statement to the TRM Bases for TLCO 3.3.6; dated August 22, 2005

5418; Replace the Existing General Service Water Pump With a New Pump With Higher Capacity; dated August 12, 2005

10 CFR 50.59 Evaluations

98-011; Revise UFSAR Section 1.3.2.8.1 “Runback of Recirculation Pump on Feedwater Pump Trip”; dated March 26, 1998

00-018; Safety Evaluation for EMA A50452 & A50453, Replacement of Crosby relief valves in HPCI and RCIC Systems; Revision 1

02-002; Allowance for Bypassing the RHRSW Strainer; Revision 1

05-001; Existing 1000 Gallon Gas Tank Relocation; dated August 30, 2005

IR17 Permanent Plant Modifications (71111.17B)

Modifications

ECP-1662; Core Spray Seal Replacement; Revision 0

ECP 1679; 1V-SF-56A Motor and Damper Control Wiring Modification; Revision 0

ECP-1696; Modify HPCI, RCIC, and RHR Piping in the Torus Airspace to Eliminate Flanged Joints; dated June 15, 2004

EMA A64464; Replacement of Control Building Heating Circulating Pump Motor 1VHP030B-M; Revision 0

EMA-A65920; Two Vent Lines Upstream of CV1621 and CV1579; dated March 9, 2004

EMA A67459; Revise Setpoint for PS2304B HPCI Booster Pump Suction Low Pressure Trip in accordance with CA-M04-11; dated February 18, 2005

EMA A69614; Raising of Setpoint for the Inlet Temperature for 1VAC021 and 1VAC022 from 127 degrees F to 140 degrees F; Revision 0

EMA 111689; Change Setting for Ground Fault Time Delay; Revision 0

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CAP038905; NRC 50.59 2005 Mod Inspection - ECP 1679; dated November 15, 2005

CAP038931; EMA A58898 - 2005 NRC 50.59/Mod Inspection; dated November 16, 2005

CAP038932; 50.59 Screening #5418 Weakness; dated November 16, 2005

CAP038933; Condenser Bay and Heater Bay Cable Ampacity Issue from 2005 NRC 50.59/Mod Inspection; dated November 16, 2005

CAP038934; Editorial Corrections to Station Blackout Analysis Report for Power Uprate; November 16, 2005

CAP038948; RCIC Vacuum Pump OOS without Compensatory Measures for Licensing Commitments; dated November 17, 2005

CAP038955; Inadequate 50.59 Evaluation 98-11 (2005 NRC 50.59/Mod Inspection); dated November 17, 2005

CAP038960; CAL-M97-008 (HPCI NPSH) Contains an Apparent Discrepancy; dated November 17, 2005

CAP038963; Ensure that the Basis for Tech Spec 3.6.2.1 Was Evaluated for Power Uprate; dated November 17, 2005

Corrective Action Program Documents Reviewed During the Inspection

CAP 029587; LT4541 (RX VESSEL WIDE RANGE (FLOOD) Cable Routing Violates Divisional Separation; dated October 30, 2003

CAP029976; STP 3.6.4.3-03 on SGTS B Stopped Due to Procedure Problems; dated December 1, 2003

CAP030242; HPCI & RCIC Steam Line Exhaust Test Connection; dated December 31, 2003

CAP030243; RHR Flanges in Torus Air Space to be Removed; dated December 31, 2003

CAP 030392; Potential 50.59 Screening Applicability Inconsistency; dated January 14, 2005

CAP031021; SSDI Unresolved Item - Station Blackout Analysis for EPU; dated March 17, 2004

CAP033128; Request TRM Bases Clarification of Table 3.3.6-01 Function 1 & 2 Instruments; dated September 24, 2004

CAP036685; Received Condenser Area Cooler High Inlet Temperature Alarm; dated June 4, 2005

CAP 036954; NRC Violation - Inspection Report 2005-010; dated June 29, 2005

CAP037401; Temporary Power Control Process Needs a Tie to 50.59 Screening Process; dated August 5, 2005

CAP038100; Error Found in Approved 50.59 Safety Evaluation #00-018; dated September 28, 2005

CAP 038108; 50.59/Mod Snapshot Evaluation Finding; dated September 29, 2005

CA 040624; NRC Violation - TAP - Analysis Review; dated July 26, 2005

CA 040625; NRC Violation - Inspection Support; dated July 26, 2005

CA 040626; NRC Violation - Mark I Analysis; dated July 26, 2005

CA 041038; 50.59 / Mod Snapshot Evaluation Finding; dated September 30, 2005

CE 001313 for CAP 029587; dated November 13, 2003

CE 001512; 4160 V “Prop Clearance” Measurement OOT (Spare Breaker)

CE002703; Received Condenser Area Cooler High Inlet Temperature Alarm; dated June 7, 2005

COM 040508; NRC Violation - Inspection Report 2005-010; dated July 1, 2005

PCR 041280; NRC Violation - Inspection Report; dated October 31, 2005

Calculations

434-E001; Cable Ampacity vs. Cable Tray Fill

CAL-E02-003; Single Standby Diesel Generator Static Loading for a Loss of Coolant Accident Plus a Loss Offsite Power; Revision 1

CAL-E04-010; Analysis of the Diesel Loading During a LOOP LOCA for a Chiller Load of 96KV; Revision 1

CAL-M04-011; HPCI Booster Pump Low Suction Pressure Setpoint - PS2304B; Revision 1

CAL-M97-008; HPCI NPSH Calculation; Revision 1

Drawings

BECH-E113, Sheet 79; Heating & Ventilation Systems; Revision 8

M063-002; Control Circuit Diagram for Control Room Chiller; Revision 5

Vendor Dwg. No. M073-052; RHR Pump Room Air Supply Fans IV-SF-56A & B; Revision 12

Procedures

ACP 103.2; 10 CFR 50.59 Screening Process; Revision 25

ACP 106.1; Procedure Preparation, Revision, Review, and Approval; Revision 33

ACP 106.1; Procedure Preparation, Revision, Review, and Approval; Revision 42

AOP 301.1; Station Blackout; Revision 25

AOP411; Loss of General Service Water; Revision 19

CKTBKR-G080-07; GE AM 4.16-350-2H Medium Voltage Breaker Overhaul; Revision 10

DGC-E100; Design Guide for Independence of Electrical Equipment and Circuits

GMP-ELEC-04; Cable Installation; Revision 6

SPEC-E512; Cable and Wire Installation Specification; Revision 11

STP 3.6.4.3-03; Standby Gas Treatment System HEPA and Charcoal Filter Efficiency Tests; Revision 10

Miscellaneous Documents

CWO-A65920; Feedwater Discharge to Regulating Valve CV-1621; dated May 11, 2005

NEDC-32980P; Safety Analysis Report for Duane Arnold Energy Center Extended Power Uprate; dated November 2000

NG-04-0271; Letter from Duane Arnold Energy Center to the NRC, "Additional Information Regarding NRC Unresolved Item from Safety System Design and Performance Capability Inspection"; dated June 4, 2004

PWR No. 25713; AR OTH036960; Reconcile the ACPs for 50.59 Screening Applicability; dated January 19, 2004

Work Order No. 1116829; Refurbish and Lubricate Breaker. Inspect/Calibrate prior to Returning to Service; dated March 12, 2002

Work Order No. 1126796; Rewire TS7538C per ECP1679 and Instructions; dated January 26, 2004

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EMA	Engineered Maintenance Action
IMC	Inspection Manual Chapter
IR	Inspection Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SSC	Structures Systems and Components
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item