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October 3, 1988
LIC-88-666

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

SUBJECT: Special Report on the Untimely Submittal of Licensee Event Reports

Gentlemen:

The Omaha Public Power District (OPPD), holder of Operating License DPR-40, submits this report to document the untimely submittal of Licensee Event Reports.

In November 1987, the process for handling Operations Incident (OI) Reports was revised and computerized to facilitate a more timely review of the new Incident Reports (IR), to assign responsibility for tracking IR's and ensure timely review. In February 1988, OPPD organized a group to close out a backlog of 343 open OI's which had been generated prior to November 1987. These OI's had been reviewed for reportability at the time of the occurrence, but had not been closed out. The group was tasked to review the circumstances of each incident, review again for reportability and perform a safety analysis per 10 CFR 50.59.

During the review process nineteen incidents were discovered to have been potentially reportable under 10 CFR 50.72 and/or 10 CFR 50.73 in effect at the time of occurrence. The LER process is designed for reporting current events not historical events; therefore, this special report is being submitted to report these events. Further investigation revealed that 15 of these 19 items were reportable under the reporting criteria in effect at the time of the event.

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The attachment includes the 19 items provided to the NRC Senior Resident Inspector. Each item is detailed as follows:

a. Event Description - Brief description of the event based on the original OI and other available references.

b. Immediate Actions Taken - Immediate actions taken as a result of the event based on the original OI and other references, i.e., Control Room Log.

c. Corrective Actions - Actions proposed or implemented based on information contained in available documents. Not all corrective actions stated could be verified due to the time span since the event occurred. In those instances, corrective actions which have been taken as a result of more recent similar events are included to address the measures taken to prevent future occurrences.

d. Nuclear Safety Significance - An assessment of the safety consequences and implications of the event is given.

e. Basis for Reportability - The applicable section of the Technical Specification, 10 CFR 50.72 and/or 10 CFR 50.73, is stated.

f. Reason for Not Reporting - The reason most of the 15 reportable items were not reported can be attributed to an inadequate OI review and tracking process. Other reasons are included if they could be determined.

Several actions have been taken to improve the review and reporting process of Incident Reports. These are broken down into two areas.

1. The IR Tracking Process

   The IR process has been computerized since November 1987. The 19 events occurred prior to this time. The IR's are initiated on computer data base and go through an extensive review process. Initially, the originator and/or the Shift Technical Advisor (STA) write the IR, and the STA consults with the Shift Supervisor on the reportability of the event. The IR is then reviewed by the Incident Evaluation Coordinator for reportability, and he assigns an action addressee. The reports are then presented to the Plant Review Committee within the week. Nuclear Licensing & Industry Affairs also reviews IR's periodically and is notified upon the determination of a reportable item. Also upon determining an item is reportable, the STA assesses the safety significance of the event if the event places the plant in an abnormal situation or for any plant parameter that affects or reflects an abnormal indication of a safety-related system. This process should preclude an inaccurate determination of reportability being made.

2. The Plant Review Committee (PRC) Training Program

A PRC training plan will be developed by the end of October 1988. Training on safety responsibilities of PRC members will begin immediately following the development of the training plan. PRC member training will begin in January 1989 and will be completed by July 1989. This training will include the determination of reportability and should preclude future errors in this area.

If you have any questions, do not hesitate to contact us.

Sincerely,

W Gary Gates

for K. J. Morris
Division Manager
Nuclear Operations

KJM/mc

Attachment

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R. D. Martin, NRC Regional Administrator
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P. H. Harrell, NRC Senior Resident Inspector

INADVERTENT ACTUATION OF 480V LOAD SHED
OI 001319

a. Event Description

On May 26, 1981, at 0921, at Fort Calhoun Station, channel "A" 480V load shed was initiated. This occurred during the performance of Surveillance Test ST-ESF-2 F.1, "Channel 'A' Safety Injection Actuation Signal Test," which verifies the proper operation of the initiation circuitry and all equipment normally operated by channel "A" safety feature actuation signals. This test requires the operator to turn the 480V Load Shed Switch CS-A/LS on Panel AI-30A to OFF prior to initiating a test signal. However, the operator did not block the Load Shed prior to initiating the test signal and the Load Shed was activated. Appropriate nonessential motors, pumps, etc. on channel "A" were properly shed.

b. Immediate Actions Taken

The loads that were shed during this event had their power supply returned to normal once the test signal was removed and the 480V Load Shed Switch CS-A/LS was returned to EMERGENCY STANDBY (its normal position).

c. Corrective Actions

This event occurred because approved procedures were not followed. The specific corrective action for this event could not be determined because the individual and his immediate supervisor are no longer employed by OPPD.

d. Nuclear Safety Significance

This event had no effect on nuclear safety. The loads shed were non-engineered safeguards and are not required for the safe shutdown of the plant.

e. Basis for Reportability

10 CFR 50.72(a)(7) (year 1981) required licensees to report, "Any event resulting in manual or automatic actuation of Engineered Safety Features, including the Reactor Protection System." Since the 480V Load Shed is an Engineered Safety Feature and was not actuated as part of a preplanned sequence, this event was reportable.

f. Reason for Not Reporting

An error was made determining the reportability during the initial review of the OI.

INOPERABLE SNUBBERS ON SHUTDOWN COOLING SYSTEM
OI 001384

a. Event Description

At approximately 1100 on September 29, 1981, during refueling shutdown conditions, it was observed that six Technical Specification Snubbers were inoperable on the shutdown cooling system. This system is required to be operable during Modes 4 and 5. The snubbers had been taken down for redesign and/or replacement.

b. Immediate Actions Taken

Generating Station Engineering (GSE) immediately began returning all of the inoperable snubbers to service. The last of the six snubbers in question was returned to service at 2000 on September 29, 1981.

This was completed within nine hours from the time of discovery.

c. Corrective Actions

No corrective actions were taken other than the immediate actions taken.

d. Nuclear Safety Significance

The shutdown cooling system is required to be operable during Modes 4 and 5.

e. Basis for Reportability

NOT REPORTABLE

There was no Technical Specification requirement for snubber operability while in the refueling mode of the time of the event. No Technical Specification was violated. Reference: Fort Calhoun Station Technical Specification Amendment 48 Section 2.18(1).

f. Reason for Not Reporting

The incident was not reportable.

VHRA DOOR LEFT OPEN
OI 001571

a. Event Description

On August 6, 1982 at 1415 a door to a very high radiation area was left open. The door, shield door in RM-11, was left open after the pressure equipment group transferred filters to drums behind the door. The door was logged open at 1415 on August 6, 1982. The same alarm was also verified active on August 7 and August 8 and had not been cleared. The wrong door was visually checked shut by both operations and security personnel. Since it appeared that the door was secure and the personnel checking the door were unaware that there were two very high radiation area doors associated with Room 11 on the circuit, a maintenance order was issued to fix the circuit. At 0900 on August 9, 1982 the MO was reviewed by a Health Physics representative. He had health physics personnel check to see that both doors were shut. At 0930 it was determined the inside door was still open. This could not be determined from outside the step off pad, due to the distance (about 21 feet) and the lack of illumination.

b. Immediate Actions Taken

The door was shut, and the alarm cleared.

c. Corrective Actions

Since the occurrence of this incident the following corrective actions have been taken:

1. Red painted padlocks are used to secure doors which allow entrance into a VHRA. The keys are controlled by the on duty shift supervisor, auxiliary building operators, and plant health physicist.

2. A sign has been posted on the VHRA doors stating, "HP Technician Required", "Two Persons Required for Entry", and "FC-647 Form Must Be Completed Upon Exit".

3. The Radiation Protection Manual has been revised to require the shift health physics technician to perform a documented physical check to ensure the doors are properly locked every four hours.

4. The Radiation Protection Manual now requires that after entry a qualified health physics technician will check the very high radiation area door and ensure that it is closed and locked.

5. The Manual also requires that one of the individuals accompanying the health physics technician into the VHRA will also check that the VHRA door is closed, latched, and locked.

d. Nuclear Safety Significance

These doors are kept locked and surveillance is maintained to prevent people from accidentally entering the very high radiation area.

e. Basis for Reportability

This was a violation of Technical Specification 5.11.2. This event was reportable pursuant to 10 CFR 50.72(a)(4) (year 1982).

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

BOTH DIESEL GENERATORS PLACED IN OFF AUTO CONDITION
OI 001686

a. Event Description

On March 29, 1983 at 0830, at Fort Calhoun Station, Emergency Diesel Generators D-1 and D-2 were placed in an off auto condition to prevent their starting during trip checks on the 345 kV System. Diesel Generators are designed to provide station power in the event of a loss of the 161 kV (offsite) and 345 kV Systems. The Reactor Coolant System (RCS) temperature was at 180°F at 0830 but was increased to 380°F during the course of the day. Fort Calhoun Station Technical Specification 2.7(1)1 requires, in part, that the reactor shall not be heated up or maintained at a temperature above 300°F unless both diesel generators are operable. Since the reactor was heated above 300°F with the Diesel Generators inoperable, Technical Specification 2.7(1)1 was violated.

b. Immediate Actions Taken

Both Diesel Generators were switched from off auto to auto standby at 1558, March 29, 1983, and trip check testing of the 345 kV System was suspended upon the discovery of Technical Specification 2.7(1)1 violation.

c. Corrective Actions

Since the occurrence of this event, there has been increased emphasis on procedure compliance. Operator training has been improved with respect to Technical Specification requirements.

d. Nuclear Safety Significance

Nuclear safety was not impacted due to the fact that the safeguard equipment powered by the diesel generators is not required by the Technical Specifications until the reactor is critical.

e. Basis for Reportability

Violation of Technical Specification Limiting Condition for Operations 2.7(1)1. This event was reportable pursuant to 10 CFR 50.72(a)(5) (year 1983).

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

MISSED SURVEILLANCE TEST
OI 001847

a. Event Description

On March 24, 1984 and March 25, 1984, while in the refueling mode, the documentation for ST-SDM-1 was not completed. This surveillance test is used to ensure that the reactor will be maintained subcritical to preclude accidental criticality in the shutdown condition. Upon investigation it was determined that the data for March 24 and March 25, 1984 was taken. However, the worksheets which are attached to ST-SDM-1 were not completed until March 26, 1984. This event resulted from a failure to follow approved procedures and resulted in a violation of Technical Specifications 3.10. Technical Specification 3.10 requires that the shutdown margin be determined daily.

b. Immediate Action Taken

Verified Boron concentration on March 24, 1984 and March 25, 1984 was adequate from primary chemistry reports.

c. Corrective Actions

A new surveillance test tracking procedure using manual scheduling and computer tracking has been in place on a trial basis since February 1988. This procedure will be incorporated into Standing Order G-23 following approval of Procedure Change 22548.

d. Nuclear Safety Significance

The Boron Dilution Event during refueling is analyzed in Section 14.3.2.5 of the USAR. The shutdown margin did not fall below 1900 PPM boron during this event. This is sufficient boron to keep the reactor subcritical. Nuclear safety was not affected by this event.

e. Basis for Reportability

Technical Specification 3.10 was violated. 10 CFR 50.73(a)(2)(i)(B) (year 1984) stated, "Any operation or condition prohibited by the plant's Technical Specifications."

f. Reason for Not Reporting

It was assumed that the event was not reportable because the tests had actually been performed, the data was satisfactory, and the event concerned only timely documentation.

MISSED SURVEILLANCE TEST
OI 001869

a. Event Description

The monthly surveillance test that verified that the spare battery charger operates correctly when connected to either bus, was not completed within its allowed time. ST-DC-2 was scheduled on April 12, 1984 and was not completed until April 20, 1984.

Technical Specification 3.1 states that "A maximum allowable extension not to exceed 25% of the Surveillance Interval unless otherwise specified". Technical Specification 3.7.(2).c states that at monthly intervals the third battery charger shall be paralleled in turn to each bus. The surveillance test was not completed within the time allowed.

b. Immediate Actions Taken

ST-DC-2 was performed on April 20, 1984.

c. Corrective Actions

A new surveillance test tracking procedure using manual scheduling and computer tracking has been in place on a trial basis since February of 1988. This procedure will be incorporated into Standing Order G-23 following approval of Procedure Change No. 22548.

d. Nuclear Safety Significance

The surveillance testing specified verifies that the spare battery charger operates correctly when connected to either bus. The specified testing interval is considered adequate, given the system redundancy to detect and correct any malfunction before it adversely affects the station battery system.

e. Basis for Reportability

This event was a violation of Technical Specification 3.1. This event was reportable pursuant to 10 CFR 50.73(a)(2)(i)(B) (year 1984) which stated, "Any operation or condition prohibited by the plant's Technical Specifications."

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

LOSS OF OFFSITE POWER DUE TO WEATHER CONDITION
OI 001885

a. Event Description

On April 29, 1984 Fort Calhoun was in refueling shutdown and station power was being supplied by the offsite 161 kV lines. The 345 kV lines were tagged out. Off-site power was lost due to lightning. The diesel generators and associated breakers actuated to supply house power.

b. Immediate Actions Taken

Restarted necessary equipment to maintain plant conditions in their normal refueling shutdown condition. The NRC was notified by phone at 2110, April 29, 1984.

c. Corrective Actions

No further corrective actions were taken since the system functioned as designed.

d. Nuclear Safety Significance

The response of the emergency electrical supply system was as described in Section 8.1.2 of the USAR.

e. Basis for Reportability

10 CFR 50.72 (b)(2)(ii) and 10 CFR 50.73 (a)(2)(iv) (year 1984), require licensees to report, "any event or condition that results in the manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protective System (RPS)."

f. Reason for Not Reporting

There was a failure to follow up and the LER was not sent.

SURVEILLANCE TEST PERFORMED LATE
OI 001906

a. Event Description

Surveillance tests ST-RM-1 F.2 and ST-RM-2 F.2, monthly tests of area and process radiation monitors, scheduled for May 2, 1984, were not completed until May 11, 1984. Technical Specification 3.1 requires these test to be performed monthly with a maximum allowable extension of 25% of the surveillance interval. This allowable extension was exceeded for these tests and Technical Specification 3.1 was violated.

This event occurred because the technicians performed annual surveillance tests before doing the monthly surveillance tests.

b. Immediate Actions Taken

The tests were completed on May 11, 1984, and no irregularities were identified.

c. Corrective Actions

A new surveillance test tracking procedure using manual scheduling and computer tracking has been in place on a trial basis since February 1988. This procedure will be incorporated into Standing Order G-23 following approval of Procedure Change 22548.

d. Nuclear Safety Significance

Surveillance tests ST-RM-1 F.2 and ST-RM-2 F.2 ensure the operability of the process and area radiation monitors important to nuclear safety.

e. Basis for Reportability

This event was a violation of Technical Specification 3.1, Table 3.3. This event was reportable pursuant to 10 CFR 50.73(a)(2)(B) (year 1984) which stated, "Any operation or condition prohibited by the plant's Technical Specifications."

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

COOLING AIR SUPPLY VALVES FAIL CLOSED
OI 001981

a. Event Description

Air Inlet Dampers YCV-871G and YCV-871H for Diesel Generator Room Number 1 are required to open upon the loss of instrument air or loss of power. However, it was discovered on August 14, 1984 that upon the loss of air, both dampers fail closed. Diesel Generator No. 2 is not involved.

b. Immediate Actions Taken

Both dampers were blocked open.

c. Corrective Action

The field work for design modification FC-84-151 was completed on November 27, 1984. This modification changed the failure mode of the valves to fail open.

d. Nuclear Safety Significance

Paragraph 8.4.1.1 of the USAR states: "The emergency diesel generators are designed to furnish reliable in-plant a-c power adequate for safe plant shutdown and standby and for operation of engineered safeguards, when no energy is available for the 345 and 161 kV systems. For adequate reliability two units are provided."

e. Basis for Reportability

When YCV-871G and YCV-871H fail close, the design basis of 8.4.1.1 of the USAR is violated. This event was reportable pursuant to 10 CFR 50.72(b)(1)(ii)(B) and 50.73(a)(2)(ii)(B) (year 1984).

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

MISSED SURVEILLANCE TEST
OI 002007

a. Event Description

Surveillance test ST-FP-9, quarterly inspection of temporary fire barriers, due April 10, 1984 was not completed. The next scheduled test was performed on July 10, 1984, and no irregularities were identified. Technical Specification 3.15(5)a, which requires that this visual inspection be performed at least once per 18 months, was not violated.

b. Immediate Actions Taken

At the time of the missed surveillance test, the plant was in an extended forced outage with the reactor in refueling shutdown condition. The temporary barriers were verified functional before the reactor was made critical.

c. Corrective Actions

Surveillance Test ST-FP-9 F.1 was revised September 24, 1988 to make it easier to follow.

d. Nuclear Safety Significance

Fire detection and fighting systems are provided to minimize the adverse effects of fires on structures, systems, and components important to safety. Surveillance testing insures the operability of these systems.

e. Basis for Reportability

NOT REPORTABLE.

f. Reason for Not Reporting

Technical Specification 3.15(5)a was not violated.

HIGH COLD LEG TEMPERATURE
OI 002009

a. Event Description

On September 25, 1984, Purification Ion Exchanger CH-8A was being placed into service. While flushing the exchanger, the reactor power and temperature began increasing. The cold leg temperature rose above 545°F for about an hour. Reactor power increased to 1515 MWt.

Technical Specification 2.10.4(5)(a)(i) requires, in part, that the cold leg temperature be less than or equal to 545°F. If the parameter exceeds the limit, it must be restored within 2 hours or reactor power must be reduced to less than 15% of rated power within the next 8 hours. This event terminated before the 2 hour limit was exceeded.

b. Immediate Actions Taken

Group 4 control rods were inserted and Boron was injected into the core to reduce system power.

c. Corrective Actions

Operating Instruction OI-CH-4 was revised just prior to this event. The date of the revision is September 11, 1988. This revision requires flushing of the ion exchanger for five minutes to prevent a dilution event. Since this event, the operator training program has greatly improved and should reduce the chance of operator error.

d. Nuclear Safety Significance

The plant was operated within the provisions of Technical Specification 2.10.4(5)(a)(i).

e. Basis for Reportability

NOT REPORTABLE

f. Reason for Not Reporting

This event did not result in a violation of Technical Specifications. The plant was operated within the Limiting Conditions of Operation.

SURVEILLANCE TEST ST-RLT-3 F.1 MISSED
OI 002126

a. Event Description

On July 13, 1984, at Fort Calhoun Station, Surveillance Test ST-RLT-3 F.1, "Reactor Coolant System Leak Rate Test", was not completed. The Reactor Coolant System (RCS) leak rate can be calculated by either one of two methods. One method uses the Plant Computer, P-250, to calculate the RCS leak rate while the other method, used when the computer is unavailable, is a hand calculation of the RCS leak rate. ST-RLT-3 F.1 is normally completed during each night shift and is the responsibility of Operations. On this date, the Plant Computer was unavailable and the night shift operators did not perform the hand calculation of the RCS leak rate. This information was not included on the Shift Turnover Log and the subsequent shifts did not perform ST-RLT-3 F.1. ST-RLT-3 F.1 has a daily frequency as required by Fort Calhoun Station Technical Specification 3.2, Table 3-5, Item 8.

The reactor was in the start-up mode and the reactor coolant system was being diluted. The surveillance test gives erroneous results under these conditions.

b. Immediate Actions Taken

Surveillance Test, ST-RLT-3 F.1 was completed on July 12, 1984 using the Plant Computer and on July 14, 1984 using the hand calculation method.

c. Corrective Actions

A new surveillance test tracking procedure using manual scheduling and computer tracking has been in place on a trial basis since February of 1988. This procedure will be incorporated into Standing Order G-23 following approval of Procedure Change No. 22548.

d. Nuclear Safety Significance

Even though this event is a violation of the Fort Calhoun Station Technical Specifications, it does not affect nuclear safety. The plant is equipped with indicators in the control room of reactor coolant leakage. The devices providing this indication are Containment Air Particulate Monitor RM-050, Containment Gas Monitor RM-051, Dew Point Monitor, and Containment Sump Pump Operation. These devices will provide adequate warning to the operators in the case of excessive reactor coolant leakage.

e. Basis for Reportability

This event was a violation of Technical Specification 3.2, Table 3-5, Item 8. This event was reportable pursuant to 10 CFR 50.73(a)(2)(i)(B) (year 1984) which required the licensee to report, "Any operation or condition prohibited by the plant's Technical Specifications."

f. <u>Reason for Not Reporting</u>

An error was made in determining the reportability during the initial review of the OI.

NON-CERTIFIED FILTERS USED IN VA-64
OI 002210

a. Event Description

On September 10, 1985, Surveillance Test ST-IR-1 F.4 for Control Room Filter (VA-64) Replacement was performed. Instead of attaching certification of the filters to the Surveillance Test, the QA material release tags affixed by the storeroom to the filters were attached to the test. A test engineer tried to find the certifications for the three filter trays. Two of the three filter trays (serial numbers 1885 and 1888) did not have certifications. The plant continued to operate with these filters in place until September 29, 1985.

b. Immediate Actions Taken

The non-certified filters were replaced with certified filters.

c. Corrective Actions

Personnel awareness of the importance of verbatim compliance with procedures is being promoted by an increased emphasis on training.

d. Nuclear Safety Significance

The Fort Calhoun Station Technical Specifications define "operable," and "operability" as follows:

> "A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s)."

Section 7, Paragraph 4.1.3 of the Quality Assurance Plan states, "Until suitable documentary evidence is available to show the equipment or material is in conformance, affected equipment shall be considered to be inoperable, unless operability has been determined to be adequate, based on a documented review."

Technical Specification 2.12(4) requires the control room air treatment system to be operable; and, if it is not, seven (7) days are allowed to make the inoperable circuit operable. If these conditions are not met, the reactor shall be placed in cold shutdown within 24 hours.

e. Basis for Reportability

Technical Specification Violation. This is a reportable event pursuant to 10 CFR 50.72(b)(ii)(B) and 10 CFR 50.73(a)(2)(ii)(B) (year 1985).

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

TWO CHARGING PUMPS INOPERABLE
OI 002455

a. Event Description

On June 4, 1986, while Fort Calhoun was operating at 100 percent power, charging pump CH-1B was tagged out at 0825 so that MO-862052 could be done. In the process of tagging out charging pump CH-1B, discharge header cross connect valve, CH-191, was closed instead of the charging pump CH-1B discharge valve, CH-192. This resulted in the loss of the normal flow path from charging pump CH-1C. The Fort Calhoun Station Technical Specifications define "operable" and "operability" as follows:

> "A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s)."

Thus, there was only one operable charging pump, CH-1A, when Valve CH-191 was closed.

b. Immediate Actions Taken

This error was discovered upon completion of MO-862052 on June 5, 1986 at 1130, and CH-191 was immediately opened to restore the flow path from CH-1C. Charging pump CH-1B was returned to service at 1235. The other valve positions associated with the three charging pumps were checked at this time and it was determined that CH-191 was the only valve that had been incorrectly set.

c. Corrective Actions

The specific corrective action for this event could not be determined. However, since this event, there has been increased emphasis on procedure compliance in the operator training programs.

d. Nuclear Safety Significance

Technical Specification 2.2(2)a states, "At least two charging pumps shall be operable." Technical Specification 2.2(3)a. states, "One of the operable charging pumps may be removed from service provided two charging pumps are operable within 24 hours."

The basis for the Limiting Conditions for Operation, Technical Specification 2.2, states:

> "The limits on component operability and the time periods for inoperability were selected on the basis of the redundancy indicated above and engineering judgment."

e. Basis for Reportability

Technical Specification violation. The Danger/Caution Tag Sheet for June 4, 1986 shows Valve CH-192 (CH-191) was tagged out at 0825 on June 4, 1986 and returned to service on 1235 on June 5, 1986. Valve CH-191 was closed for 28 hours and 10 minutes. Technical Specification Limiting Condition for Operation 2.2(3)a was violated. This is a reportable event pursuant to 10 CFR 50.72(b)(ii)(B) and 10 CFR 50.73(2)(ii)(B) (year 1986).

f. Reason for Not Reporting

Poor documentation resulted in an error in determining the reportability during the initial review of the OI.

TEMPORARY PENETRATION FIRE BARRIER DISCOVERED MISSING
OI 002481

a. Event Description

On July 9, 1986, during a quarterly fire barrier inspection, it was discovered that temporary penetration fire barrier 19-E-56 was completely missing. This barrier seals an opening around a crane rail that penetrates a steel panel door. The door separates the turbine building from an equipment hatch access in the auxiliary building. Technical Specification 2.19(7) requires that the nonfunctional penetration be restored to operable status within 7 days, or prepare and submit a report to the NRC, pursuant to Section 5.9.3 of the Technical Specifications, within an additional 30 days. The barrier was not replaced and no special report was submitted.

b. Immediate Actions Taken

Upon discovery, an hourly firewatch patrol was initiated.

c. Corrective Actions

None were required as discussed in Item d.

d. Nuclear Safety Significance

Fire detection and protection systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety.

The current Fort Calhoun Station Fire Hazards Analysis describes this door and opening as unrated. The door is protected by a water curtain, and is considered acceptable in its current configuration without a penetration barrier. A subsequent engineering evaluation concluded that a penetration barrier was not required at this location (reference memo FC-738-88 dated April 21, 1988).

e. Basis for Reportability

NOT REPORTABLE.

f. Reason for Not Reporting

NOT REPORTABLE.

WASTE GAS DISPOSAL
OI 002485

a. Event Description

On July 4, 1986, the gas in the vent header of the Waste Gas Disposal System was pumped to Waste Gas Decay Tank C. Prior to pumping, the gas was analyzed for hydrogen and oxygen concentrations. The Waste Gas Sampling System, AI-110, was available for use during this operation, but was not put into service. If the Waste Gas Sampling System is not put into service, Technical Specification 2.9.1(2)d requires that a grab sample be taken from the waste gas decay tank and analyzed within 24 hours. This analysis was not performed within 24 hours.

b. Immediate Actions Taken

A grab sample was taken from the waste gas decay tank and analyzed on July 9, 1986. The analysis showed that the flammability limit of hydrogen and oxygen was not exceeded.

c. Corrective Actions

Procedure Change No. 17891, initiated August 11, 1986, requires the recording of hydrogen and oxygen concentrations in the Operators log.

d. Nuclear Safety Significance

The basis of Technical Specification 2.9.1(2)d, "ensures that the concentration of potentially explosive gas mixtures entrained in the gas decay tank(s) will be maintained below the flammability limits of hydrogen and oxygen."

The worst case event, a gas decay tank rupture, was analyzed in Section 14.19 of the USAR which states, in part, "that a rupture of a gas decay tank would not interrupt or restrict public use of areas beyond the exclusion area boundary."

e. Basis for Reportability

This was a violation of Technical Specification 2.9.1(2)d. This event was reportable pursuant to 10 CFR 50.72(b)(ii)(B) and 10 CFR 50.73(a)(2)(ii)(B) (year 1986) that require reporting when a limiting condition of operation is not met.

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

INOPERABLE TOXIC GAS MONITORS
OI 002580

a. Event Description

On September 16, 1986, the Control Room Ventilation System was placed in the recirculation mode C in accordance with Technical Specification 2.22. Toxic Gas Monitors 6286 A/B and 6288 A/B were taken out of service so that ST-TGM-1 F.1 could be performed. Upon receipt of VIAS, due to other surveillance tests being performed, the Control Room was switched from recirculation Mode C to filtered makeup Mode B approximately 6 to 12 times from September 16 to October 9, 1986. During this time, the toxic gas monitors were still inoperable. The longest a VIAS signal was ever in was approximately 15 minutes.

b. Immediate Actions Taken

By the time the determination was made that the Technical Specification had been violated, the system had been returned to its proper configuration; therefore, no immediate corrective actions were necessary.

c. Corrective Actions

Since this event there has been an increased emphasis on operator training which will reduce the chance of this type of event happening again.

d. Nuclear Safety Significance

The basis of Technical Specification 2.22 states, "If both of the Toxic Gas Monitors are found inoperable, there is no immediate threat to the control room operators and reactor operation may continue while repairs are being made. During this repair, the control room ventilation will be switcned to internal recirculation mode of operation."

e. Basis for Reportability

This event constituted a violation of the limiting condition of operation set forth in Technical Specification 2.22. This event was reportable pursuant to 10 CFR 50.72(b)(ii)(B) and 10 CFR 50.73(a)(2)(ii)(B) (year 1986).

f. Reason for Not Reporting

An error was made in determining the reportability during the initial review of the OI.

INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES
OI 002613

a. Event Description

On December 1, 1986, at 0835, at Fort Calhoun Station, Raw Water Pump AC-10B, Low Pressure Safety Injection (LPSI) Pump SI-1B, and the Turbine Driven Auxiliary Feedwater Pump FW-10 were inadvertently started. This occurred during the performance of Surveillance Test ST-ESF-5 F.1, "Automatic Load Sequencer Check," which verifies the proper operation and timer settings of the Automatic Load Sequencers. This test requires the operator to inhibit operation of the equipment connected to the Sequencer by turning all Sequencer Isolation Switches on the particular Sequencer to OFF. Then the operator must turn the Sequencer Auto Start Test Switch to TEST which trips the Sequencer Lockout Relays and starts the sequencer timers. Next, the operator should return the Sequencer Auto Test Switch to NORM, reset the Sequencer Lockout Relays, and return the Sequencer Isolation Switches to ON. While testing Automatic Load Sequencer S2-2, the operator turned the Sequencer Isolation Switches to ON for AC-10B, SI-1B, and FW-10 prior to resetting Sequencer Lockout Relays 86-1/S2-2 and 86-2/S2-2. This resulted in the actuation of the above listed pumps.

b. Immediate Actions Taken

A second operator immediately observed control room indication of the pumps' actuation and after confirming that this event was an inadvertent actuation by the first operator, turned pumps AC-10B, SI-1B, and FW-10 off at their respective control switches.

c. Corrective Actions

The surveillance test ST-ESF-5 F.1 was revised the following day, December 2, 1986, Procedure Change No. 18727. ST-ESF-5 F.1 only had one set of procedure steps for both sequencers with a signoff for each sequencer. This was confusing. The procedure change separated the steps for sequencers S1-2 and S2-2.

d. Nuclear Safety Significance

This event does not affect nuclear safety for the following reasons. The Low Pressure Safety Injection pumps and piping are aligned for recirculation and will not inject into the Reactor Coolant System until a Safety Injection Actuation Signal (SIAS) is received. The Auxiliary Feedwater pumps and piping are aligned for recirculation and will not provide flow to the steam generators until an Auxiliary Feedwater Actuation Signal (AFAS) is received. The Raw Water pumps and piping are aligned to permit additional flow from consecutive pumps being started.

e. Basis for Reportability

The actuation was not part of a preplanned action and was reportable pursuant to 10 CFR 50.72(b)(2)(ii) and 10 CFR 50.73(a)(2)(iv) (year 1986).

f. <u>Reason for Not Reporting</u>

An error was made in determining the reportability during the initial review of the OI.

INADVERTENT ACTUATION OF ENGINEERED SAFETY FEATURES
OI 002660

a. Event Description

On February 2, 1987, at 0915, at Fort Calhoun Station, Raw Water Pumps AC-10A and AC-10C, Low Pressure Safety Injection (LPSI) Pump SI-1A, High Pressure Safety Injection (HPSI) Pumps SI-2A and SI-2C, and Auxiliary Feedwater Pump FW-6 were inadvertently started. This occurred during the performance of Surveillance Test ST-ESF-5 F.1, "Automatic Load Sequencer Check," which verifies the proper operation and timer settings of the Automatic Load Sequencers. This test requires the operator to inhibit operation of the equipment connected to the Sequencer by turning all Sequencer Isolation Switches on the particular Sequencer to OFF. Then the operator must turn the Sequencer Auto Start Test Switch to TEST which trips the Sequencer Lockout Relays and starts the sequencer timers. Next, the operator should return the Sequencer Auto Test Switch to NORM, reset the Sequencer Lockout Relays, and return the Sequencer Isolation Switches to ON. While testing Automatic Load Sequencer S1-2, the operator turned the Sequencer Isolation Switches to ON for AC-10A, AC-10C, SI-1A, SI-2A, SI-2C and FW-6 prior to resetting Sequencer Lockout Relays 86-1/S1-2 and 86-2/S1-2. This resulted in the actuation of the above listed pumps.

b. Immediate Actions Taken

A second operator immediately observed control room indication of the pump's actuation and after confirming that this event was an inadvertent actuation by the first operator, turned pumps AC-10A, AC-10C, SI-1A, SI-2A, SI-2C and FW-6 off at their respective control switches.

c. Corrective Actions

The surveillance test now requires the operator to verify that the OFF AUTO lights are on.

d. Nuclear Safety Significance

This event does not affect nuclear safety for the following reasons. The Low Pressure Safety Injection and High Pressure Safety Injection pumps and piping are aligned for recirculation and will not inject into the Reactor Coolant System until a Safety Injection Actuation Signal (SIAS) is received. The Auxiliary Feedwater pumps and piping are aligned for recirculation and will not provide flow to the steam generators until an Auxiliary Feedwater Actuation Signal (AFAS) is received. The Raw Water pumps and piping are aligned to permit additional flow from consecutive pumps are started.

e. Basis for Reportability

This actuation was not part of a preplanned actuation and was reportable under 10 CFR 50.72(b)(2)(ii) and 10 CFR 50.73(a)(2)(iv) (year 1987).

f. <u>Reason for Not Reporting</u>

An error was made in determining the reportability during the initial review of the OI.