U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-331/91015(DRP)

Docket No. 50-331 License No. DPR-49

Licensee: Iowa Electric Light and Power Company
IE Towers, P. O. Box 351
Cedar Rapids, IA 52406

Facility Name: Duane Arnold Energy Center

Inspection At: Palo, Iowa

Inspection Conducted: August 20 through September 30, 1991

Inspectors: M. Parker
C. Miller
J. McCormick-Barger

Approved: R. L. Hague, Chief
Reactor Projects
Section 3C

10/16/91
Date

Inspection Summary

Inspection on August 20 through September 30, 1991 (Report No. 50-331/91015(DRP))

Areas Inspected: Routine, unannounced inspection by the resident inspectors and a region based inspector of followup; licensee event reports followup; operational safety; maintenance; surveillance; temporary instruction; and report review.

Results: An executive summary follows:

Operations

Operating performance has been good, with no major events occurring this period. The plant operated at or near 100% power throughout the period except for routine down power maneuvers for surveillances, rod maneuvers, and load following. Upkeep of the simulator to match control room modifications is noted as a strength (Section 4.a).

Maintenance/Surveillance

Continued pipe wall thinning and failure on portions of the "B" reactor feed pump seal water return line have been observed this period. Extensive wall thinning was noted. Three elbows, including one that failed, were temporarily patched (Section 5). Continued efforts to monitor pipe wall thickness and repair the pipe, as necessary, appear warranted.

Problems with implementing management expectations with the procedure change process continue to surface. A non-cited violation for the problems was issued and is discussed in Section 6.b.

Special Test Procedure #170 (Feedwater Tracer Test) was a complex procedure which was carried out very successfully this period. Extensive technical and ALARA planning was required for this success and was evident in the entire process (Section 6.a).

Engineering and Tech Support

Engineering response to a priority one engineering work request (EWR-910031) appeared minimal and probably contributed to the failures in the reactor feed pump seal water return lines (Section 5).

Safety Assessment/Quality Verification

The licensee made a commendable proactive effort to notify the NRC of a possible need for a temporary waiver of compliance (TWOC) regarding seismic qualification of Foxboro instrument racks. The TWOC was not necessary since the equipment was not declared inoperable, and because the licensee took prompt aggressive action to restore the equipment to its qualified configuration. DAEC management kept the NRC well informed of the progress of the corrective actions so that had a TWOC been necessary, the request could have been processed efficiently and in a timely manner (Section 4.b).

DETAILS

1. Persons Contacted

*R. Anderson, Assistant Operations Supervisor
R. Anderson, Senior Outage Project Manager
*P. Bessette, Licensing Engineer
J. Bjorseth, Assistant Operations Supervisor
*D. Blair, Group Leader, Internal Audits, Quality Assurance
*C. Bleau, Systems Engineering Supervisor
C. Bock, Group Leader Systems Engineering
A. Browning, Supervising Engineer, Licensing
V. Crew, Technical Support Engineer
J. Edom, Reactor and Computer Performance Supervisor
H. Giorgio, Radiological Engineering Supervisor
P. Hansen, System Engineer
H. Johnson, Warehouse Supervisor
B. Lacy, Manager, Design Engineering
*M. McDermott, Maintenance Superintendent
R. McGee, Technical Support Specialist
*C. Mick, Operations Supervisor
W. Miller, Supervising Engineer, Analysis Engineering
*K. Peveler, Corporate Quality Assurance Manager
*J. Probst, Technical Support Engineer
*K. Putnam, Technical Support Supervisor
*D. Robinson, Technical Support Engineer
B. Schenkelberg, Fire Protection
*N. Sikka, Discipline Component Engineering Supervisor
*J. Thorsteinson, Assistant Plant Superintendent, Operations Support
*G. Van Middlesworth, Assistant Plant Superintendent, Operations
*D. Wilson, Plant Superintendent, Nuclear
K. Young, Assistant Plant Superintendent

U. S. Nuclear Regulatory Commission (NRC)

*C. Miller, Resident Inspector
*M. Parker, Senior Resident Inspector
J. McCormick-Barger, Project Inspector

In addition, the inspectors interviewed other licensee personnel including operations shift supervisors, control room operators, engineering personnel, and contractor personnel (representing the licensee).

*Denotes those present at the exit interview on October 4, 1991.

2. Followup (92701)

(Closed) Unresolved Item 331/91011-01(DRP): During review of the licensee's 10 CFR Part 21 reporting program, the inspectors identified concerns with the licensee's engineering report (record 89-11) that evaluated the failure of a T-ring seal associated with leaking containment isolation valve, CV4302. The engineering evaluation appeared to have been inadequate, in that it did not pursue an apparent change in the way the T-rings were being manufactured.

Subsequent to the initial inspection, the licensee sent a letter to the T-ring vendor requesting specific information concerning the fabrication of the T-rings, including any changes in the method of fabrication and any known failures of glued joints. The inspectors reviewed the vendor's reply to the licensee's questions and inspected T-rings located in the warehouse. From the above review, the inspectors determined that T-rings were originally fabricated using 50 Durometer EPT material. It appears, from inspection of these types of T-rings, that they were fabricated from a continuous mold. Later, the vendor changed suppliers and, via a design change, changed the material to EPDM 70 Durometer. The new T-rings were fabricated from a larger molded T-ring, which was cut, trimmed to size, and bonded at a spliced joint.

The T-ring vendor stated that the only other reported failures of T-ring glued joints were traced to excessively stressing the T-rings during receipt inspection. The vendor stated that, with the exception of the licensee's recent case, no reported failures of T-ring glued joints have occurred in-service. A review of the Nuclear Plant Reliability Data System (NPRDS) also did not identify any failures of T-ring glued joints. The vendor also stated that they believed that a T-ring glued joint failure should not result in more than bubble leakage and that the licensee's leakage would most likely have been attributed to the disc not being in its fully seated position rather than failure of the T-ring joint.

The licensee stated that they do not believe the T-ring failure constituted a substantial safety hazard and is therefore not reportable. However, the licensee stated that they are attempting to obtain molded T-rings from the vendor and plan to dispose of the glued joint T-rings currently in their warehouse. The inspectors have no further questions concerning this item. This item is closed.

No violations or deviations were identified in this area.

3. Licensee Event Reports Followup (92700) (90712)

Through direct observations, discussions with licensee personnel, and review of records, the following event reports were reviewed to determine that reportability requirements were fulfilled, immediate corrective action was accomplished, and corrective action to prevent recurrence had been accomplished in accordance with technical specifications.

a. <u>(Closed) Licensee Event Report (LER) 89-012 (331/89012-LL):</u> Loss of Secondary Containment Due to Degraded Vent Shaft and Inadequate Test Methods. This LER documented the failure of the reactor building (RB) ventilation duct work inside the RB exhaust fan room, and the inability of the secondary containment operability test to detect a failure of this nature. The LER was originally discussed in paragraph 4.c of inspection report 50-331/90009(DRP). A violation was issued concerning the inadequate surveillance test in inspection report 50-331/89026(DRP) and was closed in section 2.c of inspection report 50-331/91003(DRP). This LER remained open pending completion of the licensee's review of all Seismic Class I duct work against the original design documentation and the implementation of a trending program of secondary containment testing. Both of these activities were identified as corrective actions in the licensee's LER.

In a letter to the NRC dated November 7, 1990, the licensee revised its original completion date for performing the review of the duct work to the original design documents from December 31, 1990, to June 30, 1991. The inspectors were informed that the effort required to perform the duct work review was substantially more extensive than the licensee originally had thought. Due to an inability to obtain substantial original design documentation, the licensee has had to undergo a significant design reconstitution effort for the duct work. To date, no inoperable duct work has been identified; however, minor deficiencies were identified and are being addressed.

The licensee completed about half of the duct work review and completed an engineering judgement operability review of the remaining duct work on June 30, 1991. A more substantial design review is ongoing and is to be completed by the end of 1992. The change of scope concerning the review of duct work was documented in the licensee's Semi-Annual Report for the "Plan for the Integrated Scheduling of Plant Modifications for the Duane Arnold Energy Center" dated May 3, 1991. Completion of the design documentation review and all modifications is being tracked by the licensee's Business Plan.

A formal performance trending program of secondary containment ventilation testing results, while on line to allow early detection of system degradation, is in place. For example, the licensee was well aware that early 1991 testing results met technical specifications but had indicated that secondary containment leak tightness had significantly degraded. In response to this degradation, the licensee increased its efforts to perform maintenance on airlock doors and apply caulking to exterior wall seals. Because of these actions, the licensee obtained one of the best results for secondary containment integrity in recent history. This LER is closed.

b. (Open) Licensee Event Report (LER) 90-005 (331/90005-LL): Average Power Range Monitor Flow-Biased Trip Setpoint Shifted to the Non-Conservative Direction Due to Inadequate Flow Units Calibration. On May 25, 1990, a system engineer identified that a non-conservative deviation existed on the flow biased trip setpoint. The deviation between the actual flow biased trip setpoint and the required setpoint was 4.1%. The intermediate cause was identified to be that the flow units were calibrated to the recirculation drive flow required to achieve 100% core flow at full power during the initial plant operation, 26,550 gpm. The recirculation drive flow required to achieve 100% core flow at full power had increased since the initial startup, but the calibration procedure was not updated accordingly. The root cause was identified to be failure to recognize the potential for changes in recirculation driving flow and have an established program to monitor and feed back the changes into the calibration procedure.

The average power range monitor (APRM) flow units provide the thermal scram setpoint for safe plant operation. The setpoint is automatically adjusted in proportion to the recirculation drive flow in percent of design. At 100% indicated recirculation loop flow, the flow control trip reference unit electronically prevents the trip signal from exceeding the required 120% scram trip setting. The deviation described above caused the APRM flow bias trip setpoint to be non-conservative at recirculation loop flows less than 28,800 gpm (the current 100% recirculation flow). At recirculation loop flows greater than 26,550 gpm, the APRMs were "clamped" to the 120% setpoint due to the reference unit overriding the flow signal. The deviation was determined to have no effect on safe plant operation since the flow bias trip setpoint provides additional safety margin beyond the 120% scram trip protection assumed in the Accident Analyses.

The details of this event were initially reviewed and results of the review were documented in paragraph 3.d of inspection report 50-331/90009(DRP). The inspectors reviewed the following immediate and long-term corrective actions:

- Adjustment of the gain adjustment factor (GAF) for the APRM setting upward to compensate for the deviation in the flow units (immediate action).

- Revise the flow unit calibration procedure to reflect the current 100% recirculation flow of 28,800 gpm (immediate action).

- Recalibrate the flow units using the revised procedure and re-adjust the GAF on the APRMS (immediate action).

- Develop a procedure to periodically (following every refueling outage) determine the new 100% recirculation flow value and incorporate the value into the flow unit's calibration procedure. Recalibrate flow units following incorporation of new value (long-term action).

- Investigate the variables contributing to the increase in the recirculation driving flow.

The licensee completed all corrective actions except the development of the procedure to determine the 100% recirculation flow value following refueling outages. This action was ongoing and is to be completed prior to the upcoming 1992 refueling outage. The licensee determined that 100% recirculation flow value changes are due to jet pump aging, internal core resistance changes (different fuel), and changes during a fuel cycle as rod density changes and as the core void fraction changes.

This LER will remain open pending the inspectors review of the procedure that will be used to determine the new 100% recirculation flow value following refueling outages.

No violations or deviations were identified in this area.

4. Operational Safety Verification (71707) (71710) (42700)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators during the inspection. The inspectors verified the operability of selected emergency systems, reviewed tagout records, and verified proper return to service of affected components. Tours of the reactor building and turbine building were conducted to observe plant equipment conditions, including potential fire hazards, fluid leaks, and excessive vibrations and to verify that maintenance requests had been initiated for equipment in need of maintenance. It was observed that the Plant Superintendent, Assistant Plant Superintendent of Operations, and the Operations Supervisor were well informed of the overall status of the plant and that they made frequent visits to the control room and regularly toured the plant. The inspectors, by observation and direct interview, verified that the physical security plan was being implemented in accordance with the station security plan.

The inspectors observed plant housekeeping/cleanliness conditions and verified implementation of radiation protection controls. During the inspection, the inspectors performed a control room walkdown of all Emergency Core Cooling Systems (ECCS) to verify operability by comparing system lineup with plant drawings and present valve lineup lists.

These reviews and observations were conducted to verify that facility operations were in conformance with the requirements established under technical specifications, 10 CFR, and administrative procedures.

a. Simulator Observations

On August 30, 1991, the inspectors performed a review of the licensee's control room simulator. In addition to reviewing the simulator capabilities and functions, the inspectors reviewed with operations supervision the use of plant procedures and expected operator responses in several plant surveillances, transients, and accident scenarios. The simulator, with one exception, performed very well. Controls were nearly identical to the plant's actual controls, and the licensee has expended significant efforts to make the simulator look and respond the same as the plant's control room. The licensee has done a good job keeping up with plant modifications and incorporating the majority of the Detailed Control Room Design Review (DCRDR) modifications installed during the 1990 refueling outage. The exception was the response of the feedwater controller during a reactor scram. The simulator feedwater controller was far more sensitive to the plant transients than the actual plant feedwater controller. The licensee stated that this discrepancy was new and probably the result of a flaw in a very recent reload of the computer. A discrepancy report was written to document and address the problem.

The inspectors also noted that although extensive licensee effort was involved in keeping the simulator conditions closely resembling the actual control room, some alarms which appear frequently in the control room do not appear on the simulator. Specifically, the licensee does not always make a conscious effort to have the simulator reflect actual plant conditions/equipment out of service conditions.

b. Foxboro Instrument Racks

On September 17, 1991, the licensee questioned the seismic qualification of Foxboro instrument racks located in the control room. During a walkdown of Foxboro instrument racks, the licensee identified that dummy circuit cards were not installed and that circuit card bumper pads and mounting rails to the instrument racks were missing. The missing components could render the instrument racks and, subsequently, technical specifications (TS) related instrumentation/equipment inoperable during a seismic event. The affected instrumentation/equipment was accident monitoring instrumentation (wide range drywell pressure monitor, narrow range drywell pressure monitor, torus water level monitor, and containment water level monitor) and instrumentation that initiates or controls the core and containment cooling systems (reactor low level, inside shroud). This instrumentation is required by TS 3.2, Tables 3.2.H and 3.2.B, respectively.

While the equipment was functionally operable, operability was questioned when the licensee received vendor information which stated that seismic testing was performed on this instrumentation with all circuit cards installed and all circuit card rails and pads installed. The licensee took immediate action to determine operability. In parallel with the operability determination, the licensee took action to restore the Foxboro instrument racks to the configuration provided during seismic testing qualification. The restoration of equipment was prioritized, based on safety significance of the affected instrumentation. With the exception of the low reactor water level permissive function, all other equipment provided control room indication only. Therefore, the licensee took action to restore the permissive function of the low reactor water level first. While this instrument did not initiate a core or containment cooling system, it did provide a permissive function by preventing inadvertent operation of containment spray during accident conditions.

The licensee completed repair/restoration of the affected Foxboro instrument racks on September 19, 1991, prior to completing an operability determination. Currently, all Foxboro instrument racks are configured similarly to their arrangement during seismic qualification testing.

At present, the licensee still intends to perform an operability determination and pursue any regulatory reporting per 10 CFR 50.72. During the time the instrumentation/equipment was in question, the licensee kept the NRC informed of their proposed actions. This close communication was maintained to provide advance notification to the NRC if the licensee had determined that the equipment was inoperable, and the licensee had requested a temporary waver of compliance (TWOC) to allow additional time for repairs beyond the TS limiting condition for operation (LCO). While the TWOC was not needed due to the licensee completing repairs in a timely manner, the up front communications provided the NRC with additional time to implement necessary action had the TWOC been necessary.

No violations or deviations were identified in this area.

5. Monthly Maintenance Observation (62703)

Station maintenance activities of safety related systems and components listed below were observed/reviewed to ascertain that they were conducted in accordance with approved procedures, regulatory guides, and industry codes or standards, and in conformance with technical specifications.

The following items were considered during this review: the limiting conditions for operation were met while components or systems were removed from service; approvals were obtained prior to initiating the work; activities were accomplished using approved procedures and were inspected as applicable; functional testing and/or calibrations were performed prior to returning components or systems to service; quality control records were maintained; activities were accomplished by qualified personnel; parts and materials used were properly certified; radiological controls were implemented; and, fire prevention controls were implemented.

Work requests were reviewed to determine status of outstanding jobs and to assure that priority was assigned to safety related equipment maintenance which may affect system performance.

The following maintenance activities were observed/reviewed:

EPA "A2" breaker repair

MO2100 ("A" Core Spray Pump Outboard Torus Suction Valve) repairs

"B" Reactor Feed Pump Seal Water Return Line repairs

RPS MG Set Ammeter replacement

Seal Water Line Leak

The "B" reactor feedwater pump (RFP) 1-1/2 inch seal water return line (GBD-067) developed a 3/4 inch hole around September 25, 1991. The piping carries a two phase water/steam mixture, at about 325°F, from the "B" RFP. On September 27, 1991, the licensee used a temporary leak repair patch to repair the leak, and also patched two other locations on the GBD-067 line which appeared susceptible to erosion/corrosion wall thinning.

Another portion of this piping had been repaired in January, 1991, due to wall thinning discovered when GBD-066 ("A" RFP seal water return line) failed, requiring a plant shutdown to repair. An engineering work request (EWR-910031) has been in effect since January, 1991, to evaluate methods to prevent or minimize erosion in this piping. This EWR had a Priority 1 designation, which Design Engineering Department procedures characterize as "an immediate risk to the continued safe operation of the plant and/or personnel". The inspectors discussed the progress of EWR-910031 with system engineering supervision and the system engineer responsible for this piping. Very little engineering work had been performed prior to September 25, 1991, to come to a resolution of this problem. The system engineer is now working on a plan to resolve the problem before the pipe walls again break through.

The inspectors are concerned, since significant wall thinning had been previously detected in the piping in several areas, that another breakthrough could occur in the short term if not adequately monitored. The inspectors will continue to monitor the licensee's efforts in resolving EWR-910031, as well as their efforts to monitor and correct wall thinning problems associated with GBD-066 and 067.

Following completion of maintenance on the core spray system and the reactor protection system (RPS) electric protective assembly (EPA) Breakers, the inspectors verified that these systems had been returned to service properly.

No violations or deviations were identified in this area.

6. Monthly Surveillance Observation (61726)

The inspectors observed technical specifications required surveillance testing and verified that testing was performed in accordance with adequate procedures, that test instrumentation was calibrated, that limiting conditions for operation were met, that removal and restoration of the affected components were accomplished, that test results conformed with technical specifications and procedure requirements and were reviewed by personnel other than the individual directing the test, and that any deficiencies identified during the testing were properly reviewed and resolved by appropriate management personnel.

The inspectors also witnessed portions of the following test activities:

STP-41A015 - Local Power Range Monitor Calibration

STP-41A018-M - APRM Flow Bias Instrument Functional Test

STP-41A127 - RPS MG Set and Alternate Power Source EPA Functional Test and Calibration

STP-45A001-QA - Core Spray System Quarterly and Annual Operability Test

STP-47D004 - Main Steam Isolation Valve Trip/Closure Time Check

STP-48001-Q - Standby Diesel Generator Monthly Operability Test

NS-13C004-A - Deluge System Annual Test and Inspection

NS-93001 - Weekly Main Turbine Operation Tests

Special Test Procedure #170 - Feedwater Tracer Test

a. Feedwater Tracer Test

On September 20, 1991, the licensee performed Special Test Procedure #170, Feedwater Tracer Test. The purpose of the test was to perform an on-line calibration check of feedwater flow elements FE1581 and FE1626 using a radioactive sodium (Na-24) tracer. Differential pressure (DP) signals from FE1581 and FE1626 are the primary input to the reactor heat balance at DAEC. Maintaining an accurate calibration of these signals is a critical factor from both a safety and production viewpoint.

The inspectors reviewed the safety evaluation for performance of the test and determined that it adequately addressed possible adverse affects on reactor coolant chemistry and ALARA concerns during the test. The tracer material, one gram of sodium nitrate, was not expected to increase conductivity more than 0.02 micro mhos per centimeter, nor increase main steam line or offgas radiation levels significantly.

The inspectors attended several pre-evolution meetings to ensure that nuclear safety and ALARA concerns were being adequately addressed. The vial containing the tracer contained approximately .63 curies of sodium, and was expected to read about 2000 R/hr on contact. The licensee was involved in extensive job planning and coordination efforts to ensure that all members involved in the evolution were familiar with their tasks, expected dosage rates, limits, and actions in the event of problems. Participants staged a comprehensive walk through using a mock vial to ensure the procedure worked as written, and that personnel were familiar with how to perform their tasks efficiently. The special test procedure required a senior line manager to be present, in addition to the test conductor, to ensure the test was conducted in a safe, conservative manner. The radiation work permit (RWP-9100288) written to cover the special test procedure included detailed source, cask, dilution, and transfer tubing dose rates, special instructions for material and personnel decontamination based on lessons learned from previous industry experience, and spill control actions.

Actual conduct of the test proceeded very smoothly. After minor delays in processing the radiation work procedure, the test proceeded almost on schedule. Overall dose was approximately one rem, which was slightly under the budgeted dose for cask handling, source handling, and heater bay work. The dose was mainly accrued during setup and take down operations in the heater bay. The actual dose from handling the source during the test was about 60 mr.

The test method involved injecting a known activity concentration of Na-24 at an accurately measured flow rate into the reactor feed system, then sampling and counting the activity concentration of Na-24 downstream of the feed flow nozzles. The rate of feedwater flow is then determined by calculating the ratio between the activity concentration times injection flow rate versus activity concentration times the feedwater flow rate. The licensee also expects to be able to get an accurate determination of moisture carryover from the test as a result of sampling condensate.

Results of the test are still being evaluated by the licensee and the tracer test contractor, NWT. Initial test results showed the feedwater flow indicated by the differential pressure transmitters to be close to that indicated by the tracer test. The inspectors will continue to review the licensee's test methods and results to determine the acceptability of using the results to calibrate the feedwater flow instrumentation.

b. RPS EPA Breaker Calibration

The inspectors observed surveillance and related maintenance activities associated with the reactor protection system (RPS) EPA breakers. Several notable problems resulted during the performance of the surveillance (STP-41A127). Breaker EPA-A2 did not pass the undervoltage and underfrequency trip timing tests due to a failed logic board. The board could not be repaired onsite nor was a spare available onsite. The licensee sent the board offsite for repair while the "A" RPS bus remained on alternate power. During replacement of the logic board, a qualified resistor was broken, and a replacement was not available onsite. The licensee purchased a commercial quality resistor, dedicated it for use in the EPA breakers, and replaced the broken resistor.

During performance of the undervoltage, underfrequency, and overvoltage trip test, licensee technicians found some of the underfrequency and undervoltage trips on EPA-C1, EPA-C2, EPA-B1, and EPA-B2 breakers out-of-tolerance. Initially the technicians were unable to bring the trip time of these breakers within the TS tolerance of 100 -130 milliseconds. After further technical review, the licensee determined that the test method used by the technicians brought about inaccurate results. The trip function is measured on a recorder set to read input and output logic states. The input is changed by technicians manually turning a potentiometer to ramp the signal up to the trip level. If the technicians use a slow ramp rate to input the signal, the results of the test may be inaccurate. When the tests were reperformed by technicians using a faster ramp rate, the trip tests were within TS tolerances.

The licensee has contacted GE in order to determine a satisfactory and consistent method of testing the EPA breakers. The inspectors will continue to review the licensee's procedure enhancements to correct this problem. The inspectors also noted that the procedure, as written, did not allow the technicians to adjust the time response of the trip settings to the middle of the TS range. The time delay tolerance required by the STP is exactly the same as the TS tolerance (115 milliseconds ± 15 milliseconds). If the setpoint is within tolerance, the technician is required to proceed to the next step, without adjusting the time delay to the middle of the band. The inspectors noted one case (step 7.1.81) where the as left value of the time delay was 130 milliseconds. This practice does not allow any room for setpoint drift in between surveillance periods. The inspectors spoke with the testing and surveillance group about this problem, and were informed that the group would review this problem further.

Several steps in the procedure, such as 7.1.10, 7.1.55 and 7.3.10, require a voltage/frequency source to be connected, and an attenuator to be set to a X100 setting. While talking to the technician, the inspectors discovered that the source used did not have a X100 attenuation setting. The technician explained that after talking with the surveillance group and shift supervisor about the problem, he continued on with the procedure using an attenuator setting which was acceptable for the counter being used. The licensee did not process a procedure revision prior to proceeding as required by Procedure 1406.3, "Revision of Procedures and Instructions", and TS 6.8.3. The technician involved did write a procedure comment form describing the problem he had with the attenuator. The inspectors also reviewed a previous (1990) performance of the surveillance in which the same problem was present, and in that case the procedure was not changed and no comment form was issued. The inspectors discussed this discrepancy with the technician, the testing and surveillance supervisor, and the Assistant Plant Superintendent for Operations Support.

The licensee determined that the technician should not have proceeded without revising the procedure. Although the technician's discussion about the problem with the testing and surveillance group and the shift supervisor was intended to solve the problem, poor communication prevented the proper course of action from being taken. Licensee corrective actions for this procedure problem included counseling the individuals involved, giving additional training to the instrument and control shop and shift supervisors on procedural compliance, correcting the procedure to specify the proper attenuation value, and continuing to evaluate an easier way to use the procedure change process. Failing to use the procedure change process is a violation of TS 6.8.3 as implemented by licensee Procedure 1406.3 (331/91015-01 (DRP)). Due to the limited safety significance of this particular occurrence, and the satisfactory corrective actions of the licensee, this violation will not be cited, in accordance with 10 CFR 2, Appendix C, V.A.. Although the safety significance of this particular instance was minor because it involved test equipment, it still points out that continued efforts are necessary to ensure that management expectations are followed from the technician level, all the way through to supervision. The inspectors will closely follow the licensee's continued efforts in improving procedures and the procedure change process.

7. <u>Temporary Instruction 2515/112 - Licensee Evaluations Of Changes To The Environs Around Licensed Reactor Facilities (2515/112)</u>

Pursuant to NRC Temporary Instruction 2515/112, the inspectors reviewed plant programs and procedures for the Duane Arnold facility to determine if the licensee has implemented a program to periodically review, identify, and evaluate changes in site proximity hazards and demography to determine their effect on the safety of the plant. The inspectors also reviewed licensing information for the facility to determine if the licensee has updated the final safety analysis report (FSAR) to reflect changes to the licensing basis since the plant was licensed.

The objective of this inspection was to determine if the licensee's programs are adequate in evaluating public health and safety issues resulting from changes in population distribution or in industrial, military, or transportation hazards that could arise on or near reactor sites, and to determine if the licensee routinely documents these changes in updates to the FSAR. This information will be used to determine whether additional regulatory requirements are necessary in order to ensure that licensees periodically review public health and safety issues resulting from changes in population or industrial, military, or transportation hazards.

In reviewing Duane Arnold's technical specifications, UFSAR, and administrative programs, the inspectors noted that the licensee conducts a land use census as part of the radiological environmental monitoring program. The census is conducted annually to identify changes in use of the unrestricted area in order to recommend modifications in monitoring programs for evaluating individual doses from principal exposure pathways. While the census is performed annually per T.S. 4.16.B.2 to identify radiologically important changes in land use, informally the census provides insight into significant changes in the environs around the facility. In discussions with the individual who conducts the door-to-door inspection, the inspectors were informed that there appears to be no major changes in industrial, commercial, or military use within a three mile radius of the facility, including two sectors which were surveyed up to a five mile radius.

Additional discussions with licensee personnel indicate that programs currently in place evaluate the impact Duane Arnold has on the environs and not the impact the environs could have on the facility. However, the licensee stated that should they become aware of changes in the surrounding environs during the performance of the land use census or through any other means, they would consider the impact of the change on the facility. However, without a formal program there is no assurance that this change would be properly evaluated, dispositioned, and documented.

The inspectors also noticed in reviewing Duane Arnold's UFSAR, that the population and land use study described in Section 2.1.3, "Population Distribution and Land Use", was based on the data available from the 1970 census. This census data was used to provide projected data for the next 40 years, up to the year 2010. The inspectors did, however, note that the licensee compared 1980 census data, which showed relatively little change. Discussion with the licensee's emergency planning organization also indicated that the licensee is in the process of performing an evacuation time study utilizing the 1990 census data. The evacuation time study will be used to update evacuation times for the emergency planning zone (EPZ) and will also be used in preparation for the redefinition of the EPZ in 1993.

In conclusion, the inspectors determined that aside from the land use census, the licensee has no formal program to periodically review, identify, and evaluate changes in site proximity hazards and demography to determine their effect on the safety of the plant. This was based upon discussions with licensee personnel, review of T.S., UFSAR, and licensee programs. However, an informal assessment of the plant's environs by the inspectors showed that there appears to have been no significant changes in the area immediately surrounding the facility. Temporary Instruction 2515/112 is closed.

8. Report Review (90713)

During the inspection period, the inspectors reviewed the licensee's Monthly Operating Report for July 1991 and August 1991. The inspectors confirmed that the information provided met the requirements of Technical Specifications 6.11.1.C and Regulatory Guide 1.16.

No violations or deviations were identified in this area.

9. Severity Level V Non-Cited Violation (NCV)

During this inspection, certain of your activities, as described in paragraph 6 above, appeared to be in violation of NRC requirements. However, the violation was categorized at Severity Level V and it is not being cited because the criteria specified in Section V.A of the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy, 10 CFR Part 2, Appendix C, (1991), were satisfied.

10. Exit Interview (30703)

The inspectors met with licensee representatives (denoted in Section 1) on October 4, 1991, and informally throughout the inspection period and summarized the scope and findings of the inspection activities. The inspectors also discussed the likely information content of the inspection report with regard to documents or processes reviewed by the inspectors. The licensee did not identify any such documents or processes as proprietary. The licensee acknowledged the findings of the inspection.