NOTICE OF VIOLATION

AND

PROPOSED IMPOSITION OF CIVIL PENALTIES

Union Electric Company
Callaway Station
Unit 1

Docket No. 50-483
License No. NPF-30
EA 86-119

NRC inspections conducted during the periods April 14 - May 31, 1986 and June 4-11, 1986 identified violations of NRC requirements. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C (1986), the Nuclear Regulatory Commission proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended, ("Act"), 42 U.S.C. 2282, PL 96-295, and 10 CFR 2.205. The particular violations and associated civil penalties are set forth below:

I. A. Technical Specification 3.5.2 requires that two independent Emergency Core Cooling System (ECCS) subsystems be operable with each subsystem comprised of an operable flow path when the unit is in operational Modes 1, 2, and 3.

Technical Specification 3.0.3 requires that when a Limiting Condition for Operation is not met, except as provided in the associated Action requirements, within one hour action must be initiated to place the unit in a Mode in which the specification does not apply. When in Hot Standby (Mode 3) the unit must be placed in at least Hot Shutdown (Mode 4) within the following six hours and Cold Shutdown within the subsequent 24 hours.

Contrary to the above, on April 12, 1986, at 4:02 a.m., with the unit in Hot Standby, licensee personnel closed safety injection cold leg isolation valve EM-HV-8835 while performing a surveillance test. This action blocked the operable flow path and rendered both trains of the Intermediate Head Safety Injection System, an ECCS subsystem, inoperable. The valve remained closed until 10:10 a.m. The licensee did not take action within one hour to place the unit in Hot Shutdown (Mode 4) within the following six hours, and the valve remained closed until 10:10 a.m.

B. 10 CFR 50.72(b)(2)(iii) requires that a licensee notify the NRC within four hours of any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

Contrary to the above, on April 12, 1986 the licensee identified at 10:10 a.m. that both trains of the Intermediate Head Safety Injection System had been rendered inoperable, a condition that could have prevented the fulfillment of the safety function of a system needed to mitigate the consequences of an accident, and did not notify the NRC within four hours after identifying this condition.

II. Technical Specification 3.3.2 requires that the Engineered Safety Features Actuation System (ESFAS) instrumentation channels shown in Table 3.3-3 be operable. Table 3.3-3, Item 6.g requires that when the plant is operating in Modes 1 or 2, a minimum of three of four instrumentation channels be operable to start Auxiliary Feedwater Pumps (AFP) after the trip of all Main Feedwater pumps (MFP).

Contrary to the above, from 7:51 p.m. on May 30, 1986 until 6:48 a.m. on May 31, 1986 (a period of approximately 11 hours), the plant operated in Mode 2 with all four channels of the AFP ESFAS inoperable (blocked). As a result, the automatic start capability of the AFP on loss of the MFP was defeated.

Collectively, the above violations have been evaluated as a Severity Level III problem (Supplement I).
(Cumulative Civil Penalty - $25,000 assessed equally among the violations)

Pursuant to the provisions of 10 CFR 2.201, Union Electric Company is hereby required to submit to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 799 Roosevelt Road, Glen Ellyn, IL 60137, within 30 days of the date of this Notice a written statement or explanation, including for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps which will be taken to avoid further violations, and (5) the date when full compliance will be achieved. If an adequate reply is not received within the time specified in this Notice, the Director, Office of Inspection and Enforcement, may issue an order to show cause why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, Union Electric Company may pay the civil penalty by letter addressed to the Director, Office of Inspection and Enforcement, with a check, draft, or money order payable to the Treasurer of the United Stated in the cumulative amount of Twenty Five Thousand Dollars ($25,000) or may protest imposition of the civil penalty in whole or in part by a written answer

addressed to the Director, Office of Inspection and Enforcement. Should Union Electric Company fail to answer within the time specified, the Director, Office of Inspection and Enforcement, will issue an order imposing the civil penalty in the amount proposed above. Should Union Electric Company elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, such answer may: (1) deny the violation listed in this Notice, in whole or in part; (2) demonstrate extenuating circumstances, (3) show error in this Notice; or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the five factors addressed in Section V.B of 10 CFR Part 2, Appendix C, should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in rely pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. Union Electric Company's attention is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which has been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Keppler
Regional Administrator

Dated at Glen Ellyn, Illinois,
this 9th day of September, 1986.

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-483/86010(DRP)

Docket No. 50-483 License No. NPF-30

Licensee: Union Electric Company
Post Office Box 149 - Mail Code 400
St. Louis, MO 63166

Facility Name: Callaway Plant, Unit 1

Inspection at: Callaway Site, Steedman, MO

Inspection Conducted: April 15 through May 31, 1986

Inspectors: B. H. Little

C. H. Brown

W. L. Forney
for P. R. Wohld

6/14/86
Date

Approved by: W. L. Forney
W. L. Forney, Chief
Reactor Projects Section 1A

6/15/86
Date

Inspection Summary

Inspection on April 15 through May 31, 1986 (Report No. 50-483/86010(DRP))
Areas Inspected: A routine unannounced safety inspection by the resident inspectors and one Region III inspector of licensee actions on previous inspection findings, licensee event reports followup, followup on regional requests, TMI NUREG-0737 items closure, inspection of licensee events, monthly surveillance, operational safety verification, monthly maintenance, Cycle 2 startup.
Results: Two unresolved items relating to EQ of Limitorque Valves discussed in Paragraph 4. Two violations were identified in Paragraph 6., failure to maintain intermediate head safety injection operable, and failure to notify the NRC within four hours of an event.

DETAILS

1. Persons Contacted

D. F. Schnell, Vice President, Nuclear
S. E. Miltenberger, General Manager, Nuclear Operations
*G. L. Randolph, Manager, Callaway Plant
C. D. Naslund, Manager, Operations Support
A. P. Neuhalfen, Manager, Quality Assurance
*J. D. Blosser, Assistant Manager, Operations & Maintenance
*J. R. Peevy, Assistant Manager, Technical Services
P. T. Abbleby, Assistant Manager, Support Services
W. F. Powell, Assistant Manager, Materials
M. E. Taylor, Superintendent, Operations
D. E. Young, Superintendent, Maintenance
W. R. Robinson, Superintendent, I&C
R. R. Roselius, Superintendent, Health Physics
V. J. Shanks, Superintendent, Chemistry
J. A. Ridgel, Superintendent, Radwaste
G. J. Czeschin, Superintendent, Planning & Scheduling
W. H. Sheppard, Superintendent, Outages
J. M. Price, Superintendent, Training
G. R. Pendergraff, Superintendent, Security
J. E. Davis, Superintendent, Compliance
D. W. Capone, Manager, Nuclear Engineering
W. R. Campbell, Assistant Manager, Nuclear Engineering
A. C. Passwater, Superintendent, Licensing
T. H. McFarland, Superintendent, Design Control
R. D. Affolter, Superintendent, Systems Engineering
D. C. Poole, Consultant
W. H. Stahl, Supervisor, Engineering
*B. K. Stanfield, Assistant Engineer
*S. Petzel, Engineer
*W. R. Bledsoe, Engineer, Compliance

*Denotes those present at one or more exit interviews.

In addition, a number of equipment operators, reactor operators, senior reactor operators, and other members of the quality control, operations, maintenance, health physics and engineering staffs were contacted.

2. Licensee Actions on Previous Inspection Findings (92701)

(Closed) Open Item (483/84-48-01(DRP)): Licensee plans for modification of the pressurizer power operated relief valves (PORV). Based on problems identified during testing of the PORVs at Wolf Creek Generating Station, and the similarity of the Callaway and Wolf Creek Units, Westinghouse issued a Field Change Notice (FCN) No. SCPM 10712 for Callaway to inspect and record measurements of annular orifice gaps formed by the valve body

and valve cage. Valve modification (machining of the cage rib) is subject to inspection findings of inadequate clearances. The licensee included this FCN in Callaway Modification Package CMP-84-0651A, and accomplished the work during the Cycle 1 refueling outage.

The licensee's inspection determined that machining of the valve's cage rib was required. The work was accomplished under Work Request Nos. 54996 and 54997. The inspector reviewed the applicable WRs including the quality control inspection records. The inspector determined that the specified work was completed and the valves were successfully tested.

(Closed) Open Item (483/85006-01(DRSS)): During inspection 50-483/84-16, the licensee agreed to calibrate their gaseous and liquid effluent monitors with appropriate gaseous and liquid sources during the first refueling outage. These calibrations would be in addition to the solid source calibrations currently performed. The monitors involved include the radwaste building monitor GH-RE-10B, the unit vent gas monitor GT-RE-21B, and the liquid radwaste effluent monitor HB-RE-18.

Inspection in this matter included the review of Callaway vendor procedures HTP-ZZ-04154, HTP-ZZ-04155 and HTP-ZZ-04156 and the calibration reports issued by Radiological and Chemical Technology, Inc.

The licensee's calibration of the gaseous and liquid effluent monitors has been completed.

(Open) Unresolved Item (483/85007-01(DRS)): Closure surveillance testing of normally closed check valves that perform a safety function in the closed position. Per a previous agreement (as reported in Inspection Report No. 50-483/86012(DRS)) the licensee provided a list of all safety related check valves at the Callaway Plant for inspector review. A preliminary, onsite review of the check valve list was done in an attempt to categorize each valve as tested or not and to determine at least one obvious closure requirement for each valve. The results of this review indicate that there are a number of check valves not being closure tested that should be.

In the listing total of 225 safety related check valves, the review results were as follows:

<u>There are 143 check valves not tested for closure</u>

8 - Prevent reverse flow through an idle pump in parallel pump combinations.

22 - Isolate seismic/non-seismic pipe boundaries or different pipe classes (such as those which define the LOCA/non-LOCA pipe boundaries).

8 - Direct auxiliary feedwater flow to the proper steam generator during postulated accident conditions.

10 - Prevent overpressurization of auxiliary feedwater pump suction piping.

4 - Prevent blowdown from one interconnected steam generator to another during steam fault conditions.

23 - Had no apparent safety related closure requirement.

25 - Did not appear to be properly listed as safety related (apparent listing error) or the valve is not used (internals removed).

43 - Were not identified either as having or not having a closure safety function during the inspection because their function was not apparent from a cursory drawing review.

There are 65 valves tested for closure

10 - Are normally open check valves which are tested per the ASME Code, Section XI, closure stroke test requirements.

31 - Are tested to provide a high/low piping interface isolation (includes WASH 1400, Event V valves).

14 - Are leak tested for containment isolation.

10 - Are tested per the concerns raised by the inspector under this unresolved item.

Others (17)

2 - Are visually inspected for "free swing" on a periodic basis.

13 - Are fire protection system check valves which are covered by separate fire protection requirements and inspections.

2 - Are essential service water check valves that are not tested but the failure of which would be readily identified by the affect on normal system operation.

The inspector emphasized that the onsite check valve review was only a "quick look" and that an in-depth evaluation is needed by qualified system engineers. The licensee agreed to perform and document a review of the check valves on the list for their required closure functions and closure test requirements. The staff indicated that this would be available for inspector review in approximately 90 days.

No violations or deviations were identified.

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

Through direct observations, discussions with licensee personnel, and the review of records, the following LERs were reviewed to determine that the

events were documented and evaluated, reportability requirements were fulfilled, and appropriate corrective measures had been implemented.

(Closed) LER 85-010-00: Reactor Trip on Partial Loss of Feedwater Flow. On February 21, 1985, a reactor trip and associated actions occurred due to low steam generator water levels. The low water levels occurred when power was secured to a feedwater control panel which in turn resulted in one main feedwater pump shutting down. The remaining feedwater pump does not have the capacity for 100% reactor power. The loss of power to the control panel was due to the failure of the supply transformer. The transformer was replaced and a plant recovery was made. The transformer was found to contain foreign material and a procedure was issued to inspect and clean the transformers on an 18-month schedule. The transformer maintained the design safety function of isolation; therefore, a Part 21 report was not issued.

(Closed) LER 85-021-00: Inadequate Seismic Qualification of Class IE Batteries. On April 4, 1985, the licensee was notified of a potential problem relative to the spacing between the Class IE batteries and the battery racks. The immediate corrective action was to insert spacers between the end cells and the battery rack end stringers. The licensee and contractor evaluated the corrective action and considered the spacers a permanent fix.

(Closed) LER 85-023-00: Inadvertent Engineered Safety Features Actuation. At three different times, April 13, April 17, and May 6, 1985; inadvertent containment purge isolation and control room ventilation isolation signals were received. The cause was a faulty vacuum transducer in a radiation monitor. With a joint effort between the monitor vendor and several plants that had experienced similar problems, a more reliable transducer was developed and installed. The modification has apparently solved the problem.

(Closed) LER 85-025-02: Intermediate Range Hi Flux Reactor Trip. On May 6, 1985, a reactor trip and associated actions occurred during a reactor startup (power 0% and in Mode 2) due to a intermediate range high flux signal. The spurious signal was caused by a fuse blowing in the neutron monitoring channel. The vendor determined that a faulty switch caused the fuse to blow. The switch replacement appears to have solved the problem, but the failure mode of the switch and other possible causes that would overload the fuse are still being evaluated.

(Closed) LER 85-026-00: High Negative Flux Rate Reactor Trip. On June 7, 1985, a reactor trip occurred from 100% due to a high negative flux rate. The negative rate was generated due to rod drop when four rod control power supplies failed (thyristor bank insulator failure) during trouble-shooting an immovable control rod. The failed equipment was replaced and the review shows the failure to be an isolated occurrence and no further action was to be taken. The immovable control rod was due to a loose terminal screw.

(Closed) LER 85-042-00: Inadvertent Reactor Trip. On October 2, 1985, a reactor trip from 100% power occurred due to personnel error. Instrument and Control personnel were performing test ISP-BB-0T002, "RTD Calibration Verification" (a one time test) when an abnormal resistance reading was found on the No. 4 RCS loop, the loop was in test condition, and trouble-shooting was commenced. Loop No. 1 RTD terminal block test point was erroneously taken from the prints. When the troubleshooting process checked these terminals a signal was induced in loop No. 1 RCS protection which completed the 2 out of 4 logic for the reactor trip from over Temperature Delta T and Over Power Delta T signals. The failure to notify the shift supervisor before commencing troubleshooting was a failure to follow plant policy procedures. The error in reading the print contributed to the event. The test procedure did not allow troubleshooting if a problem was located. The test procedure also had an error which resulted in the abnormal resistance reading. The personnel involved were counseled on procedure and policy compliance.

No citation was issued since under the Enforcement Policy this was considered a Technical Specification violation of lesser severity which was identified and satisfactorily corrected by the licensee, and no further violations of a similar nature have occurred. This item is considered closed.

(Closed) LER 85-043-00: Technical Specification Hourly Firewatch Patrol Missed. The Technical Specification 3.7.10.2 requires hourly firewatch patrols to be established within one hour. On October 3, 1985, the hourly patrol was not established for one hour and twenty-five minutes in the south electrical cable chase due to a misunderstanding of the firewatch personnel. The firewatch personnel were retrained on T/S requirements for the firewatch patrols. Also the operations personnel, if possible, verify the patrol is established before taking a fire protection system out of service.

No citation was issued since under the Enforcement Policy this was considered a T/S violation of lesser severity which was identified and satisfactorily corrected by the licensee, and no further violations of a similar nature have occurred. This item is considered closed.

(Closed) LER 85-045-00: Technical Specification 3.7.10.2 Violation Due to Personnel Error. On October 16, 1985, a portion of the sprinkler system for the auxiliary building 2000 feet elevation cable trays was not identified as inoperable during surveillance testing. Therefore, a continuous firewatch per Technical Specification 3.7.10.2 was not established for about 17 hours, although a hourly firewatch did patrol these areas for this time period. The delay was due to electricians and engineers failing to communicate and recognize that sprinkler alarms were also present on the multiplexer that was being worked. The sprinkler system was inoperable due to a failed "supervision actuation module" which indicated the loss of the ability to actuate the pre-actuation sprinkler system in these areas.

The surveillance procedure MSE-KC-FW001, "Technical Specification (T/S) Fire Detection Functional and Supervisory Operability Test", has been revised to include the alarms that should be actuated or cleared during the applicable steps of the procedure. The maintenance department was provided a set of the electrical prints for the fire protection systems and the electricians received training on the fire protection system. The fire detection in these areas, which provide an alarm in the control room, were functional during the time the above sprinkler system was inoperable.

No citation was issued since under the Enforcement Policy this was considered a Technical Specification violation of lesser severity which was identified and satisfactorily corrected by the licensee, and no further violations of a similar nature have occurred. This item is considered closed.

(Closed) LER 85-047-00: Operation with a Condition Prohibited by Technical Specifications. On October 18, 1985 at 1145 CST, the plant entered Technical Specification 3.0.3 due to both centrifugal charging pumps (CCPs) being inoperable. "A" CCP had been taken out of service for maintenance and later "B" CCP's room cooler fan was discovered to have broken drive belts. The "A" CCP was made operable in less than an hour and preparations for a plant shutdown were suspended. The belts were replaced the following day. This was considered to be caused by equipment failure.

No citation was issued since under the Enforcement Policy this was considered a Technical Specification violation of lesser severity which was identified and satisfactorily corrected by the licensee, and no further violations of a similar nature have occurred. This item is considered closed.

(Closed) LER 85-050-00: Inadvertent Engineered Safety Features Actuation. On November 27, 1985, a control room ventilation isolation and a containment purge isolation occurred. The cause appeared to be a fuse failure at the microprocessor for the containment purge radiation monitor and a tripped breaker supplying power to the monitor's flow pump. The containment was not being purged at the time and the redundant monitor remained operable. Troubleshooting found no equipment damage. The fuse was replaced and the breaker closed with no further problem.

(Closed) LER 86-008-00: Technical Specification Violation. On April 3, 1986, while the plant was in Mode 5 (Cold Shutdown), the licensee determined that Train "A" Control Room Emergency Ventilation System (CREVS) had been inoperable since March 18, 1986. The "A" Train became inoperable when the air conditioning unit was deenergized to permit inspection and repair of the CREVS fire dampers. Technical Specification (T/S) 3.7.6, Action Statement requires that when in Mode 5 or 6; "With one Control Room Emergency Ventilation System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency Ventilation System in the recirculation mode".

On April 3, 1986, in response to a work request by maintenance personnel to commence work on the fire dampers, control room personnel found that the air conditioning unit ("A" Train) had been tagged out since March 18, 1986, without having placed the "B" Train CREVS in the recirculation mode. This action should have been taken by March 25, 1986. Control room personnel immediately placed the "B" Train CREVS in the recirculation mode and documented the violation on Incident Report No. 86-096.

The licensee's evaluation of the event determined the root cause to be operations personnel's initial failure to correctly assess the "operability" impact of tagging out the air conditioning unit. Consequently, the "A' Train was not declared "inoperable" (no entry in the equipment out of service logs).

To prevent recurrence, a procedural change was written to require an independent review of WPAs, Equipment Out of Service Logs, Temporary Modifications and Locked Components for each ascending mode change. Also, personnel involved were re-instructed concerning T/S operability requirements. This event has been included in the licensed operator requalification program, "Lessons Learned".

The inspector determined that, once identified by the licensee, action was promptly taken to correct the condition and report the violation. During the period that the air conditioning unit was out of service, control room temperatures were maintained below 84 degree F (the temperature specified in T/S 4.7.6.e.). The Train "B" CREVS and Train "A" pressurization and filtration systems were operable and would have protected control room personnel from airborne contamination if needed. The event posed no threat to public health and safety.

No citation was issued since under the Enforcement Policy this was considered a Technical Specification violation of lesser severity which was identified and satisfactorily corrected by the licensee, and no further violations of a similar nature have occurred. This item is considered closed.

(Closed) LER 86-015-00: Auxiliary Feedwater Actuation System (AFAS): When PK51 Feeder Breaker Was Inadvertently Tripped. On April 23, 1986, while in Mode 1 with reactor power at 31%, the control room received numerous alarms on the annunciator boards. Letdown and makeup to the volume control tank was lost, excess letdown and AFAS was initiated. Plant conditions were determined to be stable, and no reactor trip occurred.

The licensee control room personnel's initial investigation determined that a non-vital power breaker PK51 had been manually opened. Based on no work having been authorized involving PK51, the event was considered as possible tampering. The control room personnel promptly notified the shift security supervisor and the plant manager. After hearing public address (PA) instructions for the shift security officer to contact the control room, the NRC resident inspector responded by going to the control room.

The inspector observed licensee's immediate response to the event which included additional security measures being implemented. The inspector was advised of licensee's planned investigation in this matter.

On April 24, 1986, a radchem technician acknowledged to the licensee that he had inadvertently opened breaker PK51 on April 23, 1986. The technician was a contract employee during the recent Cycle 1 refueling outage and following the outage he was hired by Union Electric.

The NRC senior resident inspector and Region III security specialist interviewed members of the licensee's staff, including the technician involved in the event. The inspectors were satisfied that the information provided by the technician, with regard to operating switch PK51, was in agreement with control room observations during the event.

The licensee's investigation determined that the event resulted from unauthorized operation of breaker PK51. The technician's action, although well meaning, highlighted existing weaknesses in the licensee's "new hire" indoctrination program. Specifically, organizational interface, departmental authority/responsibility, and administrative controls on work and safety practices.

The licensee has implemented a radwaste "Indoctrination Checklist for New Personnel", and revised the general employee training program to assure responsibility/authority for equipment operation is stressed. This incident will be included in the licensee's requalification training program.

The inspector determined that the licensee response was prompt and thorough and that action has been taken to prevent recurrence.

No other violations were identified other than those noted above that were identified, reported and corrected by the licensee.

4. Followup on Regional Requests (92701)

a. Temporary Instruction (TI) 2515/75, "Inspection of Limitorque Motor Valve Operator Wiring"

An inspection was performed to ascertain the environmental qualification (EQ) of wiring used in Limitorque Motor Valve Operators. The inspection included the following:

Physical inspection of Limitorque operator wiring to determine what wiring is actually installed in the operators.

Review of licensee's environmental qualification documentation to ensure qualification of wiring is adequately established.

Review of licensee's action relative to IE Information Notice (IN) 86-03, "Potential Deficiencies In Environmental Qualification of Limitorque Motor Valve Operator Wiring".

Physical Inspection: On May 22, 1986, the inspector performed an in plant inspection of four Limitorque Motor Valve Operators. The inspector selected the below listed valves based on a review of SNUPPS Final Safety Analysis Report (FSAR), Table 3.11(B)-3, "Identification of Safety-Related Equipment and Components". Three valves selected are located in the reactor containment building and one is located in the lower piping penetration room of the reactor auxiliary building. The inspector was accompanied by representatives from licensee's Quality Assurance, Quality Control, Engineering, and Maintenance Departments.

VALVE NO.	DESCRIPTION
EP-HV-8808B	Safety Injection Tank Outlet Isolation Valve "B"
EP-HV-8808C	Safety Injection Tank Outlet Isolation Valve "C"
BB-HV-8037B	Pressurizer Relief Tank (PRT) Outlet Isolation Valve
EM-HV-8835	Safety Injection Discharge to Cold Leg Injection Isolation Valve

Inspection Findings:

VALVE NO.	*TERMINAL WIRING	*FIELD WIRING
EP-HV-8808B	Raychem Flamtrol	G34 C02, C03, C04 G2A
EP-HV-8808C	Raychem Flamtrol	G34 C02, C03, C04 G2A BRAND Rex #43
BB-HV-8037-B	Rockbestos Firewall III	G31 C07
EM-HV-8835	Raychem Flamtrol	G31 C02, C04

*Each motor operator contained some terminal and field wires which either lacked identification markings or with unreadable markings. However, these wires were similar (size/color) to other wires identified by markings. In addition, limit switch space heater wiring was unmarked, approximately 20 AWG size. SNUPPS Report of Independent Review of EQ Programs (Response to NUREG 0588) states: "In all cases, the limit switch space heater is connected in a Class IE circuit. Since the heater failure mode will result in an open circuit, it is considered that the heaters need not be qualified. However, Limitorque has performed an accident test

on a heater to demonstrate that the heater remains operative following seismic aging and a simulated LOCA (i.e., it would not fail in a manner detrimental to plant safety.)

Review of Licensee's EQ Documentation

The inspector reviewed the below listed EQ Test Reports and determined that the operator wiring in the four operators inspected had been environmentally qualified.

EQ TEST REPORT NOS.	TERMINAL WIRING
Franklin Institute Research Laboratories Test Nos. E-031.2, E-031.3	Raychem Flamtrol
Rockbestos Test Report Nos. E-057-020-03, E-057-021-06, E-057-036-02, E-057-050-02	Rockbestos Firewall III
	FIELD WIRING
Rockbestos Test Report Nos. E-057-020-03, E-057-021-06, E-057-036-02, E-057-050-02	C02, C03, C04, C07
ANACONDA Test Report No. E-58-0005-03	G2A, G31, G34
BRAND Rex Test Report No. E-057B-0014-02	BRAND Rex

The inspector reviewed licensee plant walkdown sheets to determine if licensee's identification of operator wiring was in agreement with the inspectors findings for the four operators inspected. The licensee's quality records of plant walkdowns included inspections performed through Startup Work Requests (SWRs) and Quality Control (QC) checklists. The licensee records were in agreement with the inspector's findings.

Licensee's Response to IE Information Notice 86-03

In January 1986, the licensee performed a Quality Assurance (QA) surveillance on EQ of Limitorque Motor Valve Operator wiring (QA Surveillance Report No. P8601-12). The surveillance determined that each operator was field inspected prior to initial plant startup using a Startup Work Request (SWR). QA sample inspection of approximately 10% of the SWRs determined that the operator wiring was environmentally qualified.

Union Electric Nuclear Engineering (UENE) in response to Information Notice 86-03 performed a review of all SWRs relating to the field inspection of Limitorque Valve Operators. This review identified six valves as having suspect internal wiring. Nuclear Engineering

memorandum No. 558, dated March 11, 1986, requested site engineering to perform a field inspection to identify the installed wiring. The results of the field inspection was as follows:

VALVE NOS.	WIRING	EQ STATUS
BB-HV-8000A	Rockbestos	Qualified
BB-HV-8000B	Raychem	Qualified
EJ-HV-8701A	Raychem	Qualified
EJ-HV-8716A,B	Not Identifiable	Questionable
EJ-HV-8809B	Techbestos 14 AWG 600-V	Questionable

The licensee has replaced the operator wiring associated with EJ-HV-8716A,B and EJ-HV-8809B with environmentally qualified wiring and has requested Westinghouse response regarding the EQ status of the wiring removed.

The inspector determined that the licensee was responsive to IE Information Notice 86-03 and took prompt corrective action to ensure that the installed operator wiring is environmentally qualified. The licensee's evaluation of the EQ status of the wiring replaced is in progress. This matter is unresolved pending further NRC review. Unresolved Item No. 483/86010-01(DRP)

Motor Operated Valves (MOV) Conduit Seals

On May 28, 1986, the licensee advised the inspector that an engineering department review of construction documentation was unable to establish that all required containment MOV conduit seals were in place. The licensee performed the review in response to conduit seal deficiencies identified at the Wolf Creek Plant. The licensee determined that although work authorizing documents had been issued, there was no sign offs for work accomplishment.

On May 23, 1986, while the plant was in Mode 5 (Cold Shutdown), the licensee issued Work Request Nos. 60511 through 60519 to install containment MOV conduit seals in accordance with Bechtel Drawing M-2Y007 (Conduit Seals for Containment MOVs). This action was taken to assure existing EQ status of the MOVs prior to pending plant startup. The licensee stated that external visual inspection could not readily verify conduit sealing in accordance with the design drawing, as drawing M-2Y007 requires or does not require the use of sealant depending on the actual field routing of the conduit. The licensee's evaluation of this matter is continuing and plans to perform a field inspection during the next shutdown.

The licensee reviewed Bechtel Drawing M-2Y007, SNUPPS Report of Independent Review of EQ Programs, and WR Nos. 60511 through 60519. The inspector also interviewed licensee maintenance personnel that performed the conduit sealing work. Based on this review, the inspector determined that the containment MOVs conduit seals have been installed in accordance with Bechtel Drawing M-2Y007. However,

the acceptability of conduit seals prior to May 23, 1986, could not be determined. This matter is unresolved pending further NRC review. Unresolved Item No. 483/86010-02(DRP)

b. Temporary Instruction (TI) 2515/77, "Licensee Response to Selected Safety Issues (Biofouling of Cooling Water Heat Exchangers)

An inspection was performed to assess licensee's programs for detection and prevention of biofouling of cooling water heat exchangers. The inspection included a review of applicable procedures and interviews with licensee's maintenance, chemistry, engineering, and operations personnel.

Prior to the initial startup, the licensee detected tube damage due to biofouling in the main generator hydrogen coolers which was attributed to stagnant water conditions prior to plant startup. No additional biofouling has been experienced. The licensee has implemented procedures; ETP-ZZ-03002, "Performance Testing of Plant Heat Exchangers" and ETP-ZZ-03003, "Monitoring of Plant Heat Exchangers". The licensee also maintains 1 PPM chlorine in the service water system as a preventive measure. The licensee provides procedures and operator training relating to degraded heat exchanger performance.

No violations or deviations were identified.

5. TMI NUREG-0737 Items Closure (92705)

The following TMI NUREG-0737 line items are considered to be closed:

II.B.3.3
II.B.3.4
II.F.1
II.F.2A
II.F.2B
II.F.2C

A review of Inspection Report Nos. 84-10(DRMSP), 84-16(DRMSP), 86004(DRSS), and others was made and discussions were held with the applicable inspectors to verify that these line items were completed. The item identification is included here as a correlation for NUREG-0737 tracking system as the previous closeouts were for the Safety Evaluation Report tracking system or other numbering systems.

No violations or deviations were identified.

6. Inspection of Licensee Events-Inoperable Intermediate Head Safety Injection (IHSI) System (92700)

a. Background

On April 13, 1986, the licensee advised the senior resident inspector that on April 12, 1986, while in Mode 3 (Hot Standby), the plant was placed in a condition prohibited by Technical Specification (T/S) 3.5.2, when the IHSI system was inadvertently isolated. The inspector was given a copy of Incident Report No. 86.109 which documented the violation. The inspector was also briefed on the event, the cause and immediate action taken, and of licensee's planned investigation in this matter.

An inspection was performed to assess the event, root cause/contributing factors, and licensee corrective measures. The inspection included a review of event reports, operating logs, administrative and surveillance procedures, personnel interviews, and meetings with licensee management.

b. Inspection Findings

On April 12, 1986, at 0402 CST, the safety injection (SI) cold leg isolation valve EM-HV-8835 was closed to perform surveillance test OSP-EP-V0003 (Section XI Accumulator Safety Injection Valve Operability). Technical Specification (T/S) Limiting Condition for Operation 3.5.2 specified that the IHSI system be operable in Mode 3. T/S Surveillance Requirements 4.5.2 specifies that EM-HV-8835 (Safety Injection Cold Leg Isolation Valve) be open. EM-HV-8835, being closed, isolated the common discharge path from both SI pumps to the cold leg injection, putting the plant in a condition prohibited by Technical Specifications.

On April 12, 1986, at 1010 CST, the reactor operator, while taking the required daily log readings, observed that EM-HV-8835 was closed. The reactor operator immediately informed the shift supervisor (S/S). The S/S declared both SI Trains inoperable and entered T/S 3.0.3 and had valve EM-HV-8835 opened. The S/S issued Incident Report No. 86-109 documenting the violation.

On May 7, 1986, the inspector met with the licensee to assess licensee's investigation, evaluation of cause, and corrective actions regarding the IHSI system isolation. The licensee discussed their findings of root cause and contributing factors and of corrective action taken and planned. The cause of the event was attributed to personnel scheduling and performance errors as follows:

Scheduling

Scheduling Personnel - Scheduled the performance of OSP-EP-V0003, "as required in Mode 3 prior to RCS pressure reaching 1000 psig".

Compliance Personnel - Identified OSP-EP-V0003 on an attachment to the Mode 3 Change Letter, "to be performed in Mode 3 as conditions permit".

The task performance review considered the operability requirement of the safety injection accumulators but failed to recognize the Surveillance Task Sheet (STS) task performance mode requirements, which specified Mode 4 only.

c. Performance

Operations personnel erroneously authorized and performed OSP-EP-V0003 in Mode 3. Several errors were made regarding the authorization and performance as follows:

(1) OSP-EP-V0003 - Initial condition specified performance of the test in Mode 4. Operations personnel issued a Temporary Change Notice (TCN) changing Mode 4 to Mode 3. APA-ZZ-00101 (Preparation, Review, Approval and Control of Plant Procedures) provides for temporary procedure changes, "which clearly does not change the intent". Management Directive UO 86-69 issued March 4, 1986, reemphasized(s) control of TCNs and identified that "significant changes to initial conditions are changes which are changing the intent".

(2) Operations personnel changed the Surveillance Task Sheet No. ST-00070 Task Performance Mode from "Mode 4 only" to Mode 3. This change did not receive the required review and approval as specified in APA-ZZ-00340 (Surveillance Program Administration).

(3) Operations personnel's failure to be cognizant of the overall plant effect of closing EM-HV-8835. (The isolation of a required safety system).

(4) In addition to issuing an Incident Report, the licensee classified the event as a 30 day Licensee Event Report (10 CFR 50.73). However, a four hour report to NRC Operations Center should also have been made in accordance with 10 CFR 50.72(b)(2)(iii). This report was not made. Failure to notify the NRC within four hours is a violation of 10 50.72(b)(2)(iii). No. 483/86010-03(DRP).

d. Licensee's Corrective Action to Prevent Recurrence Included:

(1) For future outages, outage scheduling will schedule OSP-EP-V0003 in Mode 4 as a Mode 3 restraint.

(2) Progressive discipline has been initiated for appropriate outage personnel. Outage Planning and Scheduling personnel have been advised concerning outages involvement in this event.

(3) An outage procedure currently in draft form will specifically address use of the STS "Task Performance Mode" for scheduling surveillances.

(4) Future mode change letters will reflect only required task performance conditions and T/S requirements for mode changes.

(5) The TCN that allowed performance of the OSP in Mode 3 was voided.

(6) Progressive discipline has been initiated for operations personnel involved in this event and the necessity to comply with programmatic controls has been reemphasized.

(7) Management will reemphasize the existing administrative controls for revisions to task sheets and surveillance procedures to appropriate plant personnel.

(8) Appropriate personnel will receive guidance concerning reporting requirements of 10 CFR 50.72.

The inspector determined that the violation, once identified by the licensee, was promptly corrected, documented, and received a high level of attention. Based on the short duration of the violation, plant conditions of low temperature and pressure with low stored heat energy and the availability of backup emergency core cooling systems, the event posed no significant threat to the public or plant safety. However, the event highlighted significant performance errors. These errors included; inadequate task reviews, failure to adhere to licensee administrative procedures, and failure to be cognizant of the overall plant effect resulting from surveillance testing.

The licensee's failure to maintain the IHSI system "operable" while in Mode 1, 2, and 3 is a violation of Callaway Plant Technical Specifications Limiting Condition for Operating 3.5.2. No. 483/86010-04(DRP).

7. Monthly Surveillance (61726)

The inspectors reviewed or observed selected portions of Technical Specification required surveillance testing during power operations and prior to mode changes relative to the startup from the refueling outage.

Items which were considered during the inspections included whether adequate procedures were used to perform the testing, test instrumentation was calibrated, test results conformed with Technical Specifications and procedural requirements, and the test was performed within the required time limits. The inspector determined that the test results were reviewed by someone other than the personnel involved with the performance of the test, and that any deficiencies identified during the testing were reviewed and resolved by appropriate management personnel.

No violations or deviations were identified.

8. Operational Safety Verification (71707)

The inspectors observed control room operations, reviewed applicable logs, and conducted discussions with control room operators throughout the inspection period. The inspector verified the operability of selected safety related systems, reviewed tagout records, and verified proper return to service of affected components. Tours of the reactor, auxiliary, and turbine buildings were conducted. During these tours, observations were made relative to plant equipment conditions, fire hazards, fire protection, adherence to procedures, radiological control and conditions,

housekeeping, security, tagging of equipment, ongoing maintenance and surveillance, containment integrity, and availability of safety related equipment.

No violations or deviations were identified.

9. Monthly Maintenance (62703)

Selected portions of the plant maintenance activities on safety related systems and components were observed or reviewed to ascertain that the activities were performed in accordance with approved procedures, regulatory guides, industry codes and standards, and that the performance of the activities conformed to the Technical Specifications.

The following items were considered during these inspections: 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 calibrating were performed prior to returning the components or systems to service; parts and materials that were used were properly certified; radiological controls were implemented as necessary; and, fire prevention controls were implemented.

No violations or deviations were identified.

10. Cycle 2 Startup (61702, 61705 through 61710)

The initial criticality of the Cycle 2 core was observed by the inspector on April 15, 1986. The startup was performed per ETP-ZZ-ST002, "Engineering Test for Initial Criticality". Selected portions of the following tests/ procedures were observed during their performance and the results were reviewed after the evaluation of data was completed.

ETP-ZZ-00007 Reactimeter Dynamic Checkout

ETP-ZZ-ST004 All Rod Out Boron Endpoint

ETP-SR-ST001 All Rods Out Flux Map

ESP-ZZ-00009 Moderator Temperature Coefficient Measurement

ESP-BB-03015 Reactor Coolant Flow Measurements

ESP-ZZ-00006 Incore/Excore Calibration

ETP-ZZ-ST005 Rod Bank Worth Measurement

The moderator temperature coefficient was slightly positive for all rods out. The rod withdrawl restriction will continue for 4000 MWD/MTV burnup of the core. The other tests indicated the results were about where they were expected. The mode changes were observed and requirement check sheets were reviewed. Selected requirements for mode changes were verified to have been performed.

No violations or deviations were identified.

11. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain wheter they are acceptable items, violations, or deviations. Two unresolved items disclosed during the inspection are discussed in Paragraph 4.

12. Exit Interview

The inspector met with licensee representatives (denoted under Persons Contacted) at intervals during the inspection period. The inspector summarized the scope and findings of the inspection. The licensee representatives acknowledged the findings as reported herein. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.

13. Enforcement Conference

An Enforcement Conference was held on June 3, 1986, at the NRC Region III office, Glen Ellyn, Illinois between Mr. D. F. Schnell and members of the NRC Region III staff. During the meeting the Licensee presented facts relative to the event on April 12, 1986, discussed in Paragraph 6 above. The Licensee presented background information, corrective action to prevent recurrence, and potential mitigating facts which the NRC will use to determine the appropriate escalated enforcement action.

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-483/86017(DRP)

Docket No. 50-483 License No. NPF-30

Licensee: Union Electric Company
Post Office Box 149 - Mail Code 400
St. Louis, MO 63166

Facility Name: Callaway Plant, Unit 1

Inspection At: Callaway Site, Steedman, MO

Inspection Conducted: June 4 through 11, 1986

Inspector: William L. Forney for B. H. Little — June 24, 1986 Date

Approved By: William L. Forney W. L. Forney, Chief Reactor Projects Section 1A — June 24, 1986 Date

Inspection Summary

Inspection on June 4 through 11, 1986 (Report No. 50-483/86017(DRP))
Areas Inspected: A special unannounced safety inspection by the Senior Resident Inspector regarding the blocking of auxiliary feedwater actuation instrumentation and NRC Region III Management site visit.
Results: One apparent violation was identified (failure to maintain the Auxiliary Feedwater Actuation System instrumentation operable while in Modes 1 and 2 - Paragraph 2.).

DETAILS

1. Persons Contacted

D. F. Schnell, Vice President, Nuclear
*S. E. Miltenberger, General Manager, Nuclear Operations
*G. L. Randolph, Manager, Callaway Plant
*C. D. Naslund, Manager, Operations Support
*A. P. Neuhalfen, Manager, Quality Assurance
*J. D. Blosser, Assistant Manager, Operations & Maintenance
*J. R. Peevy, Assistant Manager, Technical Services
*M. E. Taylor, Superintendent, Operations

*Denotes those present at one or more management interviews.

In addition, a number of reactor operators, senior reactor operators, and other operations department personnel were contacted.

2. Inspection of Auxiliary Feedwater Pumps (AFP) Engineered Safety Features Actuation System (ESFAS) Blocking

a. Background

On May 31, 1986, the licensee determined that the plant was in a condition prohibited by Technical Specification (T/S) Limiting Conditions for Operation (LCO) 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation". The violation occurred when the Auxiliary Feedwater Pumps (AFPs) ESFAS Blocking Switches (FC-HS-25 and FC-HS-26) were left in the "Block" position after the first main feedwater pump was put in service. The "Block" position placed the Auxiliary Feedwater Actuation System-Motor (AFAS-M) in a degraded condition such that the automatic start of the AFPs on loss of the main feedwater pumps was defeated. The prohibited condition existed for approximately 11 hours when, during a control board inspection, a reactor operator observed that the ESFAS blocking switches were in "Block". The ESFAS blocking switches were then placed in "permit" restoring operability of the AFAS-M, and satisfying the operability requirements of T/S 3.3.2.

The licensee issued Incident Report No. 86-144, dated May 31, 1986, which documented the violation and declared the event as a Potential Licensee Event Report (LER) No. 86-018-00.

A special safety inspection by the senior resident inspector was performed to assess the licensee's activities associated with the blocking of the AFP ESFAS. The inspection included:

(1) The review of documentation; Final Safety Analysis Report (FSAR), Safety Evaluation Report (SER), Incident Report (IR) No. 86-144, and plant operating log.

(2) Interviews with Operations Department personnel.

(3) NRC/Licensee meetings.

b. Inspection Findings

Event Chronology

May 30, 1986 3:30 P.M. CST	The "B" main feedwater pump (MFP) tripped resulting in an Auxiliary Feedwater Actuation. Both "A" and "B" AFPs (motor driven) started. Steam generator levels were restored in accordance with off-normal procedure OTO-AE-00001, "Feedwater System Malfunction".
May 30, 1986 5:43 P.M. CST	The AFP ESFAS blocking switches FC-HA-25 and FC-HS-26 were placed in the block position in preparation for MFP startup. Placing the switches in "block" was not a procedural action for the existing plant condition (Mode 2 with AFPs maintaining steam generator levels). Blocking of the switches was done to avoid an AFAS-M actuation during the specified startup trip test of the MFPs. The blocking action was not logged.
May 30, 1986 7:51 P.M. CST	The "B" MFP was placed in service. Both "A" and "B" AFPs were stopped. The blocking switches were left in "block".
May 31, 1986 6:48 A.M. CST	The off going RO, during a control board walkdown prior to shift turnover, observed the mispositioned blocking switches, notified the SS, and placed the switch in "permit" terminating the violation.

Through document review and personnel interviews, the inspector determined that the violation resulted from a combination of procedure deficiencies and personnel performance errors.

(1) Procedure Deficiencies

Plant off-normal procedure OTO-AE-00001, "Feedwater System Malfunction", provides brief instructions for loss of main feedwater pump(s). The procedure specifies starting AFPs and restoring steam generator levels to normal. The procedure does not provide recovery instructions nor reference recovery/restart procedures.

General Operating Procedure OTG-ZZ-00003, "Plant Startup Below 5% to 20% Power", Step 4.2.3 provides instructions to start a MFP in accordance with OTN-AE-00001, "Feedwater System", Step 4.2.4 specifies stopping the motor driven startup feedwater pump after the MFP is in service. The procedure provides no instructions regarding the AFP ESFAS blocking switches.

Normal Operating Procedure OTN-AE-00001, "Feedwater System", Section 4.3 (MFP Startup) provides instruction for startup, testing (trip), and placing the MFP in service. This section provides no instructions regarding the AFP ESFAS blocking switches. Subsequent sections do provide blocking switch instructions and is the normal method for plant startup:

> OTN-AE-00001 - Section 4.5 (Maintaining steam generator (SG) levels with the startup (SU) feedwater pump)
>
> "Caution - Ensure FC-HS-25 and FC-HS-26 (AFP ESFAS Blocking Switches) are in the block position (this prevents an inadvertent AFAS due to both MFP turbines being in a tripped condition".
>
> OTN-AE-00001 - Section 4.6 (Shifting from the startup main feedwater pump to the main feedwater pump)
>
> Step 4.6.1 - Ensure Section 4.3 MFP SU is complete.
>
> Step 4.6.2 - Place FC-HS-25 and FC-HS-26 AFP ESFAS Blocking Switches in the "permit" position.

The inspector determined that plant procedures applicable for transferring from the auxiliary feedwater pumps to the main feedwater pump failed to provide procedural control of the AFP ESFAS blocking switches. In addition, the existing plant startup procedure (OTN-AE-00001) which permits the blocking of the AFP ESFAS switches, is not compatible with T/S in that blocking switches FC-HS-25 and FC-HS-26 renders all four AFP-M channels inoperable. T/S 3.3.2 requires that the Engineered Safety Features Actuation System (ESFAS) instrumentation channels shown in Table 3.3-3 be Operable. Table 3.3-3 Function Unit (auxiliary feedwater) 6.g. (Trip of All Main Feedwater Pumps - Start Motor Driven Pumps (Auxiliary Feedwater Pumps (AFP)) requires a minimum of three channels operable in Modes 1 and 2.

(2) Personnel Performance Errors

(a) Placing the AFP ESFAS Blocking Switches in "block" was not a procedural action, thus negating procedural control of these switches.

(b) Blocking the AFP ESFAS was not logged; failure to log placing the switches in "Block" defeated an opportunity of detection during shift turnover log review.

(c) Control board walkdowns during the shift change at 11:00 P.M. on May 30, 1986, (specified in Operations Department Procedure ODP-ZZ-00003, "Shift Relief and Turnover"), failed to identify/recognize the T/S violation.

Indication provided on the control board:

Panel RL027/28

Two AFP ESFAS BLOC switch positions

Train A FC-HS-25 "BLOCK"
Train B FC-HS-26 "BLOCK"

ESF SYS STATUS Panel

Six red lights - (three each train)

Aux Actuate
FW SYS
Loss of MFP TURB BLOCK

c. Safety Reports

SNUPPS Final Safety Analysis Report (FSAR), Chapter 7 provides AFS description. SNUPPS FSAR, Chapter 15 provides AFS safety analysis for loss of normal feedwater flow and feedwater system pipe break. Applicable FSAR sections are attached. No credit is taken in the analysis for automatic start of the AFPs on loss of main feed pumps.

In the Callaway Safety Evaluation Report NUREG-0830, dated October 1981, the staff considered the initiation of the AFS on loss of the MFPs as an "anticipatory signal" for which no credit was taken in the analysis. However, the staff specified that the design included appropriate features to ensure that the block is removed to provide AFS initiation on loss of main feedwater.

d. Safety Significance

Plant operations, while the AFP ESFAS blocking switches were in the "block" position, resulted in the Auxiliary Feedwater System (AFS) being in a degraded condition such that automatic start of the motor driven auxiliary feedwater pumps on loss of the MFPs was defeated. However, the AFS was operable in all other aspects and would have responded to safety injection, loss-of-off-site-power, Lo-Lo steam generator levels or manual initiation signals.

While the lack of the automatic start feature of the AFPs on MFP trip posed no significant threat to public/plant safety, the event is considered significant based on procedural and personnel performance deficiencies, which placed the plant in a condition prohibited by Callaway Technical Specifications.

Callaway plant records show that the plant operated in Modes 1 and 2 from 7:51 pm CST on May 30, 1986 until 6:48 am CST on May 31, 1986, a period of approximately 11 hours, with all four channels of AFP ESFAS "blocked" inoperable. The licensee's failure to maintain the AFP ESFAS operable while in Modes 1 and 2 is a violation of Technical Specifications Limiting Condition for Operation 3.3.2. No. 483/86017-01(DRP)

3. NRC Region III Management Site Visit

On June 10, 1986, the resident inspectors accompanied NRC Region III Management (Messrs. A. Bert Davis and R. F. Warnick) on a plant tour. The tour included an inplant walkdown and interviews with licensee's operating staff. General observation are as follows:

Plant Conditions indicate that good housekeeping practices are being implemented.

Control room personnel were attentive and knowledgeable of plant conditions and control room activities were being conducted in a business-like manner.

Operating crew personnel displayed a positive attitude toward plant operations in general, and a personal determination toward event reduction.

NRC observations were discussed with the licensee in an exit meeting following the visit. During this meeting, the licensee discussed progress made in reducing the number of reactor trips and Licensee Event Reports and licensee's event reduction programs in place. The discussion focused on personnel errors in general and specifically on the blocking of the AFP ESFAS. This discussion included the licensee's safety assessment, root cause, andcorrective actions. Corrective action includes the following:

TAKEN	PLANNED
Use of the switch in Modes 1 and 2 has been prohibited by Night Orders until a T/S interpretation is issued.	A proposed T/S change will be initiated.
Operations personnel have been instructed to take prompt action to bring procedure anomalies to the attention of supervisor.	Applicable procedures will be revised to address the various pump shifts.
	Operations personnel will be briefed on the liabilities of taking action outside of

Letters to all Operations personnel concerning attention to control board indication have been issued.	procedures. The procedure change will include a note to identify the potential for an AFAS. The operators will be instructed about "unplanned actuations".

The inspection has shown that the licensee identified and documented the violation. The violation once identified, was given prompt and thorough management attention. Licensee's corrective actions appear adequate to prevent recurrence. The effectiveness of these actions will be assessed in subsequent NRC inspections.

4. Exit Interview

The inspector met with licensee representatives (denoted under Persons Contacted) at intervals during the inspection period. The inspector summarized the scope and findings of the inspection. The licensee representatives acknowledged the findings as reported herein. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.

5. Enforcement Conference

An Enforcement Conference was held on June 3, 1986 at the NRC Region III office, Glen Ellyn, Illinois between Mr. D. F. Schnell and members of the Region III staff. During the meeting the Licensee presented facts relative to an event on April 12, 1986, discussed in Inspection Report No. 50-483/86010. Messrs. A. Bert Davis and Robert F. Warnick of the NRC Region III staff conducted a site visit and tour on June 2, 1986. During the visit Mr. Steve Miltenberger and other members of the Licensee staff presented facts relative to the May 31, 1986 event, discussed in paragraph 2 above. The Licensee presented background information, corrective action to prevent recurrence, and potential mitigating facts which the NRC will use to determine the appropriate enforcement action. On June 23, 1986, Mr. William L. Forney of the Region III staff called Mr. D. F. Schnell to discuss the need to conduct an additional Enforcement Conference. Mr. Schnell declined his option to attend an Enforcement Conference stating that all of the pertinent facts had been presented to Messrs. Davis and Warnick during their site visit.

Attachment: SNUPPS FSAR Section 7.3.6.1.1.a

control panel. The AFS flow to each steam generator is indicated on both the main control board and at the auxiliary shutdown control panel.

The AFS equipment is described in Section 10.4.9.

In addition to initiating functions described above, the auxiliary feedwater actuation signal (AFAS) closes the steam generator blowdown and sample isolation valves, when auxiliary feedwater is required by plant conditions. All remote manually operated valves in the normal suction from the CST and in the discharge to the steam generators are normally open.

7.3.6.1.1 System Description

a. Initiating circuits

The motor-driven pumps are started on the occurrence of any one of the following signals:

1. Manual start

2. Safeguards sequence signal (initiated by safety injection signal or loss-of-offsite-power)

3. Auxiliary feedwater actuation (AFAS-M)

AFAS-M is generated on the occurrence of any one of the following events:

1. Trip of both main feedwater pumps (Manual block of the main feed pump trip signals is provided at the main control board, and is indicated on the ESFAS status panel. This block permits startup and shutdown of the plant without automatic start of the AFPs, while allowing the AFPs to remain available to respond to a demand from any other source.)

2. 2 out of 4 low-low level signals in any one steam generator

3. Manual AFAS-M initiation

The turbine-driven pump is started on the occurrence of either of the following signals:

1. Manual start

2. Auxiliary feedwater actuation (AFAS-T)