Letter to N. C. Moseley from Duke Power Company dated October 29, 1974.

Letter to N. C. Moseley from Duke Power Company dated October 23, 1974.

Letter to N. C. Moseley from Duke Power Company dated October 18, 1974.

Letter to N. C. Moseley from Duke Power Company dated October 11, 1974.

Letter to N. C. Moseley from Duke Power Company dated September 24, 1974.

Letter to N. C. Moseley from Duke Power Company dated September 23, 1974.

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AD.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

October 29, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 2
Docket No. 50-270

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Abnormal Occurrence Report AO-270/74-15.

Very truly yours,

A. C. Thies

A. C. Thies

ACT:vr
Attachment

cc: Mr. Angelo Giambusso

DUKE POWER COMPANY
OCONEE UNIT 2

Report No.: AO-270/74-15

Report Date: October 29, 1974

Occurrence Date: October 10, 1974

Facility: Oconee Unit 2, Seneca, South Carolina

Identification of Occurrence: Missing guide pin from valve 2LP-14

Conditions Prior to Occurrence: Unit 2 in cold shutdown

Description of Occurrence:

On July 1, 1974, an inspection of valve 2LP-12 on Oconee Unit 2 revealed a missing guide pin. This incident was described in report UE-270/74-3. On October 10, 1974, while the unit was shut down for maintenance, inspection of the identical valve 2LP-14 in the redundant train of the Low Pressure Injection System revealed that its guide pin was also missing. The guide pin is made of 304 stainless steel with dimensions 3/4" x 4".

Designation of Apparent Cause of Occurrence:

The apparent cause for the failure of the guide pin in valve 2LP-14 is cavitation damage when operating in the decay heat removal mode. Consultation with the Crane Company, manufacturer of the valve, and Babcock & Wilcox revealed that operation at flow rates less than 3000 gpm could cause cavitation to occur.

The integrity of the guide pin was verified in January 1974. Since then, the system has operated at flow rates less than 3000 gpm in the decay heat removal mode for considerable periods of time. Although it is possible for the pin to have broken off anytime since January, it was not recovered during reactor refueling in March 1974, nor was it detected using loose parts monitoring equipment during the loose parts investigation in July, 1974.

Analysis of Occurrence:

Valve 2LP-14 is located on the outlet of Decay Heat Cooler B. There are two check valves, 2LP-47 and 2CF-12, one flow orifice, and one Engineered Safeguards (ES) valve, 2LP-18 between 2LP-14 and the reactor vessel.

The function of the Low Pressure Injection System upon an Engineered Safeguards (ES) signal is to supply borated water to the reactor core. This requires valves 2LP-47, 2CF-12, and 2LP-18 to open to provide one of two redundant paths for low pressure injection flow. Valve 2LP-14 does not

receive an ES signal but is open during reactor operation. The dimensions of the missing part (3/4" x 4") are such that it could only prevent full opening or closing of the eight-inch valves 2LP-47, 2CF-12, or 2LP-18 should it interfere. This would only reduce the flow of water through one redundant path to the core. However, these three valves have been satisfactorily tested.

Low points of the piping between 2LP-14 and the reactor vessel have been radiographed without revealing the pin location. Flow orifice 2FT-4A has been removed and inspected for signs of the pin colliding with the orifice; none were noted.

In a previous instance of operating with a loose part in the reactor vessel in January, 1974, detection with loose parts monitoring equipment was possible during startup and coastdown of reactor coolant pumps. On October 20, 1974, all four reactor coolant pumps were run individually for short periods of time while monitoring for loose parts was conducted. A faint metallic noise was apparent during all single pump runs and was more obvious during pump starts and stops. It is concluded that this noise is caused by this guide pin since it was not detected in July, 1974.

The guide pin apparently entered the reactor vessel through the core flood header carried by the decay heat removal coolant. With four pumps operating, the part becomes lodged against the bottom of the core. The worst possible safety-related situation for operating with this part in the lower reactor vessel head is considered to be the lodging of the object within a fuel assembly. It would then be assumed that local fuel clad failure occurs due to either departure from nucleate boiling or mechanical wear. As a result of this, reactor coolant activity would increase; hence, activity would be a satisfactory parameter for judging the status of fuel clad integrity during operation. Only localized fuel damage is postulated; therefore, the core would remain in a coolable geometry. Coolant activity will be monitored daily for 30 days and the plant promptly shut down for further investigation if fuel damage is indicated.

Since this part becomes lodged during normal operation, wear to the reactor vessel internals is not probable. There is also no indication that the valve guide pin is interfering with the operation of one of the redundant flow paths of the Low Pressure Injection System. Consequently, the continued operation of Oconee 2 does not represent undue risk to the health and safety of the public.

Corrective Action:

A monitoring and surveillance program is established to ensure continued safe operation of Oconee Unit 2. To ensure early detection of the part, the Loose Parts Monitoring System provides an alarm whenever a preset limit is exceeded. The reactor coolant is being monitored for gross activity and isotopic content daily to detect fuel clad failure. This surveillance program will be continued until sufficient data have been accumulated to justify its termination and for a minimum of 30 days.

Procedures are being revised to ensure flow rates of at least 3000 gpm in 2LP-14 and similar valves to eliminate the possibility of cavitation. However, several maintenance procedures such as operation with the water level below the reactor vessel nozzles will not permit flow in excess of 3000 gpm without the low pressure injection pumps cavitating. A study is underway to determine the best means for eliminating the cavitation problems associated with the flow rates through the valves.

Failure Data:

2LP-14 - 10 inch, 300 lb., cast alloy steel globe valve, Crane Company Catalog No. 151½LU.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

October 23, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 2
Docket No. 50-270

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Unusual Event Report UE-270/74-6.

Very truly yours,

A. C. Thies By Hal B. Tucker

ACT:vr
Attachment

cc: Mr. Angelo Giambusso

DUKE POWER COMPANY
OCONEE UNIT 2

Report No.: UE-270/74-6

Report Date: October 23, 1974

Event Date: September 26, 1974

Facility: Oconee Unit 2, Seneca, South Carolina

Identification of Event: Incorrect torque setting on Engineered Safeguard valve operator

Conditions Prior to Event: Unit at 70 percent of full power

Description of Event:

On September 26, 1974, the Reactor Building isolation valve exercise functional test (operational) was performed at Oconee Unit 2. The test is performed using manual control pushbuttons located on the Engineered Safeguards Panel (RZ module) which simulate an Engineered Safeguards signal. In the process of the test, valve 2LPSW-5, low pressure service water to decay heat removal cooler B, did not give indications of proper performance. The test was performed again with visual observation of valve 2LPSW-5. The valve started to move when the signal was applied; however, it stopped with the valve just barely open. This valve is intended to open fully upon an Engineered Safeguards signal.

Designation of Apparent Cause of Event:

The torque limiting switch for the valve motor operator for 2LPSW-5 was discovered to be at its lowest setting. The valve motor was therefore shut off by the torque limiting switch when a light drag was experienced by the valve operating mechanism.

Analysis of Event:

Valve 2LPSW-5 allows low pressure service water to pass through decay heat removal cooler B. This cooler is one of two redundant coolers designed to remove decay heat from water recirculated from the emergency sump, after the borated water storage tank had been emptied, following a loss of coolant accident. Recirculation would not be required within 30 minutes following initiation of Engineered Safeguards System actuation. Due to system redundancy and the relatively long time before low pressure service water would be necessary following an accident, it is concluded that the health and safety of the public was not affected.

Corrective Action:

The torque switch on valve 2LPSW-5 was adjusted and satisfactory performance was demonstrated. An investigation is being conducted to determine a more consistent method of setting all torque controlled motor operators.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

October 18, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 2
Docket No. 50-270

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Abnormal Occurrence Report AO-270/74-14.

Very truly yours,

A. C. Thies

ACT:vr
Attachment

cc: Mr. Angelo Giambusso

DUKE POWER COMPANY
OCONEE UNIT 2

Report No.: AO-270/74-14

Report Date: October 18, 1974

Occurrence Date: October 4, 1974

Facility: Oconee Unit 2, Seneca, South Carolina

Identification of Occurrence: Failure to provide cooling water while in the decay heat removal mode

Conditions Prior to Occurrence: Reactor shutdown and cooled down in the decay heat removal mode

Description of Occurrence:

Oconee Unit 2 was shut down and in the decay heat removal mode of operation on October 4, 1974. Low pressure injection pump (LPI) 2A was operating with both decay heat removal coolers to remove reactor decay heat. Valves 2LPSW 251 and 252 were used to throttle low pressure service water (LPSW) to the coolers. A low differential pressure alarm was noted on LPI pump 2A and the suction temperature was 210°F. Further investigation revealed that low pressure service water flow to the decay heat coolers was zero. The controllers for these two valves were noted to be set for 15 percent open to throttle LPSW flow. The Control Operator opened 2LPSW251 and 252 further to restore low pressure service water to the decay heat removal coolers.

Designation of Apparent Cause of Occurrence:

The lack of low pressure service water flow to the decay heat removal coolers was due to a peculiarity of the valve operating mechanism. Tests performed with valve 2LPSW-251 indicate that flow through the valve begins when the controller is in the 15 percent open position. The valves and control mechanisms were verified to be operating correctly.

Analysis of Occurrence:

Low pressure service water valves 2LPSW-251 and 2LPSW-252 are used for throttling of low pressure service water through the decay heat removal coolers only during the decay heat removal mode. When the Low Pressure Injection System is lined up to support Engineered Safeguards (ES) operation, both valves are fully open. Failure of the air supply to the valves also results in the valves being fully open. This incident in no way affected Engineered Safeguards operation.

The reactor coolant temperature was allowed to slowly increase to 210°F because the decay heat removal coolers were not being supplied with low pressure service water. The rate of temperature increase due to reactor decay heat is slow enough to allow correction of any malfunction and the restoration of decay heat removal. It is concluded that the health and safety of the public was not affected.

Corrective Action:

Low pressure injection pump 2A has been tested to ensure that the pump head curve was not altered due to pump cavitation. The effectiveness of the purification demineralizer will be inspected during power escalation testing to ensure that the efficiency of the resin has not been reduced.

In order to give the Control Room operator better indication of flow through the decay heat removal coolers, flow indicators will be mounted in the control room next to the controllers for valves 2LPSW-251 and 2LPSW-252.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

October 11, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 2
Docket No. 50-270

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Abnormal Occurrence Report AO-270/74-13.

Very truly yours,

A. C. Thies

A. C. Thies

ACT:vr
Attachment

cc: Mr. Angelo Giambusso

DUKE POWER COMPANY
OCONEE UNIT 2

Report No.: AO-270/74-13

Report Date: October 11, 1974

Occurrence Date: September 28, 1974

Facility: Oconee Nuclear Station, Unit 2

Identification of Occurrence: Non-verification of redundant component operability

Conditions Prior to Occurrence: Unit 2 at steady-state power; testing of several Low Pressure Injection System components completed

Description of Occurrence:

At approximately 0705 on September 28, 1974, an attempt was made to open Engineered Safeguards valve 2LP-22 in order to vent Low Pressure Injection Pump 2B; however, the valve did not respond due to actuation of a thermal overload coil in the valve motor circuit. The thermal overload was reset and the valve was cycled twice to verify operability.

At approximately 1000, the Shift Supervisor initiated a work request to check the thermal overload coils in valve 2LP-22 motor circuit. At approximately 1100, he opened the circuit breaker for 2LP-22 valve motor for inspection without verifying operability of valve 2LP-21 (redundant component). Valve 2LP-21 was tested later and found to be operable.

Designation of Apparent Cause of Occurrence:

Inspection of the motor controller for valve 2LP-22 revealed an undersized thermal overload coil in the valve motor circuit and an oversized coil in the location for the thermal overload coil which provides a status signal to the computer. This condition was corrected; however, due to an oversight by Operations and Maintenance personnel, the redundant component (Valve 2LP-21) was not verified operable prior to removing valve 2LP-22 temporarily (approximately five minutes) from service.

Analysis of Occurrence:

Assuming that valve 2LP-21 was not operable during the brief period of time that 2LP-22 was out of service, the Low Pressure Injection pumps could still be utilized for long-term core cooling, if needed, using the Reactor Building Emergency Sump as a source of water. Similar consequences could also result from the misplaced thermal overload coils in 2LP-22. With valve 2LP-21 operable and 2LP-22 out of service, the Engineered Safeguards System is still completely functional. The health and safety of the public was not endangered as a result of this occurrence.

Corrective Action:

The necessity for redundant component checks will be reviewed with Operations supervisory personnel. Engineered Safeguards valves having this thermal overload coil arrangement will be inspected to ensure that the coils are in their proper locations.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. BOX 2178

September 24, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 2
Docket No. 50-270

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Abnormal Occurrence Report AO-270/74-11.

Very truly yours,

A. C. Thies

A. C. Thies

ACT:gje
Attachment

cc: Mr. Angelo Giambusso

DUKE POWER COMPANY
OCONEE UNIT 2

Report No.: AO-270-74-11

Report Date: September 24, 1974

Occurrence Date: September 10, 1974

Facility: Oconee Unit 2

Identification of Occurrence: Power level cutoff exceeded during transient xenon conditions

Conditions Prior to Occurrence: Unit at approximately 80 percent full power

Description of Occurrence:

On September 10, 1974, Oconee Unit 2 was operating at approximately 80 percent full power while waiting for xenon to approach equilibrium. The following sequence of events took place:

1547 Turbine bypass valves 2MS-19 and 2MS-22 opened

1550 Reactor power increased to 81.4 percent full power. An Instrument and Control Engineer investigated but could not identify any problem.

1553 2MS-19 and 2MS-22 closed

1554 2MS-19 and 2MS-22 opened

1602 Reactor power increased to 82.9 percent full power

1607 Reactor power decreased below 82.5 percent full power

1635 Control operator put A loop turbine bypass hand/auto station in manual and slowly closed the bypass valves; then returned the hand/auto station to auto. Instrument and control personnel checked the bypass valve controls and found nothing abnormal.

1749 2MS-19 and 2MS-22 opened

1751 Control operator put hand/auto station in manual and closed the bypass valves

1754 Reactor power had increased to 84.1 percent full power

1807 Control operator closed the bypass valves and decreased reactor power to 78.5 percent full power. He then verified valve controller was calling for the valves to be closed and placed the hand/auto station in auto. Both A loop bypass valves opened.

1809 Reactor power increased to 81.9 percent full power. Control operator placed A loop turbine bypass hand/auto station in manual and began closing the bypass valves.

1816 Bypass valves closed and hand/auto station left in manual.

1817 2MS-19 and 2MS-22 opened

1824 Reactor power increased to 88.4 percent full power. The shift supervisor closed the bypass valves from the auxiliary shutdown panel.

1832 With reactor power still slightly above the power cutoff level of 82.5 percent full power, a control rod drive programmer motor fault caused the ICS to go into track and the Diamond control station to go into manual. Reactor power, megawatts electric and feedwater began increasing and unit average temperature began decreasing.

1833 The control operator reset the motor fault and placed the Diamond control station in auto with a slight positive neutron error. The control rods began withdrawing and the control operator returned the Diamond control station to manual.

1834 The control operator placed the Bailey reactor master station in manual and the Diamond station in auto and began to withdraw the control rods to return the unit average temperature to normal.

1844 Reactor power had increased to 88.4 percent full power and the control operator began an immediate reduction of power.

1854 Reactor power below 82.5 percent full power.

Refer to Figure 1 for a schematic arrangement of the ICS stations involved.

Designation of Apparent Cause of Occurrence:

Oconee Technical Specification 3.5.2.5.d does not allow reactor power to be increased above 82.5 percent full power unless xenon reactivity is within 10 percent of the value for operation at steady-state rated power. Reactor power was increased above 82.5 percent full power each time by the Integrated Control System in response to the opening of the A loop turbine bypass valves. As the bypass valves opened, dumping steam directly to the condenser, the ICS increased reactor demand in order to hold unit load. The bypass valves opened improperly because of two failed modules within the turbine bypass valve controls. While reactor power was still above 82.5 percent full power, the control operator, in error, increased reactor power further above the power level cutoff in order to return the unit average temperature to normal.

Analysis of Occurrence:

This occurrence increased reactor power above the power level cutoff. The power level cutoff is designed to maintain power peaking within the limits of the AEC criteria for a Loss of Coolant Accident. Although xenon reactivity was not within prescribed limits, the other parameters, which affect power peaking, power tilt and imbalance, were within the normal operating limits.

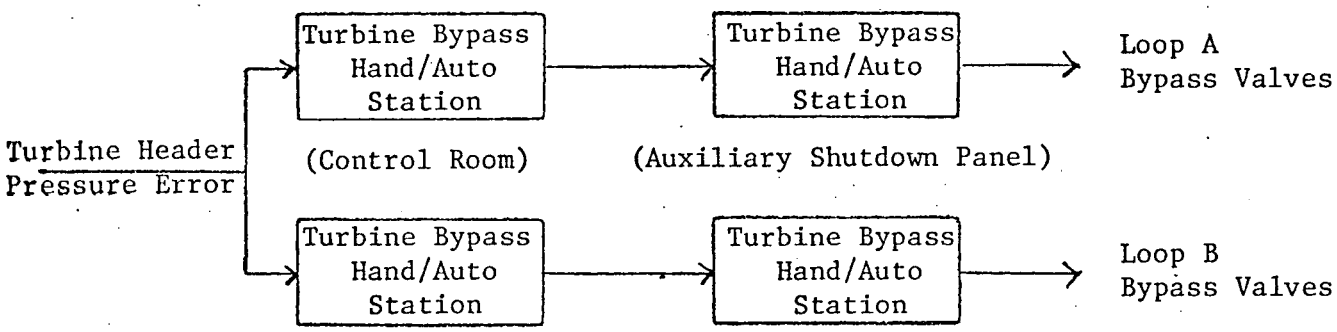
Operation above the power level cutoff was limited to a short period of time. It is concluded the occurrence did not affect the safe operation of the unit nor the health and safety of the public.

Corrective Action:

The two failed modules in the turbine bypass valve control section of the ICS were replaced and a functional check performed to verify their correct operation.

The occurrence was reviewed with all operations personnel to identify corrective actions such as immediate power reductions that should be taken to avoid improperly exceeding the power level cutoff.

TURBINE BYPASS VALVE CONTROL



REACTOR CONTROL

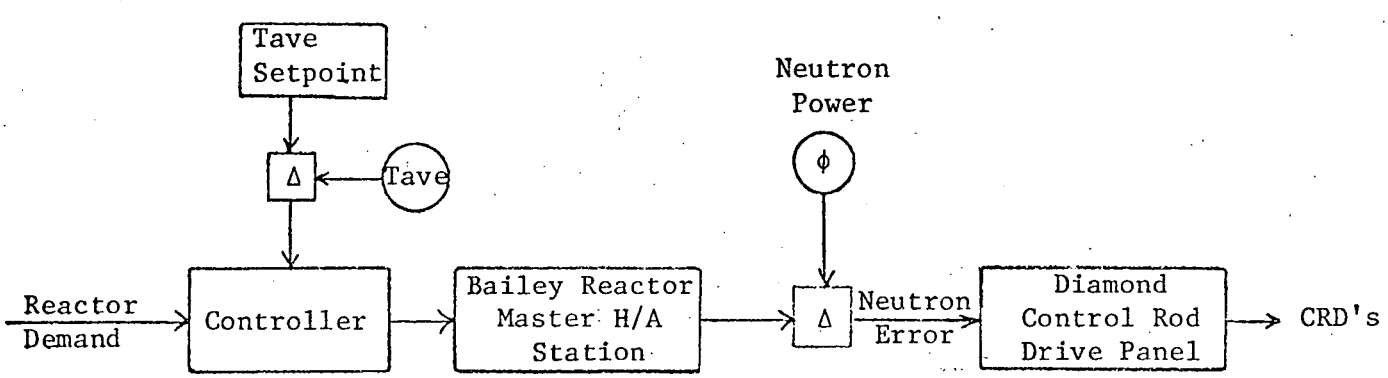


Figure 1

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

September 23, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 2
Docket No. 50-270

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Unusual Event Report UE-270/74-4.

Very truly yours,

A. C. Thies

ACT:gje
Attachment

cc: Mr. Angelo Giambusso

DUKE POWER COMPANY
OCONEE UNIT 2

Report No.: UE-270/74-4

Report Date: September 23, 1974

Event Date: August 9, 1974

Facility: Oconee Unit 2, Seneca, South Carolina

Identification of Event: Lee Steam Station Combustion Turbine Trip During Test

Conditions Prior to Event: Unit 2 Shut Down, Reactor Coolant System Partially Drained

Description of Event:

On August 9, 1974, a test of the 100 kV power supply from an isolated Lee Steam Station combustion turbine was conducted. The Unit 2 loads on the 4160 volt Engineered Safeguard Switchgear Buses were transferred to the 100 kV transmission line at approximately 1330. When Lee Unit No. 6C was isolated from the system grid, the combustion turbine tripped and breakers SL1 and SL2, located between transformer CT5 and Standby Buses 1 and 2, tripped on undervoltage of the 100 kV transmission line. The Oconee Unit 2 loads were immediately transferred to Startup Transformer CT2.

At 1530, Lee Unit No. 5C was placed in service and isolated from the Lee grid and the Central substation. Engineered Safeguards Switchgear Bus 2TE was transferred to the isolated Lee unit, but it tripped as the load was picked up. Breakers SL1 and SL2 tripped on 100 kV transmission line undervoltage. The Switchgear Bus 2TE was immediately re-energized by transferring it to Startup Transformer CT2.

A transfer of Oconee Unit 2 auxiliary loads to the Central substation through Transformer CT-5, at 1600, indicated there were no grounds in the equipment. No further testing was performed on August 9, 1974.

Designation of Apparent Cause of Event:

No problems were encountered with energizing the standby buses from a Lee unit. In both instances, the combustion turbine tripped as load was being placed upon the units. Extensive subsequent testing did not reveal the cause of the turbine trips nor could the turbine trips be reproduced using the same procedures.

Analysis of the Event:

The primary source of power for the Engineered Safeguards Switchgear buses is the 230 kV switchyard through either the startup transformer or the

auxiliary transformer. Emergency power is supplied from the Keowee Hydro unit via the underground feeder. The overhead transmission line from Keowee is another backup for emergency power. The 100 kV transmission line provides a standby backup source of emergency power from the Central switchyard or it can be supplied from an isolated Lee Steam Station combustion turbine. The use of the Lee combustion turbines normally involves a 20 minute delay to bring the turbine up to speed and to isolate it from the grid before power can be supplied to an Oconee unit. Therefore, use of the Lee combustion turbine is a planned evolution to provide a standby backup source of power.

Since all of these sources of power were available during this test of the Lee combustion turbines, and Unit 2 had been in a cold shutdown condition for one week prior to this incident, and the loads were re-energized immediately, it is concluded that the health and safety of the public was not affected by the tripping of the Lee combustion turbines.

Corrective Action:

Extensive testing of equipment at the Lee Steam Station and Oconee Nuclear Station has not revealed the cause of the turbine trips. The Oconee Unit 2 auxiliary loads were successfully placed on isolated Lee Unit 5C and 6C on August 14, 1974. No equipment or procedural changes had been made since the Lee units tripped on August 9, 1974. Oconee Unit 2 loads were again placed on an isolated Lee unit on August 16 and 17, 1974 with no problems encountered.

The periodic test procedure for the 100 kV power supply has been revised to provide for actually carrying Oconee unit loads from the Lee gas turbines. This will simulate actual emergency conditions during the test and check the performance of the Lee units periodically.