June 27, 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 ENERGY CENTER
NRC INSPECTION REPORT 05000331/2005010(DRS) AND NOTICE OF VIOLATION

Dear Mr. Van Middlesworth:

On June 7, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Duane Arnold Energy Center. The enclosed inspection report documents the inspection findings which were discussed on June 7, 2005, with you and other members of your staff.

The inspection 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 purpose of this inspection was to follow up on an unresolved item from the safety system design and performance capability inspection conducted from January 26 to February 16, 2004. Within these areas, the inspection consisted of discussions with personnel and selected examination of procedures and representative records.

Based on the results of this inspection, the NRC has identified a violation of very low safety significance which is cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection report. The violation was evaluated in accordance with the NRC current Enforcement Policy which is included on the NRC's Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Enforcement Policy**. The violation is being cited in the notice because it had not been entered into your corrective action program and no actions to restore compliance have been taken.

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

Additionally, the NRC identified a second finding, which also involved a violation and was evaluated under the risk significance determination process. Because this second violation was of very low safety significance and because it was entered into the licensee's corrective action program, the NRC is treating this finding as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. It is also described in the subject inspection report. If you contest 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region III, Resident Inspector and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures and your response 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). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Cynthia D. Pederson, Director
Division of Reactor Safety

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

Enclosures: 1. Notice of Violation
2. Inspection Report 05000331/2005010(DRS)
w/Attachment: Supplemental Information

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. McGhee - Department of Public Health

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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. McGhee - Department of Public Health

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NOTICE OF VIOLATION

Nuclear Management Company, LLC | Docket No. 50-331
Duane Arnold Energy Center | License No. DPR-49

During an NRC inspection conducted on January 25 through June 7, 2005, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600 (Enforcement Policy), the violation is listed below:

> Criterion III, "Design Control," of 10 CFR Part 50, Appendix B, requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis, for those systems, structures and components for which this appendix applies, are correctly translated into specifications, drawings, procedures and instructions. It further requires that the design control measures provide for verifying or checking the adequacy of the design. Finally, it requires that design changes be subject to design control measures commensurate with those applied to the original design.
>
> Contrary to the above, in October 1996, the licensee installed a design change to the high pressure coolant injection system that was not subject to the same design control measures as the original design. The high pressure coolant injection system is a safety-related system which is governed by the requirements of 10 CFR Part 50, Appendix B, Criterion III. Specifically, the design change incorporated the results of a calculation regarding the acceptability of moving steam discharge check valve V22-0016 approximately 50 feet closer to the torus on the torus attached piping loads. The calculation used a simplified methodology which contained an assumption that frequency response changes within ten percent were insignificant. The original methodology, which was specifically reviewed and approved by the NRC, did not contain such an assumption. Furthermore, the licensee did not implement any measures to verify the adequacy of this design assumption or to otherwise show that the design basis for torus attached piping had been correctly translated into the modification's specifications, drawings, procedures and instructions.

This violation is associated with a Green significance determination process (SDP) finding.

Pursuant to the provisions of 10 CFR 2.201, Nuclear Management Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Duane Arnold Energy Center, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level; (2) the corrective steps that have been taken and the results achieved; (3) the corrective steps that will be taken to avoid further violations; and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

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

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 27th day of June, 2005

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No:	05000331/2005010(DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Duane Arnold Energy Center
Location:	3277 DAEC Road Palo, Iowa 52324-9785
Dates:	January 25 through June 7, 2005
Inspector:	P. Lougheed, Senior Engineering Inspector
Approved by:	A. M. Stone, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000331/2005010(DRS); 01/25/2005 - 06/07/2005; Duane Arnold Energy Center; Safety System Design and Performance Capability Inspection

This report follows up on two unresolved items from the Safety System Design and Performance Capability Inspection conducted from January 26 through February 13, 2004. The inspection was conducted by a Region III reactor inspector. The inspection identified two Green findings, one of which involved a Cited Violation (NOV) and the other a Non-Cited Violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after Nuclear Regulatory Commission (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. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- Green. A violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance was identified by the inspector. Specifically, the licensee failed to demonstrate that a 1996 high pressure coolant injection (HPCI) modification was subjected to design control measures commensurate with those applied to the original design. The licensee also failed to apply design control measures to verify the adequacy of the design in order to assure that the design basis for torus attached piping was correctly translated into the modification's specifications, drawings, procedures and instructions.

  The finding was more than minor because the finding was associated with the cornerstone attribute of design control in the mitigating system cornerstone and the finding was determined to affect the associated cornerstone objective of ensuring the availability of the HPCI system when called upon. Under the worst case scenario, movement of the torus with the additional valve weight on the HPCI turbine exhaust line would result in crimping of the line. Crimping of the line would create additional backpressure in the HPCI turbine and would result in a decrease in the amount of water being injected into the reactor vessel. The finding was determined to be of very low safety significance based upon a Phase 2 analysis of those transients which would involve movement of the torus.

- Green. A Non-Cited Violation of Technical Specification 3.5.1 having very low safety significance was identified. Specifically, the licensee failed to ensure that the HPCI discharge line was filled with water from the pump discharge valve to the injection valve as required by Technical Specification surveillance 3.5.1.1. The issue is considered NRC identified because the licensee vented the system in response to an NRC unresolved item from the safety system design and performance capability inspection and had not otherwise planned to vent the system. As corrective action, the licensee planned to vent the system on a periodic basis.

The finding was more than minor because the finding could reasonably be viewed as a precursor to a significant event, specifically a hydraulic transient of the HPCI system when called upon to inject. The finding was determined to be of very low safety significance based upon a Phase 2 analysis of those transients where HPCI was required to operate.

B. Licensee-Identified Violations

None

REPORT DETAILS

1R21 Safety System Design and Performance Capability

a. Inspection Scope

The inspector reviewed actions taken to resolve two unresolved items identified during the 2004 safety system design and performance capability (SSDPC) inspection. The issues had been held as unresolved items pending sufficient additional information from the licensee to determine if violations existed.

This inspection did not represent an inspection sample.

b. Findings

.1 Torus Attached Piping

Introduction: The inspector identified a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” Specifically, a design change to the high pressure coolant injection turbine exhaust subsystem was not subject to design control measures commensurate with those applied to the original design and the licensee did not implement any measures to verify the adequacy of the design assumption which differed from that applied to the original design.

Description: During the 2004 SSDPC inspection, an unresolved item was opened regarding a 1996 high pressure coolant injection (HPCI) modification. Specifically, on July 12, 1996, the licensee approved modification ECP 1575 which allowed valve V22-0016 to be moved approximately 50 feet closer to the torus in order to decrease the potential to siphon suppression pool water into the HPCI turbine exhaust line due to steam condensation. The HPCI turbine exhaust line penetrated primary containment at torus penetration N214, and terminated inside the torus below the suppression pool water level. This modification was later field completed in October 1996.

During the 2004 inspection, the NRC determined that the 1996 calculation performed by the licensee did not use the same methodology as that previously reviewed and accepted by the NRC in the 1983 Duane Arnold plant unique analysis report for the Mark I containment. The original methodology was a detailed analysis of the reaction of torus attached piping to loss of coolant accident (LOCA) and safety relief valve (SRV) discharge loads. The revised analysis included an assumption that frequency response changes that were less than 10 percent were insignificant and would not affect the system acceleration response for the modified piping. However, based upon review of the original analyses, the inspector determined that the response spectra for LOCA and SRV discharge loads typically had sharp peaks at narrow critical frequency ranges. Therefore, the inspector determined that small changes in the piping system resonant frequency near LOCA or SRV discharge load critical frequencies could produce large changes in the piping system analysis results. Based on this determination, the inspector could not verify that the licensee’s assumption was valid. The inspector considered this verification to be critical as the piping stress level was at 90 percent of design acceptance limits, and the pipe support allowable reactions were at 94 percent of design acceptance limits. This item was identified as an unresolved item in inspection report 05000331/2004006, pending sufficient additional information from the licensee to verify the adequacy of the design assumption.

On January 25, 2005, the licensee provided a copy of a letter from a consultant to the inspector. The licensee viewed the letter as providing justification that the simplified methodology was acceptable, as the consultant stated that it viewed the calculation as a "competent and reasonable approach" to addressing Mark I containment load. The letter stated, "More specifically, given the sensitivity of the LOCA [loss of coolant accident] and SRV [safety relief valve] related loads to the modal characteristics of the model, using a comparison of the original and as-modified piping system frequencies and the application of unit acceleration cases to quantify mass re-distribution as a method for estimating the modified system response to the LOCA and SRV related loads is considered a reasonable *alternative* approach" [emphasis in original]. The consultant continued its letter by recommending additional comparisons that could be performed to provide more justification that the alternative methodology was acceptable and concluded by stating that "If the commission is not receptive to an alternative approach, no amount of additional justification will suffice."

The inspector noted that the letter supplied by the licensee supported the inspector' original concern by reiterating the sensitivity of the LOCA and SRV related loads to the modal characteristics of the model. However, the letter did not provide any information that verified the adequacy of the design assumption about the acceptability of ignoring low level frequency response changes. Therefore, the inspector concluded that the licensee had not shown that the overall change in the analyses results and conclusions would be insignificant if the resultant changes in the frequency response were less than ten percent. The inspector concluded that the licensee had not demonstrated that the design basis for torus attached piping had been correctly translated into the design change. The inspector further concluded that the licensee did not verify the adequacy of the new design as it did not provide any information to show that its design assumption was valid.

<u>Analysis</u>: The inspector determined that a performance deficiency existed because the licensee failed to show that the design control measures applied to modification ECP 1575 were commensurate with the original design. Furthermore, the inspector determined that it was reasonably within the licensee's control to have identified that the simplified methodology needed to be verified in order to show that the design requirements for torus attached piping were correctly translated into the modification specifications, drawings, procedures and instructions.

The inspector determined that the finding did not specifically impact the NRC's ability to perform its regulatory function. However, in a telephone call on February 22, 2005, the licensee acknowledged that using the simplified methodology in the future would require a specific license amendment from the NRC due to the revision in the requirements of 10 CFR 50.59 since 1996.

The inspector reviewed the issues described in Appendix E to Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," and determined that none of them were applicable to this case. Using the criteria in Appendix B to MC 0612, the finding was determined to be more than minor because the finding was associated with the cornerstone attribute of design control in the mitigating system cornerstone and the finding was determined to affect the associated cornerstone objective of ensuring the availability of the HPCI system when called upon. Under the worst case scenario, movement of the torus with the additional valve weight on the HPCI turbine exhaust line would result in crimping of the line. This would cause an increased back pressure in the turbine exhaust line which would affect the ability of the HPCI system to inject into the reactor vessel.

The inspector also evaluated whether the finding would be associated with the barrier integrity cornerstone as the finding involved piping attached to the torus. The inspector determined that movement of the torus and the attached modified piping line would cause additional stresses on flanged joints such that they might leak. However, the only flanged joint in the line was downstream of the single containment isolation valve. Assuming that the containment isolation valve would continue to perform its safety function, the inspector determined that it was unlikely that the containment barrier would be breached. Therefore, the inspector determined that the barrier integrity cornerstone was not affected.

The inspector entered Phase 1 of the Significance Determination Process (SDP), in accordance with the guidance in IMC 0609, "Significance Determination Process," Appendix A. The Phase 1 analysis asks five questions in order to perform an initial screen of the finding. The inspector determined that, based on the lack of analysis, the HPCI system could not be shown to be within design limits for all accident scenarios such that the operability of the HPCI system could not be proven. Therefore, the inspector continued in Phase 1. The next question was if the finding represented a loss of system safety function. Again, based on the lack of analysis which could positively demonstrate that the HPCI system would perform its safety function under all accident scenarios, the inspector answered this question positively. Answering this question positively resulted in the inspector being sent to Phase 2 of the SDP.

In performing a Phase 2 analysis, the inspector determined that the only transients affected would be those which would reasonably involve significant movement of the torus; these transients were determined to be the medium and large loss of coolant accidents and an anticipated transient without scram. The inspector further determined that the HPCI system was not required to operate for the large break loss of coolant accident. For the remaining two transients, the inspector completed the Phase 2 worksheets and determined that loss of the HPCI system would be of very low risk significance (Green).

Enforcement: Criterion III, "Design Control," of 10 CFR Part 50, Appendix B, requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis, for those systems, structures and components for which this appendix applies, are correctly translated into specifications, drawings, procedures and instructions. It further requires that the design control measures provide for verifying or checking the adequacy of the design. Finally, it requires that design changes be subject to design control measures commensurate with those applied to the original design.

Contrary to the above, on July 12, 1996, the licensee approved a modification which allowed valve V22-0016 to be moved closer to the torus. To support this design change, the licensee performed a calculation which was not held to the same design control measures as that of the original design. Specifically, the calculation contained an assumption that frequency response changes within ten percent were insignificant. The original methodology, which was specifically reviewed and approved by the NRC, did not contain such an assumption. Furthermore, the licensee did not implement any measures to verify the adequacy of this design assumption or to otherwise show that the design basis for torus attached piping had been correctly translated into the design.

As of June 7, 2005, this condition adverse to quality had not been entered into the licensee's corrective action program and had not been corrected.
(NOV 05000331/2005010-01)

.2 HPCI Injection Piping Hydraulic Transient Susceptibility

Introduction: The inspector identified a finding of very low significance involving a Non-Cited Violation of Technical Specification 3.5.1. Specifically, the licensee failed to ensure that the HPCI discharge line was filled with water from the pump discharge valve to the injection valve as required by Technical Specification Surveillance 3.5.1.1.

Description: During the 2004 SSDPC inspection, an unresolved item was identified regarding the potential for a hydraulic transient to occur in the HPCI discharge piping line when the system was called upon to operate. The unresolved item documented a concern that conditions appeared to exist in the Duane Arnold HPCI discharge piping which mirrored conditions at the Dresden Station, where a hydraulic transient actually occurred. The item was left unresolved pending the Duane Arnold licensee providing information as to why the similar conditions at Duane Arnold would not result in a hydraulic transient, or performance of an analysis which would show that HPCI system operation would not be affected if a hydraulic transient occurred.

One of the issues pertinent to the issue was the presence of air at the high point in the HPCI discharge piping. Technical specification surveillance, SR 3.5.1.1, required verification that the piping was filled with water from the pump discharge valve to the injection valve once every 31 days. The licensee considered this surveillance requirement to be met as long as the HPCI system was properly vented after any system drainage, was aligned to the condensate storage tank (CST), or was aligned to the torus with the keepfill system in operation. Therefore, the licensee did not normally vent the HPCI discharge piping to verify that the line was filled with water.

On March 28, 2005, in an effort to resolve the URI from the SSDPC, the licensee opened the vent valve at the high point of the HPCI system. With the 3/4-inch valve opened approximately 1/8 turn, air was discharged from the system for approximately 12 seconds. The licensee did not precisely measure the amount the valve was open or the time that the air escaped; therefore, the amount of air present in the system could not be quantified. Informal calculations performed by the licensee indicated that the air pocket could have ranged from only filling the vent piping to occupying a 28-inch high segment of the pipe. While the licensee believed the former measurement was more likely, there was insufficient information for the NRC to draw a similar conclusion.

Analysis: The inspector determined that a performance deficiency existed because the licensee failed to verify at least once every 31 days that the HPCI system was filled with water from the pump discharge valve to the injection valve as required by surveillance requirement 3.5.1.1. This was based upon the licensee finding air in the system when it had previously been properly vented, had not been drained, and had been either aligned to the CST or to the torus with the keepfill system in operation. Furthermore, it was reasonably within the licensee's control to have identified that keeping the HPCI system aligned to the CST or having the keepfill system in operation was insufficient to ensure that the system was water filled. This was based not only on the licensee receiving operating experience which showed that air could come out of solution, most recently regarding a July 2001 Dresden hydraulic transient in the HPCI system, but also on an informal survey performed by the licensee, which showed that Duane Arnold was an outlier in the industry by not routinely venting the HPCI discharge piping.

The inspector determined that the finding did not impact the NRC's ability to perform its regulatory function. The inspector reviewed the issues described in IMC 0612, Appendix E, and determined that none of them were applicable to this case. Using the criteria in Appendix B to MC 0612, the finding was determined to be more than minor because the finding could reasonably be viewed as a precursor to a significant event. Specifically, the Duane Arnold HPCI discharge piping might be susceptible to a hydraulic transient similar to the one experienced by the Dresden Station in July 2001, based on the similarities between the Dresden and Duane Arnold HPCI piping configurations.

The inspector entered Phase 1 of the SDP and determined that, based on the licensee's inability to quantify the size of the air bubble plus the lack of a hydraulic transient analysis, that the HPCI system could not be shown to be operable under all accident scenarios. Therefore, the inspector continued in Phase 1. The next question was if the finding represented a loss of system safety function. Again, based on the lack of analysis which could positively demonstrate that the HPCI system would perform its safety function under all accident scenarios, the inspector answered this question positively. Answering this question positively resulted in the inspector being sent to Phase 2 of the SDP.

In performing a Phase 2 analysis, the inspector determined the initiating event sequences which were affected. The inspector also determined that the initiating event likelihood should be based on a time duration of less than three days. This determination was made based upon the assumption that the hydraulic transient would not occur until HPCI was called upon to function in response to the initiating event. Based on this assumption, the inspector completed the Phase 2 worksheets and determined that loss of the HPCI system would be of very low risk significance (Green).

Enforcement: Technical Specification Surveillance requirement 3.5.1.1 requires the licensee to verify, for each emergency core cooling system injection or spray subsystems, that the piping is filled with water from the pump discharge valve to the injection valve once every 31 days while the reactor is critical.

Contrary to the above, from approximately April 19, 2003, when the reactor was taken critical, until March 28, 2005, when the plant was shut down for its latest refueling outage, the licensee failed to verify that the HPCI system, an emergency core cooling injection subsystem, was filled with water from the pump discharge valve to the injection valve every 31 days. Instead the licensee relied upon the fact that the HPCI system was properly vented following the last time the system was drained and that the system was either aligned to the CST or to the torus with the keepfill system operating. However, on March 28, these methods were shown to be insufficient to verify that the HPCI system was filled with water as air flowed from the system for approximately 12 seconds when the licensee opened the high point vent valve by approximately 1/8 of a turn.

The licensee entered this issue into their corrective action system as condition report CA 040204 and decided to perform periodic venting of the HPCI discharge line. This finding is considered NRC identified because the licensee had not planned to vent the line prior to NRC raising the concern in the unresolved item. Because the licensee entered the violation into their corrective action system and took corrective actions, this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000331/2005010-02).

4. OTHER ACTIVITIES (OA)

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000331/2004006-03: HPCI Pump Discharge Piping Hydraulic Transient Susceptibility

This issue is discussed in Section 1R21.b.2 above. The unresolved item is closed.

.2 (Closed) Unresolved Item 05000331/2004006-04: Unverified Methodology for Analysis of Torus Attached Piping

This issue is discussed in Section 1R21.b.1 above. The unresolved item is closed.

4OA6 Meetings

.1 Exit Meeting

The inspector presented the inspection results to Mr. J. Bjorseth and other members of licensee management on June 7, 2005. The licensee acknowledged the findings presented. The licensee also confirmed that they had not entered the torus attached piping issue into the corrective action program and had not taken any corrective actions for this issue. The inspector asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
John Bjorseth, Site Director
Steve Catron, Regulatory Affairs Manager
Dean Curtland, Plant Manager
Wayne Bentley, Assistant Operations Manager (Acting Operations Manager)
Steve Huebsch, Engineering Supervisor (Mechanical)

Nuclear Regulatory Commission
A.M. Stone, Chief, Engineering Branch 2, Division of Reactor Safety
B. Burgess, Chief, Branch 2, Division of Reactor Projects
G. Wilson, Senior Resident Inspector
R. Baker, Resident Inspector
J. Neurauter, Engineering Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000331/2005010-01	NOV	Failure to Demonstrate Adequacy of Design Assumption for Torus Attached Piping
05000331/2005010-02	NCV	Failure to Properly Vent HPCI Pump Discharge Piping

Closed

05000331/2004006-03	URI	HPCI Pump Discharge Piping Hydraulic Transient Susceptibility
05000331/2004006-04	URI	Unverified Methodology for Analysis of Torus Attached Piping
05000331/2005010-02	NCV	Failure to Properly Vent HPCI Pump Discharge Piping

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

CA040204; Revision of OP-20 to Require Venting; dated May 10, 2005

CAP 030715; Reevaluate OE 16542 Evaluation – Venting HPCI Discharge Piping; dated February 13, 2004

COM004840; Vent the HPCI Pump Discharge Line; dated March 7, 2005

FSK-4813; Isometric – Steam Tunnel Areas 4 and 5 – 1"-EBB-5 & 4"-EBB-1 “Test” Vent Piping; Revision 3

ISO-DLA-001-01; Isometric – Turbine Building Feedwater System; Revision 5

ISO-EBB-005-01; Isometric – HPCI Pump Discharge; Revision 0

ISO-HBB-006-01 and -02; Isometrics – HPCI Turbine Steam Exhaust; Revision 1

Letter from DAEC to the NRC; Torus Attached Piping and Suppression Chamber Penetrations Analysis; Volume 6; Revision 0; dated June 17, 1983

TNV-04-022; Letter from Automated Engineering Services Corporation to Duane Arnold Energy Center; Brief Review of DAEC Calculation CAL-M96-010; dated November 15, 2004

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DAEC	Duane Arnold Energy Center
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
LOCA	Loss of Coolant Accident
NCV	Non-Cited Violation
NOV	Notice of Violation
NRC	Nuclear Regulatory Commission
SDP	Significance Determination Process
SRV	Safety Relief Valve
SSDPC	Safety System Design and Performance Capability
URI	Unresolved Item