U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-331
License No:	DPR-49
Report No.	50-331/96011
Licensee:	IES Utilities Inc. 200 First Street S.E. P. O. Box 351 Cedar Rapids, IA 52406-0351
Facility:	Duane Arnold Energy Center
Dates:	October 25 - December 20, 1996
Inspectors:	K. Riemer, Senior Resident Inspector C. Lipa, Resident Inspector M. Kurth, Reactor Engineer
Approved by:	M. Jordan, Chief Reactor Projects Branch 5

EXECUTIVE SUMMARY

Duane Arnold Energy Center
NRC Inspection Report 50-331/96011

This inspection report included resident and regional inspectors' evaluation of aspects of licensee operations, engineering, maintenance, and plant support.

Operations

- In two cases, operators did not perform surveillance activities according to procedures. See Sections M1.3 and M1.6 for details.

Maintenance

- The "A" RRMG tripped while operators were conducting a test. Poor pre-job planning resulted in the decision to perform this work and post maintenance testing on-line without considering the effects of a failed test on plant operation. (Section M1.2)

- The inspectors were concerned with the recent examples of problems with document change forms (DCFs). Plant operators failed to implement requirements of a DCF into a surveillance. This was a violation. (Section M1.3)

- Average Power Range Monitor (APRM) 15% scram setpoint was rendered inoperable when maintenance was performed that replaced Local Power Range Monitors (LPRMs). The root cause was determined to be an inadequate procedure for LPRM replacement. This was a violation. (Section M1.4)

- The inspectors identified loose lock nuts on a drywell stabilizer. The inspectors concluded that the surveillance procedure was weak. (Section M1.5)

- Operators used incorrect acceptance criteria during a surveillance test. This was an unresolved item. (Section M1.6)

- The inspectors were concerned about material condition, especially since the plant had just completed a refueling outage. (Section M2.1)

Engineering

- The inspectors determined that two events that occurred during the inspection period, were not reported according to 10 CFR Parts 50.72 and 50.73. This was an unresolved item. (Section E8.1)

- The inspectors were concerned that a weakness in pre-job planning resulted in unnecessary additional dose while mechanically cleaning two drywell coolers. This is an unresolved inspection item. (Section E1.1)

- The licensee identified that the residual heat removal inject check valve was chattering. This was an inspection followup item. (Section E1.2)

- One drywell seismic monitor was damaged due to heat. According to the UFSAR, the instrumentation was to be constructed so that it will perform within the range of environmental conditions expected at the plant site. This was an unresolved inspection item. (Section E2)

<u>Plant Support</u>

- No concerns were identified in the plant support area.

Report Details

Summary of Plant Status

At the beginning of the inspection period, the plant was in a shutdown condition for refueling outage fourteen (RFO-14). Startup from RFO-14 was on November 14, 1996, with full power reached on November 24, 1996. On December 2, power was reduced to approximately 85 percent for several hours when the "A" reactor recirculation motor generator set tripped due to a breaker malfunction during post-maintenance testing. On December 13, operators reduced power to 50 percent to remove the "A" feedwater regulating valve from service due to fluctuations in valve position and feedwater flow. The plant was restored to full power on December 14. The plant operated at full power for the remainder of the inspection period.

I. Operations

01 Conduct of Operations

01.1 General Comments (71707)

The inspectors conducted frequent reviews of plant operations. This included observing routine control room activities, accompanying in-plant operators on daily rounds, attending shift turnovers and crew briefings, and performing panel walkdowns. The conduct of operations was professional. Noteworthy observations are detailed in the sections below.

01.2 Reactor Plant Startup From Refueling Outage

a. Inspection Scope

On November 14, 1996, the licensee commenced a reactor plant startup following RFO-14. On November 16, 1996, the main generator was synchronized to the electrical grid. The inspectors observed prestartup evolution briefings. The inspectors also observed in-plant and main control room startup activities.

b. Observations and Findings

The inspectors observed the following during the startup: effective shift management oversight of the activities, formal communications between operators, and strict procedural adherence. The plant had just reached criticality and operators were maintaining power low in the intermediate range when a shift turnover was scheduled to occur. Extra people entered the control room to prepare for turnover activities. The inspectors concluded that the presence of extra personnel was an unnecessary distraction to the operators while at an important point in the startup. The lead control room operator independently determined a distraction existed and directed unnecessary people to leave the control. The turnover was appropriately delayed until the plant was in a stable condition. The inspectors observed that the shift turnover occurred smoothly. The startup continued without problems.

c. Conclusions

The inspectors concluded that the startup was well controlled and conducted in a slow and conservative manner. The inspectors noted excellent coordination existed between control room operators and in-plant operators while bringing systems on line to support plant startup activities.

02 Operational Status of Facilities and Equipment

02.1 Engineered Safety Feature System Walkdowns (71707)

a. Inspection Scope

The inspectors used Inspection Procedure 71707 to walk down accessible portions of the following Engineered Safety Feature systems:

- residual heat removal
- standby gas treatment
- core spray
- standby liquid control
- standby diesel generators
- safety related batteries

c. Conclusions

Equipment operability, material condition, and housekeeping were acceptable in all cases. Minor discrepancies were brought to the licensee's attention and were corrected. The inspectors identified no substantive concerns as a result of these walkdowns.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62703) (61726)

The inspectors observed or reviewed portions of the following testing and work activities:

- Control Rod Scram Time Testing
- High Pressure Coolant Injection Simulated Automatic Actuation Test
- Reactor Core Isolation Cooling Simulated Automatic Actuation Test
- Reactor Core Isolation Cooling Quarterly Operability Test
- Feedwater Regulating Valve Modification Testing
- Standby Diesel Generator Operability Test
- Fuel Movement/Core Alterations

- "A" Standby Diesel Generator 18 Month Preventive Maintenance Activities
- V-20-0082 Chack Valve Repairs
- Loss of Offsite Power and Loss of Coolant Accident (LOOP-LOCA) Testing
- Drywell-Suppression Chamber Vacuum Breaker Test
- Residual Heat Removal (RHR) Motor Operated Valve, MOV-1904 replacement of worn gear
- "D" RHR Pump Breaker Lubrication and Inspection
- "A" Control Rod Drive Pump Seal Repair
- Thermolag removal
- Crosstie Breaker 1B0505 Preventive Maintenance

M1.2 Breaker Failure to Close During Testing Results in Trip of "A" Reactor Recirculation Motor Generator (RRMG)

a. Inspection Scope (62703)

On December 2, 1996, the "A" RRMG tripped while operators were conducting a test. The inspectors followed up on the event and verified that the plant was in a stable condition.

b. Observations and Findings

Operators were performing post-maintenance testing on breaker 1B0505 according to Preventive Maintenance Action Request 1096166 and Operating Instruction 304.1. Breaker 1B0505 is a cross-tie between two non-essential 480 volt buses, 1B5 and 1B6. During the test, breaker 1B0505 did not close as expected. Because breaker 1B0505 did not close, power was temporarily lost to the 1B5 bus. This resulted in a lost of power to two RRMG lube oil pumps and the RRMG tripped on low lube oil pressure.

Corrective actions included: 1) documenting the event on Action Request (AR) 962790, which initiated a root cause analysis, 2) restarting the "A" RRMG, 3) repair of 1B0505, and 4) plans to review of the appropriateness of performing maintenance and testing of cross-tie breakers on-line. The inspectors considered the corrective actions to be appropriate.

c. Conclusions

There was no violation of NRC requirements because the 1B0505 breaker and the RRMG are not safety related. However, the inspectors were concerned with ineffective work planning being a contributor to the event. The inspectors concluded that poor pre-job planning resulted in the decision to perform this work and the associated post maintenance testing on-line without considering the effects that a failed test would have on plant operation.

M1.3 Document Change Form (DCF) Not Incorporated During Surveillance

a. Inspection Scope (62703)

On November 5, 1996, operators identified that instructions on DCF 96-T-0313 were not incorporated into Surveillance Test Procedure (STP) 48A001-SA, "Standby Diesel Generator Semi-Annual Operability Test," Revision 22. As a result, vibration data was not obtained following the two hour run as required. The inspectors independently reviewed this issue and compared it to other DCF problems discussed in two recent NRC inspection reports.

b. Observations and Findings

On November 5, 1996, at the completion of STP 48A001-SA, operators realized that the required vibration data had not been obtained. The instructions on DCF 96-T-0313 were not incorporated into the STP due to personnel error.

The licensee documented the issue on AR 962553. Corrective actions included, 1) review by engineering of the acceptability of the November 5, 1996 test without the vibration data, 2) revising Operations Department Instruction ODI-5 to emphasize that the Operations Shift Supervisor is responsible for ensuring DCFs are incorporated before authorizing the STP, and 3) AR 961838 was issued to require a solutions team to address the recent problem with DCFs not being incorporated into procedures.

c. Conclusions

The inspectors were concerned with the recent examples of problems with DCFs. See NRC inspection reports 50-331/96-06 and 96-07 for details. The inspectors concluded that the licensee's plan to address this problem more broadly was appropriate.

Duane Arnold Technical Specification (TS) 6.8.1 requires that written procedures covering areas such as testing requirements of equipment that could have an effect on the nuclear safety of the facility be implemented. Surveillance Test Procedure (STP) 48A001-SA, "Standby Diesel Generator Semi-Annual Operability Test," Revision 22 was modified by DCF 96-T-0313. DCF 96-T-0313 added a requirement to obtain vibration data following the two hour run at rated load. The plant operators' failure to obtain the required vibration data is considered a violation of TS (50-331/96011-01).

M1.4 Non-conservative Average Power Range Monitor (APRM) 15% Scram Setpoint Due to Inadequate Procedure

a. Inspection Scope (92700)

The inspectors reviewed the details of Licensee Event Report (LER) 96-06. The inspectors independently investigated TS requirements, and reviewed the facts surrounding the event, licensee corrective actions, and root cause.

b. Observations and Findings

As discussed in LER 96-06-00, the licensee identified on October 26, 1996, that the APRM 15% scram setpoint was rendered inoperable when the Local Power Range Monitors (LPRMs) were replaced. The cause of the event was a failure to bypass the 24 LPRMs when they were replaced on October 21, 1996. Since the selector switches were left in "Operate," this caused the APRMs to indicate a neutron power level lower than actual. As a result, the APRM Neutron Flux 15 percent power trip function was inoperable.

Upon identification of this issue, the licensee promptly stopped moving fuel, verified all control rods were full in, and requested further evaluation. The root cause was determined to be an inadequate procedure for LPRM replacement. Refueling Procedure RFP-504, "LPRM Replacement," Revision 2, did not contain instructions to bypass LPRMs. The licensee's corrective actions included 1) placing affected LPRMs in bypass, 2) revision of RFP-504 and other associated procedures, and 3) plans to emphasize this event during training for appropriate personnel.

c. Conclusions

The inspectors concluded that the corrective actions were appropriate. The inspectors also concluded that technical specification requirements were violated.

Technical Specification Table 3.1-1, 2.a requires action to be taken to suspend all operations involving core alterations and to insert all insertable control rods within one hour when the APRM Neutron Flux 15 percent power trip function is inoperable. From October 25, 1996, at 0558 hours until October 26, 1996, at 0625 hours, core alterations were in progress while the APRM Neutron Flux 15 percent power trip function was inoperable. This was considered a violation (50-331/96011-02).

M1.5 Loose Drywell Stabilizer Fasteners Identified After Completion of Surveillance

a. Inspection Scope

On November 1, 1996, during a drywell inspection, the inspectors identified loose lock nuts on a drywell stabilizer. The licensee documented this nonconformance on AR 961273.

b. Observations and Findings

After the inspectors identified two loose nuts on the drywell stabilizer, the licensee initiated a change to surveillance test procedure (STP) 47A001, "Suppression Chamber and Drywell Visual Inspection," to add requirements for inspection of drywell stabilizers. The STP was completed on October 24, 1996. STP 47A001 required only a visual inspection of stabilizers and did not specifically require examining the fasteners. The licensee planned to revise the STP to add inspection requirements for bolts and lock nuts.

The licensee subsequently inspected other fasteners on the stabilizers and repaired additional loose fasteners. The licensee concluded that the effect of the loose fasteners was minor, based on previous experience with evaluating loose fasteners.

c. Conclusions

The inspectors concluded that the surveillance procedure was inadequate. The inspectors concluded that the safety significance was minor and that corrective actions were appropriate. The inadequate procedure was in violation of 10 CFR Part 50, Appendix B, Criterion V. However, because the licensee identified that the procedure was inadequate and appropriate revisions are planned, this is a Non-Cited Violation (NCV) (50-331/96011-03).

M1.6 Incorrect Acceptance Criteria Used for Completion of Daily Surveillances

a. Inspection Scope

On December 8, 1996, operators identified that acceptance criteria used in STP 48A001, "Daily Instrument Checks," had not been correct since October 25, 1996. The inspectors reviewed this issue in parallel with the licensee's investigation.

b. Observations and Findings

The quarterly performance of Emergency Service Water (ESW) STP 48E001-Q determined the maximum allowable river water temperature that would support ESW system operability. The daily STP 42A001 compared actual river water temperature to this maximum allowable value. On October 25, 1996, operators failed to record the data on the "Emergency Service Water Temperature" log sheet. Subsequently, operators transferred the incorrect data to the daily STP 42A001. As a result, the acceptance criteria used in the STP 42A001 (91.8°F for "A" and 93.0°F for "B") were not the actual values (90.5°F for "A" and 94.3°F for "B"). During the winter months, actual river temperature is typically less than 40°F. Therefore, actual river temperature was significantly below the limit and there was low safety significance associated with the use of the incorrect acceptance criteria.

c. Conclusions

The inspectors were concerned with the practice of using the "ESW Temperature" log sheet and transferring data from one STP to the next, rather than referencing the source document. The inspectors planned to review other similar checks made during the performance of STP 42A001 to determine whether this was an isolated case. Pending further NRC review, this is an unresolved item (50-331/96011-04).

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Plant Material Condition

a. Inspection Scope

The inspectors noted that there were several material condition issues and self-revealing equipment failures during the report period. The inspectors reviewed the failures to determine if there was any affect on plant safety. In each case, the inspectors observed appropriate licensee efforts to determine root cause and schedule repair. The examples are listed below:

- On November 1, 1996, the inspectors identified loose lock nuts on a drywell stabilizer as discussed in Section M1.5.

- On November 2, 1996, the low pressure coolant injection swing bus transfer breaker closed in five minutes instead of five seconds as expected. See Section E8.1 for details.

- On November 7, 1996, a drywell seismic monitor was found damaged by excessive heat. See Section E.2 for details.

- On November 24, 1996, operators identified a banging noise in the RHR valve room. The cause was later determined to be chattering of the RHR check valve V20-82. See Section E1.2 for details.

- From November 24 until December 2, 1996, there were intermittent alarms of high flow on the "B" reactor recirculation pump seal flow switch. Based on alternate indication, the alarms were determined to be invalid. This is documented on AR 962385. The inspectors independently verified alternate control room indication, such as identified leakage rate, seal pressure, and drywell temperatures and found no concerns.

- Beginning November 25, 1996, control room chart recorders indicated higher than expected drywell particulates. This is documented on AR 962631.

- On November 26, 1996, the "B" river water supply pump breaker would not close in from the control room. The breaker was subsequently replaced and the original will be sent to the manufacturer to determine the root cause per AR 962779.

- On December 2, 1996, the "A" RRMG set tripped when cross-tie breaker 1B0505 failed to close during post-maintenance testing. See Section M1.2 for details.

- On December 3, 1996, the licensee identified that 5 out of 8 main steam relief and safety valves failed the as-found setpoint test. See Section E8.1 for details.

- On December 13, 1996, operators reduced power to 50 percent due to fluctuations on the "A" feedwater regulating valve. The cause was determined to be a cracked solder joint in the positioner assembly, which was subsequently repaired.

- Throughout the inspection period, unidentified drywell leakage continued to increase slightly. Although the value of 0.40 gallons per minute (gpm) on December 20, 1996, was well below TS limit of 5 gpm, the licensee was monitoring to attempt to determine the source of the leakage.

c. <u>Conclusions</u>

The inspectors were concerned about the equipment problems discussed above, especially since the plant had just completed a refueling outage. For example, drywell unidentified leakage was at 0.40 gpm, which was greater than the value of 0.18 gpm at the end of the previous cycle. In each case, the inspectors concluded that the licensee's response to the issue was appropriate.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 <u>(Closed) Inspection Follow-up Item 50-331/96003-08:</u> Discrepancy between UFSAR and licensee's test method for secondary containment. The licensee previously closed both sets of secondary containment isolation dampers before testing to ensure capability to maintain 1/4 inch of water vacuum. Test procedure STP 47J001-CY was revised before refuel outage fourteen (October 1996) to test with only one set of dampers closed at a time. The inspectors verified the results of STP 47J001-CY performed on October 16 and October 18, 1996. The tests were satisfactory. This item is closed.

M8.2 <u>(Closed) Licensee Event Report (LER) 50-331/96-06-00:</u> Non-conservative APRM 15% scram setpoint due to inadequate procedure. The inspectors reviewed this LER and determined that a violation occurred as discussed in Section M1.4. The licensee's corrective actions will be reviewed as part of the NOV closure. This item is closed.

III Engineering

E1 Conduct of Engineering

a. Inspection Scope (37551)

The inspectors evaluated engineering involvement in resolution of emergent material condition problems and other routine activities. The inspectors reviewed areas such as operability evaluations, root cause analyses, safety committees, and self assessments. The effectiveness of the licensee's controls for the identification, resolution, and prevention of problems was also examined.

E1.1 Weak Pre-Job Planning for Cleaning of Drywell Coolers

a. Inspection Scope

On October 25, 1996, the licensee identified that two drywell cooler isolation valves that were deenergized open as part of a temporary modification, (TM) 95-148 were found closed. AR 962053 was written to document the issue. The inspectors reviewed the licensee's resolution of this issue and conducted independent interviews.

b. Observations and Findings

The licensee had installed TM 95-148 in April 1995 after identification of low resistance short circuits in drywell penetration JX105C. The purpose of the TM was to deenergize open drywell cooling valves to prevent undesired operation or continued insulation breakdown within the penetration.

As a part of resolution of AR 962053, the licensee concluded that the drywell cooling valves may have been closed during the installation of the TM due to cross connection of circuits to the valve. This allowed the valves to be powered closed after the fuses had been pulled. The licensee subsequently relocated the control circuits to spare conductors within the penetration. These spare conductors had good insulation.

The inspectors reviewed another aspect of the incorrect position of the drywell cooler isolation valves. During RFO-14, chemical cleaning of drywell coolers was planned to resolve a long term material condition problem with drywell coolers (IR 50-331/96-04). The engineer planning the job did not establish a valve line-up before the flush. Instead, the engineer assumed the valves would be open since TM 95-148, which failed valves open, was still in effect. The isolation valves for coolers 3A and 5A were actually closed during the chemical cleaning. Subsequent mechanical cleaning of the coolers was required and workers received an additional 1.6 Rem of dose as a result of this error.

c. Conclusions

The inspectors concluded that the problem with inadequate cable insulation within penetration JX105C was resolved. This was accomplished when the control circuits for drywell cooling valves were re-routed to spare cables with good insulations. The inspectors had no concerns with the licensee's additional efforts to resolve problems with the penetration. The inspectors were concerned that a weakness in pre-job planning resulted in unnecessary additional dose to mechanically clean the two drywell coolers. The review of this event was not completed prior to the end of the inspection. The inspectors will review the licensee's processes for configuration control of TMs. In addition, the inspectors will review the licensee's operability evaluation. This is an unresolved inspection item (50-331/96011-05) pending the inspectors' review.

E1.2 RHR Check Valve Chattering

a. Inspection Scope

On November 24, 1996, the licensee identified a banging sound in the RHR valve room. Through extensive troubleshooting efforts, engineering determined that the "A" RHR loop inject check valve was chattering. The inspectors reviewed the licensee's initial operability evaluation, the details of the work done on the check valve at the end of the outage, and walked down the acceptable portion of the system.

b. Observations and Findings

Through discussions with the valve manufacturer, the licensee concluded that the chattering did not affect RHR system operability. The licensee commenced a detailed operability evaluation in January 1997. The licensee also determined through discussion with the valve manufacturer that repairs performed on the check valve during RFO-14 were appropriate.

c. Conclusions

The inspectors determined that further review was required by NRC valve specialists. This is an inspection follow-up item (50-331/96011-06) pending review of the licensee's operability determination.

E1.3 Conclusions on Conduct of Engineering

The inspectors considered engineering efforts to investigate or resolve material condition issues discussed in Section M2.1 to be appropriate.

E2 Engineering Support of Facilities and Equipment

a. Inspection Scope

The inspectors reviewed plant equipment and activities against the UFSAR descriptions. Two discrepancies were identified.

b. Observations and Findings

- As discussed in Section E8.1, below, the inspectors identified a difference between an analysis described in the UFSAR and the Reload Analysis for relief valve and safety valve allowable set-points.

- On November 7, 1996, the licensee identified that drywell seismic monitor QR0008 was damaged due to excessive heat. The design temperature of the monitor was 300 degrees Fahrenheit. Although ambient temperatures in the drywell were well below 300^{0}F, the licensee indicated that this monitor, installed on the reactor pressure vessel, may be exposed to temperatures greater than 300^{0}F. The monitor was subsequently replaced and high temperature film was installed. The licensee installed temperature indicators to monitor the equipment during the cycle and initiated preventive maintenance to be done each refuel outage.

  The inspectors reviewed the licensee's response to Regulatory Guide 1.12, Instrumentation for Earthquakes, contained in UFSAR Section 1.8.12. Regulatory Position 7 states that the instrumentation should be designed to perform its function over the appropriate range of environmental conditions, such as temperature. In response to this Regulatory Position, the licensee stated that the instrumentation was constructed in such a way that it will perform in a satisfactory manner within the range of environmental conditions expected at the plant site.

c. Conclusions

The inspectors were concerned that seismic monitor QR0008 was not designed to perform its function over actual temperature ranges as discussed in the UFSAR. This discrepancy will be reviewed further as an unresolved inspection item (50-331/96011-07).

E8 Miscellaneous Engineering Issues (37551, 92902)

E8.1 Reportability of Events under 10 CFR 50.72 and 50.73

a. Inspection Scope

The inspectors identified two events that appeared to be not reported according to 10 CFR Parts 50.72 and 50.73. In both cases, the licensee determined that the events were not reportable. The inspectors reviewed the licensee's operability evaluations, independently verified the facts of the events, and held discussions with the licensee on their justification for the events not being reportable.

b. Observations and Findings

- During surveillance testing on November 2, 1996, low pressure coolant injection swing bus transfer breaker 1B4401 failed to close within five seconds as expected. Instead, the breaker took approximately five minutes to close. The cause was determined to be an oil leak that affected the undervoltage trip time delay and this was subsequently repaired and tested satisfactorily.

  The licensee originally reported the event under 10 CFR Part 50.72 (b)(2)(i) based upon the scenario of a LOOP-LOCA with worst single failure of one division of 125 VDC. In this scenario, the failure of the bus to transfer for 5 minutes would have resulted in only one train of core spray being available. According to licensee analysis, a single core spray pump, at rated flow, is not sufficient to maintain the peak cladding temperature below the regulatory limit during the initial 5 minute period post-LOCA.

  The notification was subsequently retracted based on licensee's review of reportability requirements. Since only a single component, 1B4401, was inoperable, with no other indications of other component problems, the licensee considered that this event was not reportable.

- On December 3, 1996, the licensee received as-found testing results which revealed that 4 out of 6 main steam relief valves (MSRVs) and 1 out of 2 main steam safety valves (MSSVs) failed to meet the requirements of Technical Specifications (TS) 2.2.1.B and 2.2.1.D, respectively. This was a repetitive problem at Duane Arnold in that setpoint testing results since 1980 indicated an as-found failure rate of 48 % for MSRVs and 65 % for MSSVs. The TS specified an allowable range of plus or minus 11 psi for MSRVs and plus or minus 12 psi for MSSVs (plus or minus approximately 1%) and Section 5.4.13.3 of the UFSAR stated that the transient analysis assumed that the valves operated at the setpoints plus 1%.

  The licensee based their decision that the results were not reportable to the NRC on the Reload Analysis, which assumed each of the valves was set 3% above the nominal settings. All eight pilot valves were subsequently replaced with valves that were set within TS requirements. The licensee planned to submit a "voluntary" LER in the near future.

c. Conclusions

Part 50.73(a)(2)(ii) of 10 CFR requires licensees to report events or conditions that resulted in the condition of the nuclear power plant being in a condition that was outside the design basis of the plant. The inspectors were not able to resolve this issue during the inspection period and will review further to determine whether 10

CFR 50 reportability requirements were violated. Pending further NRC review, this item is considered an unresolved item (50-331/96011-08).

IV Plant Support

R1 Radiological Protection and Chemistry Controls

a. Inspection Scope (71750)

The inspectors observed radiological postings and evaluated radiological work practices while observing maintenance and test activities. One concern was noted with poor communications during a High Pressure Coolant Injection draining evolution, which is discussed in detail in NRC inspection report 96-09. The inspectors had no other concerns in the plant support area.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on December 20, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Franz, Vice President Nuclear
G. Van Middlesworth, Plant Manager
R. Anderson, Manager, Outage and Support
P. Bessette, Manager, Engineering
J. Bjorseth, Maintenance Superintendent
D. Curtland, Operations Manager
R. Hite, Manager, Radiation Protection
K. Peveler, Manager, Regulatory Performance

INSPECTION PROCEDURES USED

IP 37551:	Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observation
IP 62703:	Maintenance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901:	Followup - Operations
IP 92902:	Followup - Engineering
IP 92903:	Followup - Maintenance
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-331/96011-01	NOV	DCF Not Incorporated During Diesel STP
50-331/96011-02	NOV	APRM 15% Trip Function Inoperable During Core Alterations
50-331/96011-03	NCV	Inadequate surveillance procedure
50-331/96011-04	URI	Incorrect Acceptance Criteria in Daily STP
50-331/96011-05	URI	Configuration control of temporary modifications
50-331/96011-06	IFI	RHR Check Valve Chattering
50-331/96011-07	URI	Drywell Seismic Monitor Design Temperature Exceeded
50-331/96011-08	URI	Reportability of Events Under 10 CFR 50.72 and 50.73

Closed

50-331/96003-08	IFI	Discrepancy Between UFSAR and Secondary Containment Test Method
50-331/96006-00	LER	Non-conservative APRM 15% Scram Setpoint Due to Inadequate procedure

LIST OF ACRONYMS USED

APRM	Average power range monitor
AR	Action Request
CFR	Code of Federal Regulations
DAEC	Duane Arnold Energy Center
DCF	Document Change Form
ESW	Emergency service water
IFI	Inspection followup item
IP	Inspection procedure
IR	Inspection report
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPRM	Local power range monitors
NOV	Notice of Violation
NRR	Office of Nuclear Reactor Regulation
RFP	Refueling Procedure
RHR	Residual heat removal
RRMG	Reactor recirculation motor generator
STP	Surveillance Test Procedure
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

NRC Form 8-C
(4-79)
NRCM 0240

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