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John F. Franz, Jr.
Vice President, Nuclear

September 27, 1996
NG-96-1968

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-37
Washington, D.C. 20555-0001

Subject: Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49
Reply to a Notice of Violation Transmitted with Inspection Report 96005

File: A-105, A-102

Dear Sir:

This letter and attachment are provided in response to the Notice of Violation transmitted with NRC Inspection Report 96005.

This letter contains the following new commitments.

Present the issue concerning the inadequate safety evaluation for Emergency Service Water makeup to the Spent Fuel Pool to plant management and the technical staff as part of continuing Engineering Support Training by March 30, 1997.

Complete the Quality Assurance department review of the identified procedural deficiencies, those previously identified in past inspections, and those identified internally for commonalties and initiate corrective actions as necessary by November 1, 1996.

Update the permanently installed nameplate labels on the SBDG to match the wording in OI 324 (SBDG), Section 5.2, by December 21, 1996.

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If you have any questions regarding this matter, please contact my office.

Sincerely,

John F. Franz
Vice President, Nuclear

Attachment: Reply to a Notice of Violation Transmitted with Inspection Report 96005

cc: R. Murrell
L. Liu
G. Kelly (NRC-NRR)
H. Miller (Region III)
NRC Resident Office
DOCU

IES Utilities Inc.
Reply to a Notice of Violation
Transmitted with Inspection Report 96005

VIOLATION ONE

Part 50.59 of 10 CFR requires, in part, that records of changes to the facility include a written safety evaluation which provides the bases for the determination that the change did not involve an unreviewed safety question.

Contrary to the above, Safety Evaluation No. 95-06, written to support the low Emergency Service Water (ESW) makeup flow rate to the Spent Fuel Pool (SFP), identified on February 20, 1995, assumed a value for heatup rate that was much lower than published data for the SFP configuration in place during refuel outage 13, which began February 23, 1995. Therefore, the safety evaluation did not provide sufficient bases to determine if the condition may have involved an unreviewed safety question.

This is a Severity Level IV violation (Supplement 1).

RESPONSE TO VIOLATION ONE

1. REASON FOR THE VIOLATION

Section 9.1.3.3 of the Updated Final Safety Analysis Report (UFSAR) states that the ESW system provides Seismic Category I makeup capability to the SFP in the event that all external cooling to the pool is lost. The makeup flow is provided via a connection to the ESW system through a fire hose stored near the SFP in the reactor building. The UFSAR states that the makeup flow rate required to maintain a SFP level of 36 feet would be 38.8 gpm. A calculation performed in support of a re-rack amendment calculated that for the worst case heat load, the ESW makeup requirement would be approximately 43 gpm (reference calculation CAL-M93-83, rev 3, "Thermal-Hydraulic Analysis of DAEC Spent Fuel Pool with Maximum Density Storage," dated February 2, 1994). The ESW makeup to the fuel pool had a design flow of 75 gpm; however a test on February 20, 1995, revealed that the ESW system was only capable of providing about 10 gpm flow to the pool.

On February 21, 1995, Duane Arnold Energy Center (DAEC) management reviewed this issue and determined that Residual Heat Removal (RHR) Supplemental Fuel Pool Cooling would provide additional cooling capability in the event that all external cooling was lost to the pool and that this was not a safety concern. While a formal operability evaluation for the degraded condition was not performed, the management review discussed the technical aspects of this issue (ie. other sources available,

potential system reconfigurations to increase flow, etc.). However, management realized the importance of this issue and requested a technical evaluation (as part of our corrective action program) of the non-conforming condition.

On February 24, 1995, DAEC entered a scheduled refueling outage. At the time, it was understood that RHR Supplemental Fuel Pool Cooling would be available in the event cooling was lost to the SFP (as stated in UFSAR section 9.1.2.3.2.4). Therefore, the need for a formal operability evaluation or safety evaluation was not recognized at this time.

On March 8, 1995, the core was fully off loaded and the SFP gates were installed. This represented the highest heat load of the outage.

On March 23, 1995, engineering completed the technical evaluation as requested by our corrective action program. The evaluation concluded that while the safety significance of the non-conforming condition was small, it should be corrected in that an ESW design basis function was degraded. Specifically, the evaluation stated that the ESW makeup line to the SFP was designed as the only fully qualified, safety related and seismically supported source of makeup water to the SFP. Although other potential makeup sources exist, including the Condensate Service Water system, the RHR system and the Fire Protection system, each of these sources has particular limitations (e.g., the RHR system can not be used in the SFP assist mode unless the reactor is shutdown), and none is as qualified as ESW for this purpose. The evaluation concluded that a potential licensing concern existed and actions should be taken to either restore the ESW makeup capacity or update the UFSAR.

On March 31, 1995, after reviewing the technical evaluation, management concluded that a potential "Technical Issue" existed in that there was an ongoing unresolved inconsistency between licensing bases documentation and the actual performance of the plant and requested that the issue be evaluated per the 10 CFR 50.59 process.

The safety evaluation was completed on May 2, 1995 and concluded that an adequate level of safety existed even when the ability of the ESW system to provide makeup to the SFP was degraded and that no unreviewed safety question existed. However, the safety evaluation was written assuming the time since shutdown was 60 days. The bases for this assumption was the fact that the plant would be in the non-conforming condition during the upcoming operating cycle. The evaluation was targeted at the future interim condition and did not view the historical condition nor was it intended to support a permanent modification. Additionally, actions were initiated to re-establish the ESW makeup capacity to values higher than those in the re-rack amendment prior to the next refueling outage.

After discussions with your Staff on June 14, 1996, it was determined that the original safety evaluation needed to be updated to include worst case (maximum heat load in the SFP) heat load during the previous refueling outage. The revised safety evaluation concluded that the minimum time to boil that occurred during the outage (12.6 hrs), was more than an adequate amount of time to allow the operating crew to align the RHR system to the SFP, had a loss of the SFP cooling system occurred, and that the RHR system was available during that period with the needed capacity to remove the worst case heat load.

2. CORRECTIVE ACTIONS TAKEN AND THE RESULTS ACHIEVED

On July 30, 1996, the revised safety evaluation, assuming worst case heat load during the previous outage, was issued. The safety evaluation concluded that no Unreviewed Safety Question existed.

As a result of procedure changes and maintenance actions, ESW makeup capability to the SFP was restored to a value above that assumed in the re-rack amendment.

An UFSAR change request has been initiated to update the current design requirements (those calculated as part of the re-rack amendment) for ESW makeup to the SFP.

Management has emphasized to the staff the importance of maintaining compliance with our design bases and UFSAR.

3. CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

This issue will be presented to plant management and the technical staff to re-iterate the need to fully recognize plant conditions that must be considered in a review against 10 CFR 50.59. This will be accomplished as part of continuing Engineering Support Training by March 30, 1997.

4. DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance was achieved on July 30, 1996, with the issuance of the revised safety evaluation.

VIOLATION TWO

Part 50 of 10 CFR, Appendix B, Criterion V, requires in part that activities affecting quality be prescribed by procedures of a type appropriate to the circumstances.

Contrary to the above, the inspectors identified the following deficiencies in plant operating and alarm response procedures that made the procedures not appropriate to the circumstances:

- Operating Instruction (OI) 878.8, "NUMAC Rod Worth Minimizer System," Revision 9, Section 8.2.4, contained no requirement or steps to be performed to correct a rod substitution value once "SUBSTITUTE" key had been pushed.

- In OI 358, "Reactor Protection System," Revision 26, Attachment 4, "RPS Power Supply Transfer Half-Scram Recovery Checklist," contained no requirement or step to reset the carbon bed vault radiation monitor.

- Annunciator response procedure for annunciator 1C03A, C-5, "SRV/SV TAILPIPE HI PRES OR HI TEMP," Revision 4, Section 3.6.a, required the operator to reduce reactor power 25% and cycle the affected Safety/Relief valve's hand switch. The step contained no guidance on how to accomplish the 25% downpower.

- In OI 324, "Standby Diesel Generator System, " Revision 37, Section 5.2, quoted name plate labels differently from the actual name plate labels on the equipment in the plant.

This is a Severity Level IV violation (Supplement 1).

RESPONSE TO VIOLATION TWO

1. REASON FOR THE VIOLATION

A review of the NRC identified procedural weaknesses and deficiencies has identified the following:

Operating Instruction (OI) 878.8, "NUMAC Rod Worth Minimizer System," Revision 9, Section 8.2.4 contained no requirement or steps to be performed to correct a rod substitution value once the "SUBSTITUTE" key had been pushed. The procedure did not contain steps to correct a rod substitution error due to the fact that the procedure assumed that the correct value would be entered. Previous procedure reviews and usage did not identify this potential procedure enhancement.

OI 358, "Reactor Protection System," Revision 26, Attachment 4, "RPS Power Supply Transfer Half-Scram Recovery Checklist," contained no requirement or step to reset the carbon bed vault radiation monitor. Previous procedure reviews and procedure usage did not identify the need to add this step to the procedure.

Annunciator response procedure for annunciator 1C03A, C-5, "SRV/SV TAILPIPE HI PRES OR HI TEMP," Revision 4, assumed that the plant would be at 100% power, hence a power reduction of 25% would provide some margin for the potential pressure changes when the SRV handswitch is cycled. The intent of the procedure is to ensure that power level is at least below 75% prior to cycling the SRV. Previous procedure reviews and procedure usage did not identify this potential enhancement to the procedure.

OI 324 (SBDG) Section 5.2 quoted name plate labels differently from the actual name plate labels on the equipment in the plant. This discrepancy had not been recognized by the plant staff.

2. CORRECTIVE ACTIONS TAKEN AND THE RESULTS ACHIEVED

Operating Instruction (OI) 878.8, "NUMAC Rod Worth Minimizer System," was revised to add a caution to section 8.2 to state that "A rod with a substituted position must be moved prior to substituting another position."

OI 358, "Reactor Protection System," Revision 26, Attachment 4, "RPS Power Supply Transfer Half-Scram Recovery Checklist," has been revised to add a step for resetting the carbon bed vault radiation monitor.

Annunciator response procedure for annunciator 1C03A, C-5, "SRV/SV TAILPIPE HI PRES OR HI TEMP," has been revised to clearly state that power must be below at least 75% prior to cycling a stuck open SRV.

Although the consequences of the identified procedure weaknesses were minor, we fully understand the need for complete, detailed and accurate procedures. As documented in a letter to your staff , NG-94-3137 "Adequacy of the Review and Approval Processes for Changes to the Facility," several efforts have been undertaken over the past few years to improve overall procedure quality. These efforts have resulted in several improvemei, the procedure review and approval process.

3. CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS

The permanently installed nameplate labels on the SBDG will be updated to match the wording in OI 324 (SBDG), Section 5.2 by 12-21-96.

The Quality Assurance department is currently reviewing the identified deficiencies, those previously identified in past inspections, and other internally identified

procedure deficiencies for commonalties concerning our procedure review process. Corrective actions will be initiated as necessary. This review will be completed by November 1, 1996.

4. DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED

Full compliance will be achieved on December 21, 1996, when the above changes are made to the SBDG plant labeling.