November 21, 2007

Mr. Richard L. Anderson
Vice-President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER
NRC EVALUATION OF CHANGES, TESTS, OR
EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS
BASELINE INSPECTION REPORT 05000331/2007007(DRS)

Dear Mr. Anderson:

On October 19, 2007, 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 Energy Center. 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 October 19, 2007.

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 (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of an NCV, 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 Duane Arnold Energy Center.

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. 50-331; 72-032
License Nos. DPR-49

Enclosure: Inspection Report 05000331/2007007(DRS)
w/Attachment: Supplemental Information

cc w/encl: J. Stall, Senior Vice President, Nuclear and Chief Nuclear Officer
R. Helfrich, Senior Attorney
M. Ross, Managing Attorney
W. Webster, Vice President, Nuclear Operations
M. Warner, Vice President, Nuclear Operations Support
R. Kundalkar, Vice President, Nuclear Engineering
J. Bjorseth, Site Director
D. Curtland, Plant Manager
S. Catron, Manager, Regulatory Affairs
Chief Radiological Emergency Preparedness Section, Dept. Of Homeland Security
M. Rasmusson, State Liaison Officer

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Inspection Report to Mr. R. Anderson from Mr. D. E. Hills dated November 21, 2007

SUBJECT: DUANE ARNOLD ENERGY CENTER
NRC EVALUATION OF CHANGES, TESTS, OR
EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS
BASELINE INSPECTION REPORT 05000331/2007007(DRS)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331; 72-032
License Nos.	DPR-49
Report No:	05000331/2007007(DRS)
Licensee:	Florida Power and Light
Facility:	Duane Arnold Energy Center
Location:	Palo, IA 52354-9783
Dates:	September 24 through October 19, 2007
Inspectors:	R. Daley, Senior Reactor Inspector J. Jandovitz, Reactor Inspector V. Meghani, Reactor Inspector (In Training)
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000331/2007007(DRS); 09/24/2007 through 10/19/2007; Duane Arnold Energy Center, 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 (NCVs) and one Unresolved Item (URI) 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 an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance for the failure to translate the design bases into procedures and instructions. Specifically, the lift height limit assumed in the drop load analysis for transporting reactor vessel head stud tensioners over the refueling floor was not translated into the lift procedure allowing the licensee to potentially exceed the lift height established in the design basis calculation. This issue was entered into the licensee's corrective action program.

The issue was more than minor, because the failure to provide procedural controls for lifting of the reactor head tensioner could become a more significant safety concern. Specifically, a load drop from a higher elevation could have led to slab failure and potential damage to safe shutdown and safety related equipment on the floors below. 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, even though procedural controls were not in place to ensure that the reactor head tensioner would not be lifted above 6 feet, it could not be determined whether the head had actually ever been lifted above that threshold. (Section 1R02.1.b.1)

Green. The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," that was of very low safety significance. Specifically, MOV stroke time delays which result from Emergency Diesel Generator (EDG) voltage drops during load sequencing were not accounted for in assumed Emergency Core Cooling System (ECCS) required Motor Operated Valve (MOV) stroke times. This 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 MOV delays caused by voltage dips during ECCS load sequencing were not accounted for in the licensee's design basis and resulted in a substantive margin reduction (up to 5.3 seconds) in the ECCS injection response time. 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, even though the MOV delays were substantial and resulted in a large margin reduction, a comparison of current In-Service Testing (IST) times verses design basis maximum stroke times revealed that adequate margin still existed to meet the required ECCS response times. (Section 1R02.1.b.2)

Green. The inspections identified an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," that was of very low safety significance. Specifically, the licensee found safety related cable 1S0104-E to be severely degraded due to heat related aging and failed to initiate a corrective action document to evaluate the condition and perform an extent of condition in accordance with plant procedures. This issue was entered into the licensee's corrective action program.

The issue was more than minor because the failure to identify safety related cable failures and perform a proper extent of condition could lead to more significant safety conditions. Specifically, cables failures are adverse conditions that are primarily caused by heat induced aging. If a heat source exists, it is highly probable that other cables are adversely affected. By not writing a corrective action document and performing an extent of condition to replace damaged cables, those cables would instead fail potentially causing plant transients or a loss of mitigating 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, the cable that was degraded was replaced during the last outage and no additional cables have yet failed in the proximity of the original failed cable. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution because the licensee did not properly identify the cracked and brittle cabling through their corrective action program. (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 September 24 through October 19, 2007, the inspectors reviewed seven evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 13 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 Drop Load Evaluation for Stud Tensioner

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, a 6 foot lift height limit used in the stud tensioner drop load calculation was not translated into the tensioner strongback rigging instructions.

Description: Safety Evaluation 07-001 was used to determine the acceptability under 10 CFR 50.59 of the application of new heavier stud tensioners for the reactor vessel head. While reviewing Safety Evaluation 07-001, the inspectors identified a discrepancy between calculation CAL-273-13, which was used to support Safety Evaluation 07-001, and the field implementation for rigging reactor vessel head stud tensioners over the refueling floor. This calculation evaluated the effects of a dropped tensioner on the floor slab along the rigging path. The inspectors found that the calculation evaluated acceptability of a dropped tensioner based upon an assumption that the lift height would not exceed 6 feet. After reviewing applicable rigging procedures, the Inspectors determined that no rigging instructions or mechanical restraints were in place to insure such a limit. In fact, based upon the actual field conditions, a lift height of approximately 11 feet was possible. The inspectors considered this to be a significant discrepancy, since there was no basis for lifting the tensioner above 6 feet. Without knowing the consequences of a dropped tensioner, there was the possibility that if it were to drop, the flooring could fail causing adverse consequences to safety related equipment on the floors below. However, since no records could be found to determine the maximum lift height used in the past, the inspectors could not conclude that the 6 feet height lift had ever been exceeded.

The inspectors determined that this calculation was a part of the licensee's design basis that was not translated appropriately into the licensee's procedures and instructions, because the Duane Arnold Updated Final Safety Analysis Report (UFSAR) stated, in Section 9.1.4.4.3, that the licensee would control heavy loads, and that specific procedures would be provided for the handling of loads by the reactor building crane above the refueling floor. These procedures would include safe load paths for movement of heavy loads.

Because of this issue, the licensee initiated Corrective Action Program (CAP) 053197 to consider various options to meet their license basis requirements. These potential options included installing the necessary procedural controls, using a single failure proof strongback, or revising calculations.

Analysis: The inspectors determined that this failure to implement controls consistent with the assumptions made in the drop load analysis was a performance deficiency warranting a significance determination. The issue was more than minor, because the failure to provide procedural controls for lifting of the reactor head tensioner could become a more significant safety concern. Specifically, a load drop from a higher elevation could have led to slab failure and potential damage to safe shutdown and safety related equipment on the floors below.

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, even though procedural controls were not in place to ensure that the reactor head tensioner would not be lifted above 6 feet, it could not be determined whether the head had actually ever been lifted above that threshold. This finding did not have any cross-cutting aspects, because the failure to provide adequate procedural controls for lifting the head tensioner was an old design issue. The calculation that established the 6 foot lift threshold was developed in 1982.

Enforcement: Title 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, the design basis assumption/threshold of 6 feet that was established in calculation CAL-273-13 was not translated into the procedure used for lifting the reactor head tensioner.

Because this failure to provide adequate procedural controls for lifting the reactor head tensioner was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CAP 053197, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000331/2007007-01(DRS))

b.2 Failure to Account for Delays in ECCS MOVs Due to Voltage Dips During Load Sequencing

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, MOV stroke time delays which result from Emergency Diesel Generator (EDG) voltage drops during load sequencing were not accounted for in assumed ECCS required MOV stroke times.

Description: During review of draft Screening 7354, the inspectors determined that the proposed changes and the basis for those changes were not comprehensive. Even though Screening 7354 was only a draft, the changes to be performed were based upon the technical evaluations contained in Operability Evaluation OPR 000303. OPR 000303 evaluated the operability of the EDG and supported ECCS equipment overall response for load sequencing during a Loss of Offsite Power/Loss of Coolant Accident (LOOP/LOCA). The primary concern was voltage dips that the EDG experienced during load sequencing. As large loads sequenced onto the EDG, the voltage at the safety related 4160 VAC bus was experiencing voltage dips as low as 75 percent. The OPR performed an evaluation of the lower voltages and the effect that these would have on ECCS equipment for the LOOP/LOCA event. In regard to MOVs that needed to reposition for ECCS purposes, the licensee evaluated low voltage at the motor starters. In their Operability Evaluation, the licensee evaluated MOV delays due to this low voltage at the MOV starters and determined that even with these delays, ECCS response times for injection could still be met. However, the inspectors noted that even though the MOV starters had been evaluated, the MOVs themselves were not adequately evaluated. Specifically, MOV calculations at Duane Arnold Energy Center assume 89.9 percent of nominal voltage at the 4160 VAC buses. The inspectors were concerned, because during these periods of low voltage, MOVs that are required to reposition during a LOOP/LOCA could potentially stall. The MOVs would recover as soon as the voltage dip was over; however, during this period of low voltage, the MOVs would not run causing a delay in their opening or closing times. These delays could adversely affect ECCS response times.

Based upon this concern, the licensee initiated corrective action documents CAP 52776 and OPR 000366. In CAP 052776, the licensee evaluated these potential stall times and determined that for the "A" Division, the delay times could be as much as 5.3 seconds and for the "B" Division as much as 3.98 seconds. Reviews of the LOCA analysis determined that even with these substantial delay times, adequate margin still existed to be able to meet the Design Basis ECCS injection function. Specifically, even though the IST valve stroke times did not account for, or consider these delays, a comparison of current IST times verses design basis maximum stroke times revealed that adequate margin existed to still meet the required ECCS response times.

The inspectors determined that this failure to account for MOV stalling due to low voltages was a violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control.

Analysis: This failure to account for MOV delays caused by voltage dips during ECCS load sequencing 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 MOV delays caused by voltage dips during ECCS load sequencing were not accounted for in the licensee's design basis and resulted in a substantive margin reduction (up to 5.3 seconds) in the ECCS injection response time.

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, even though the MOV delays were substantial (5.3 seconds) and resulted in a large margin reduction, a comparison of current IST times verses design basis maximum stroke times revealed that adequate margin still existed to meet the required ECCS response times. This finding did not have any cross-cutting aspects, because the failure to account for these MOV time delays was an old design issue. While the voltage dip issue was discovered in December 2005, the issue had been a deficiency in the plant's design basis since initial startup. As a part of the corrective action document, new calculations were in the process of being created that would address the voltage dip issue. While a focused look and structured technical analysis such as a calculation should have addressed the MOV stalling issue, the corrective action document may have missed the deficiency because of the significant technical complexity of the issue.

Enforcement: Title 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, since initial plant startup, design basis information relating to MOV delays caused by voltage dips during load sequencing were not translated into and accounted for in MOV stroke times. The licensee's design basis assumption that the MOVs would receive 89.9 percent of nominal voltage at the 4160 VAC buses was not correct leading to the performance of an operability evaluation to assure that the IST valve stroke times had enough margin to still meet the required ECCS response times when adequate voltage was not available.

Because this failure to account for delays due to MOV stalling was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as AR 00668845, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000331/2007007-02(DRS)) b.3. Digital Upgrade for the Reactor Building Vent Shaft and Control Building Air Intake Radiation Monitors

Introduction: The inspectors identified an Unresolved Item involving a digital upgrade for the Reactor Building Vent Shaft and Control Building Air Intake Radiation Monitors. Specifically, the inspectors identified that potential failure modes for the digital software appeared to have not been adequately addressed in the safety evaluation performed in accordance with 10 CFR 50.59.

Description: In 2001, Duane Arnold replaced the analog Reactor Building Vent Shaft and Control Building Air Intake Radiation Monitoring systems with a new digital Sorrento radiation monitoring system. This change was evaluated by the licensee under 10 CFR 50.59. NEI 01-01/EPRI TR-102348, "Guideline on Licensing Digital Upgrades," provides NRC endorsed guidance that a licensee should use for evaluating this type of digital upgrade in accordance with 10 CFR 50.59. The licensee states in their 10 CFR 50.59 evaluation that NEI 01-01 was used as a guideline for their evaluation. However, after review of the 10 CFR 50.59 evaluation, the inspectors determined that the evaluation performed by the licensee appeared to be less than adequate. NEI 01-01 provides guidance for using a failure analysis to address potential impacts to the plant. Specifically, in reference to potential malfunctions of the equipment, it states:

> "The evaluation needs to compare results of malfunctions evaluated in the UFSAR with the results of failures that the proposed activity could create. The key issue is the effect of failures of the digital device on the system in which it is installed. The failure analysis will provide insights to system failures and their effects on Systems, Structures, and Components (SSCs)."

For digital systems, particularly with safety related applications, a Failure Modes Effects Analysis is performed to determine the potential failures that the digital software could experience. This potential failure modes analysis is usually performed by the vendor and evaluated by the licensee for adverse effects on the plant.

At Duane Arnold, the digital upgrade of the Reactor Building Vent Shaft and Control Building Air Intake Radiation monitoring systems was evaluated against the historical failure history of the digital system. While historical failure analyses may be useful to determine failures that have already occurred, they do not provide the necessary insight that is provided by a failure analysis of potential failure mechanisms, since this analysis would determine all potential failures as opposed to only failures that have actually occurred. The only potential failure analysis performed by the licensee was contained in the 10 CFR 50.59 evaluation. This failure analysis states, in part, that,

> "This comparison reveals that the failure of new components due to loss of power, a short circuit, an open circuit or loss of input signal is not any different that the failure of the existing components. Two other potential failures of concern include a common-mode software failure, i.e., a simultaneous or nearly simultaneous failure in both system trains and a processor lockup event. Consideration of these events, however, shows that no new failure modes have been created. The new monitors include a watchdog timer circuit that will place the unit in an alarming/tripped condition upon a non-self-evident failure (lockup) of the microprocessor or software. This feature combined with the fail-safe configuration of the monitor will act to

prevent a common-mode software failure from introducing a new and unanalyzed failure mode into the component."

The inspectors were concerned, because based upon review of the limited failure analysis available, and based upon discussions of the upgrade with the licensee, it appeared that the basis for acceptance was a functional failure state type analysis rather than an in-depth evaluation of the digital equipment and software. Based upon this, the inspectors asked if the vendor had performed an in-depth potential failure analysis of the digital equipment. The licensee did not know if one had been performed. Additionally, even though the licensee seemed to emphasize the importance of the watchdog circuitry in their evaluation, they were unable to address the inspectors' questions concerning potential failures of the watchdog circuitry.

Because of the complexity of the digital software and because of the need for technical assistance with the inspection of this evaluation, this issue is unresolved pending further NRC review of the modification and the 10 CFR 50.59 evaluation.
(URI 05000331/2007007-03(DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From September 24 through October 19, 2007, 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 (IP) 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 reviewed to ensure the functionality of the modification, its associated system, and any support systems. 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 Initiate a Corrective Action Document for Degraded Cabling

Introduction: The inspectors identified an NCV having very low safety significance (Green) of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, the licensee found safety related cable 1S0104-E to be severely degraded due to heat related aging and failed to initiate a corrective action document and perform an extent of condition in accordance with plant procedures.

Description: Engineering maintenance action (EMA) A77025, Revision 1, was initiated to replace cable 1S0104-E. This cable was found to be cracked and brittle during replacement of solenoid valve, SV4639, the Nitrogen Supply Isolation Solenoid Valve for

CV4639 which provides isolation for a reactor recirculation sample line. When the degraded cabling was found, the licensee initiated EMA A77025 to evaluate and initiate the work to replace the cable. The cable was located in the drywell and categorized as both safety related and EQ Class 1.

The inspectors asked to review the corrective action document associated with this adverse condition; however, the licensee determined that no corrective action document was initiated. This was contrary to the licensee's own procedure. ACP 114.5, "Action Request System," defines a Condition Adverse to Quality as follows:

> "A failure, defect, deviation, malfunction or deficiency of plant equipment, materials, procedures or personnel that has or could have an effect on the health and safety of the public, safety related equipment or maintenance rule related equipment, affects reliable operation of the station, or could adversely impact meeting NRC regulatory requirements."

Additionally, the same procedure states:

> "The Corrective Action Program (CAP) Action Request (AR) Process SHALL be used to document and track significant conditions adverse to quality (SCAQ) and condition adverse to quality (CAQ) to resolution."

The licensee did not follow these procedural requirements.

Additionally, since no corrective action document was generated, there was no extent of condition. This is important for a degraded cable condition, because the majority of cracked and brittle cable conditions are caused by heat aging. If there is excessive heat in the area, the condition could adversely affect other electrical cabling and equipment.

Because of this issue, the licensee issued corrective action document CAP 053115. Additionally, the licensee determined that the cause for the damaged cable was most probably due to proximity of the cable to a heat source. The licensee also looked at drawings to see if other equipment could have been affected and determined that it was most likely that only this cable was affected; however, since the cabling was located in the drywell, a definitive extent of condition could not be performed at the time of the inspection.

The inspectors determined that this failure to initiate a corrective action document for a condition adverse to quality was a violation of 10 CFR Part 50, Appendix B, Criterion V, Instructions, procedures, and Drawings.

<u>Analysis</u>: The failure to initiate a corrective action document for this condition adverse to quality was a performance deficiency warranting a significance determination. The issue was more than minor because the failure to identify safety related cable failures and perform a proper extent of condition could lead to more significant safety conditions. Specifically, cables failures are adverse conditions that are primarily caused by heat induced aging. If a heat source exists, it is possible that other cables are adversely affected. By not writing a corrective action document and performing an extent of condition to replace damaged cables, those cables would instead fail potentially causing plant transients or a loss of mitigating 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, no additional cables have yet failed in the proximity of the original failed cable. The primary cause of this finding was related to the cross-cutting area of problem identification and resolution because the licensee did not properly identify the cracked and brittle cabling through their corrective action program.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, in March 2007 the licensee failed to accomplish activities in accordance with their procedure, ACP 114.5, Action Request System, when a degraded safety related cabling in the drywell was discovered. Specifically, this condition adverse to quality was not entered into the corrective action program as required by their instructions.

Because this failure to enter this condition adverse to quality into the licensee's corrective action program was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CAP 053115, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000331/2007007-03(DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From September 24 through October 19, 2007, the inspectors reviewed six Corrective Action Process (CAP) 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 (CA) 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 Meeting(s)

.1 Interim Exit Meeting

The inspectors presented the inspection results to Mr. J. Bjorseth and others of the licensee's staff, on October 19, 2007. 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
R. Bierman, Electrical/I&C Design Engineering Supervisor
J. Cadogan, Engineering Director
M. Lingenfelter, Design Engineering Manager
R. Murrell, Licensing Engineer
J. Swales, Mechanical Design Supervisor
L. Swenzinski, Licensing Engineer

Nuclear Regulatory Commission
R. Baker, Resident Inspector
K. Feintuch, NRR Project Manager
D. Hills, EB1 Branch Chief
R. Orlikowski, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000331/2007007-03	URI	Digital Upgrade for the Reactor Building Vent Shaft and Control Building Air Intake Radiation Monitors

Opened and Closed

05000331/2007007-01	NCV	Drop Load Evaluation for Stud Tensioner
05000331/2007007-02	NCV	Failure to Account for Delays in ECCS MOVs Due to Voltage Dips during Load Sequencing
05000331/2007007-04	NCV	Failure to Initiate a Corrective Action Document for Degraded Cabling

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.

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

10 CFR 50.59 Screenings

Screening 4271; FSAR 038367, CE1712, UCR 04-013; dated September 27, 2005

Screening 5476; ARP IC04B; PWR 29509; dated September 06, 2005

Screening 5572; PWR30407 and AOP 639; dated September 30, 2005

Screening 5579; OI152A2 - PWR30408/OI 537A2 - PWR 30409; dated October 04, 2005

Screening 5893; EMA A73279; dated April 4, 2006

Screening 5919; DDC 4867; dated February 10, 2006

Screening 5922; UCR 60-010 replace references to outdated methodology for the SLC shutdown margin; dated February 13, 2006

Screening 6065; EMA A70751 Replace SV7615, 7614 and 7613 - Reactor Building Exhaust Fan Isolation Damper Solenoids; dated March 23, 2006

Screening 6098; EMA A71996; dated April 01, 2006

Screening 6301; Change to UFSAR Table 9.2-1; dated May 18, 2006

Screening 6737; Change to UFSAR CST Low Level Swap Values; dated September 27, 2007

Screening 6781; CA 043854, UFSAR Change Request No. 2006-027; dated April 18, 2007

Screening 6834; ECP 1781 Drywell Penetration Support Modifications; dated November 2, 2005

10 CFR 50.59 Evaluations

00-014; ECP 1630 Penetration Upgrade/Replacement; Revision 1

01-025; Evaluation of ECP 1628, Reactor Building Vent Shaft and Control Building Air Intake Radiation Monitors Replacement; Revision 0

05-001; Decommission Fuel Oil Tanks Inside Protected Area; dated August 30, 2005

06-001; ECP 1720 SLDS Riley Module Replacement; Revision 1

06-003; Change to Procedure AOP 301.1, “Station Blackout”; Revision 0

06-004; ECP 1787 HPCL High Pressure Keep Fill; dated December 22, 2006

07-001; RFPs 110 and 210, RPV Disassembly/Reassembly; dated January 30, 2007

IR17 Permanent Plant Modifications (71111.17B)

Modifications

ECP 1665; Control Building Chiller 1VCH001A/B Modification; dated June 15; 2007

ECP 1726; FIC5828A/B Control Loop Replacement; Revision 0

ECP 1736; 1A4 Appendix R Issue; Revision 0

ECP 1781; Evaluation of Specific Drywell Penetrations and Connected Piping/Tubing Inside and Outside the Drywell for Drywell Thermal Movements; Dead Weight and Seismic Loading; dated March 13, 2007

ECP-1793; Replace Filter Elements of RHRSW Strainers; dated May 04, 2007

ECP 1797; HPCL High Pressure Keep Fill System; dated December 22, 2006

ECP-1803; Pipe Support Modification of the Feedwater and HPCL Piping Outside the Drywell; Revision 2

EMA A77025; Replacement of Cable 1S0104-E for SV4639; Revision 1

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CAP 052774; Update ACP 013.2 Transition Rules to Match RIS 2001-0; dated September 26, 2007

CAP 052776; Potential MOV Stroke Delay Times Are not Accounted for; dated September 26, 2007

CAP 052791; PWR 36191 Has Incorrect Description for RFP 110 Revision 6 Changes; dated September 27, 2007

CAP 052793; Inadequate Design Documentation for ECP 1736; dated September 27, 2007

CAP 052798; 50.59 Screen for EMA A73279 – No Justification for Why Change is not Adverse; dated September 27, 2007

CAP 052801; Review Local 50.59 Screening Practices as Applied to Surveillance Procedures; dated September 27, 2007

CAP 052817; NCAQ – Scaffolding Erected too Close to RCIC Discharge Piping; dated September 28, 2007

CAP 053170; CAL-E95-006 Requires Clarification; dated October 15, 2007

CAP 053187; Inadequate Labeling on 1C496 and 1C497; dated October 16, 2007

CAP 053189; Screening 6065 Needs Revision; dated October 16, 2007

CAP 053190; EMA-A73279 Does not Contain an Evaluation of the Weight Added by the Change; dated October 16, 2007

CAP 053197; CAQ – No Procedural or Mechanical Control to Limit Stud Tensioner Max. Lift Height; dated October 16, 2007

CAP 053205; NCR Question 10 CFR 50.59 Part 2 for Screening 6834; dated October 16, 2007

CAP 053208; CAP 052817 Did not Consider Past Operability or Extent of Condition Aspects; dated October 16, 2007

CAP 053215; Screenings Do not Have Adequate Descriptions; dated October 17, 2007

CAP 053220; ASME Section XI Requirements for EBB-5 Piping Design Pressure Rerate not Met; dated October 17, 2007

CAP 053232; NCAQ – Remove Cross Hatch from 4 inch HBC-113 Line on Drawing BECH-HLR-M271; dated October 17, 2007

CAP 053240; Potential Issue with 50.59 Evaluation 01-025 FMEA; dated October 17, 2007

CAP 053260; Missing Weld Checklist in Final Documentation Package; dated October 18, 2007

CAP 053266; MT Examination Improperly Identified; October 18, 2007

CAP 053271; Drawing. BECH-M005 Shows Incorrect Position for Stud Tensioner and Head Strongback; dated October 18, 2007

OPR 000366; Engineering to Perform Evaluation to Address Impact on Affected MOV Stroke Times; dated September 26, 2007

Corrective Action Program Documents Reviewed During the Inspection

AR 16214; Failure Investigation for Electrical Shorting Found Between Two Conductors in DAEC Electrical Penetration 1JX105C

CAP 032687; GE SC04-11, Narrow Range Water Level Instrument Level 3 Trip; dated August 18, 2004

CAP 040419; ECP 1726 Screening Needs Revision; dated February 16, 2006

CAP 043569; Unplanned LCO Entry, 1P022D INOP Due to Loss in Upper Sight-Glass; dated June 30, 2006

CAP 043731; UFSAR Table 90.2-1 Is Inconsistent and Inaccurate; dated August 18, 2006

OTH 005061; Perform Equipment Failure Analysis of Electrical Penetrations Removed during RF; July 20, 1999

OTHO20716; Evaluate “hole size” in RHR SW Pumps Discharge Strainers. August 13, 2002

Calculations

CAL-273-13; USNRC-NUREG 0612 “Control of Heavy Loads at Nuclear Power Plants”; dated November 8, 1982

CAL-E92-010; Low Reactor Water Level Scram and ADS Confirmatory Setpoints, LIS4592A, B, C, and D and LIS4561 and 4562; Revision 5

CAL-E93-032; Temperature Transient Evaluation for HPCI Room during Station Blackout; Revision 1

CAL-E93-033; Temperature Transient Evaluation for RCIC Room during Station Blackout; Revision 1

CAL-M07-017; EBB-5 Piping Design Pressure ReRate Between V-23-0081 and M02312; dated June 25, 2007

CAL-M05-043; Evaluation of Drywell Well Water Supply Piping at Penetration X-24A Outside Drywell - HLE-34, JBD-28 and JBD-34; dated May 17, 2007

Drawings

Drawing M123-028; 2”-1978 Reduced Ports Socket Ends Stainless Steel Piston Check Valve; Revision 1

Drawing M304-037; RHR SW Strainer Element; Revision 0

Procedures

ACP 1408.2; Scaffold Control; Revision 22

GMP-CNST-09; Scaffolding; Revision 20

OI 730; Control Building HVAC System; Revision 92

STP 3.3.5.1-15; RHR Logic System Functional Test; Revision 6

STP 3.3.5.1-37; RHR LSFT – Operating; Revision 0

STP 3.7.5-01; Control Building Chiller Operability – 'B' Chiller; dated July 24, 2007

STP 3.7.5-01; Control Building Chiller Operability – 'A' Chiller; dated July 20, 2007

Miscellaneous Documents

Data Report for RHR 'B' Heat Transfer Test; dated January 09, 20/07

Data Report for RHR 'A' Heat Transfer Test; dated December 14, 2006

MFN 04-110; Part 21 Final Report: Narrow Range Water Level Instrument Level 3 Trip; dated October 11, 2004

PWR 31442; Revision to STP 3.5.1-06 To Delete Stroke Time Requirements; dated December 2, 2005

PWR 33306; Revision to STP 3.3.1.1-13 To Incorporate Use of RPS Test Box; dated April 20, 2006

PWR 36127; Revision of STP 3.3.1.1-22 For Installation Of Jumper To Bypass PS 4550 A2; dated December 22, 2006

PWR 36201; Revise RHR LSFT STP 3.3.5.1-15 to Split Out Sections That Can Be Done Online and Put Them Into New STP 3.3.5.1-37, RHR LSFT – Operating; dated January 3, 2007

PWR 37463; Revision to STP 3.5.1-05 To Delete Stroke Time Requirements; dated May 9, 2007

SC04-14; 10 CFR Part 21 Communication, Narrow Range Water Level Instrument Level 3 Trip Final Report; dated October 11, 2004

Screening 1876; EMA A588898/EMA A58899; dated January 10, 2003

Screening 2646; ECP 1665/ECN 1665-05 CB Chiller TCV 6924A/B; dated April 27, 2005

Screening 6838; Change to AOP 301.1 under PWR 35186; dated October 13, 2006

Screening 7156; Change to STP 3.0.0.01 under PWR 36509; dated February 4, 2007

Section XI Repair/Replacement Plan for EBB-5-SR-12; dated January 30, 2007

Work Order 1119092; GL 89-13 Program. Clean Tubes And Inspect Internals – RHR SW Heat Exchanger B; dated April 08, 2003

Work Order 1128081; GL 89-13 Program. Clean Tubes And Inspect Internals – RHR SW Heat Exchanger A; dated April 16, 2005

Work Order 1138179; Install New Support Per Drawing M119AC-11867; dated February 26, 2007

VT-07-112; NDE Data Sheet for Support EBB-5-SR-12; dated March 16, 2007

Data Report for RHR 'B' Heat Transfer Test; dated January 09, 20/07

Data Report for RHR 'A' Heat Transfer Test; dated December 14, 2006

50.59 Evaluation 02-002; Operation with RHR SW Strainers Bypassed; dated August 08, 2002

EMA A71996; Equivalency Change Evaluation; dated September 19, 2006

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
AR	Action Request
CAP	Corrective Action Program
CAQ	Condition Adverse to Quality
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EMA	Engineering Maintenance Action
EQ	Environmental Qualification
IMC	Inspection Manual Chapter
IR	Inspection Report
IST	In-Service Testing
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SSC	System, Structure, and Component
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item