

DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

April 6, 1982

TELEPHONE: AREA 704
373-4083

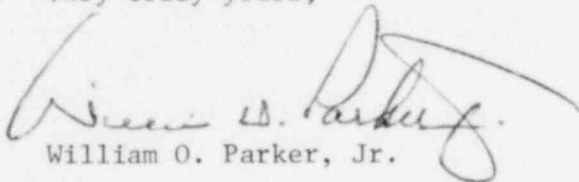
Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Re: Oconee Nuclear Station
Docket No. 50-269

Dear Mr. O'Reilly:

Please find attached Reportable Occurrence Report RO-269/82-08. This report is submitted pursuant to Oconee Nuclear Station Technical Specification 6.6.2.1.a(2) which concerns an operation subject to a limiting condition for operation which was less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications, and describes an incident which is considered to be of no significance with respect to its effect on the health and safety of the public.

Very truly yours,


William O. Parker, Jr.

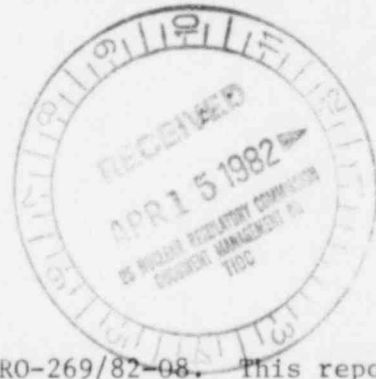
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Attachment

cc: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. W. T. Orders
NRC Resident Inspector
Oconee Nuclear Station

Records Center
Institute of Nuclear Power Operations
1820 Water Place
Atlanta, Georgia 30339

Mr. Philip C. Wagner
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555



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DUKE POWER COMPANY
OCONEE NUCLEAR STATION UNIT 1

Report Number: RO-269/82-08

Report Date: April 6, 1982

Occurrence Date: March 23, 1982

Facility: Oconee Unit 1, Seneca, South Carolina

Identification of Occurrence: Violation of containment integrity

Conditions Prior to Occurrence: Hot shutdown

Description of Occurrence: On March 23, 1982, air was found flowing into the Reactor Building through an instrument line test tee in penetration WB-13. The cap for the test tee was missing, thereby allowing flow from the penetration room into the Reactor Building. This constitutes a violation of Technical Specification 3.6.1.

Apparent Cause of Occurrence: The apparent cause of this occurrence is personnel error, in that the Reactor Building Pressure Switch was not properly returned to service upon completion of calibration testing during the unit refueling outage in July 1981.

Analysis of Occurrence: The test tee with the cap removed creates a 0.19 inch ID leak path in the Reactor Building. At the Reactor Building design basis pressure (59 psig) the resultant leak rate via this path is not significantly greater than the Reactor Building design leak rate of 0.25 weight percent/day. Leakage from the Reactor Building under these conditions would be filtered by the Penetration Room Ventilation System and radiation levels would be monitored on RIA-51 gaseous radiation monitor.

There is no evidence indicating that radiation levels in the penetration room during unit operation in February and March 1982 were above normal levels experienced during previous unit operation. This incident would have caused pressure switch 1PS22 (Channel 7 for the Reactor Building Spray System) to function improperly. This pressure switch is one of three inputs in a two-out-of-three logic for Channel 7 of Reactor Building Spray. Therefore, an Engineered Safeguards signal would have been transmitted if Reactor Building pressure had increased to the setpoint. Also, Channel 8 for Reactor Building Spray System is a redundant train to Channel 7. Since the two trains of the Reactor Building Spray System are equally sized, in the event of an additional single failure one train would provide the capability to remove heat from the containment. Additionally, as discussed in the Oconee FSAR, Section 6, the Reactor Building Cooling System, acting independently from the Reactor Building Spray System, is capable of limiting the containment pressure below the design pressure in the event of a loss-of-coolant accident. Thus, it is considered that this incident had no significant effect on the health and safety of the public.

Corrective Action: The immediate corrective action identified the source of the air leak as a missing cap for the test tee for 1PS22. The test tee was capped, thereby reestablishing containment integrity. An investigation was performed on the instrumentation associated with the reactor protective system and engineering safety feature systems to assure that all instruments were properly in service. The investigation revealed no additional abnormalities.

In January 1982 the station Instrumentation and Electrical Section had commenced the process of instituting a specific identification of each item that may have been removed from its normal state and an independent verification that the item had been returned to its normal state. Prior to January 1982, a redundant verification had been utilized by the crew members performing a calibration. Prior to this incident, procedural steps had identified that an instrument was "returned to normal" but did not identify specific valves, caps, etc. for which verification was required. The Instrumentation and Electrical procedures dealing with safety-related components will be modified to require an independent signature verification that each affected component has been returned to its normal state. This will include an independent verification (different from the person or persons performing the work) that the equipment is properly returned to service if it cannot be confirmed by diverse means (control room indication, string checks, etc.). This will require independent verification of all instrumentation connected to system piping when the system is not in service and in some cases when the system is in service. If equipment is determined to be functional by diverse means it shall be so documented by the individual or individuals performing the work. It is intended that the third person check be as independent a verification as possible. These verification requirements have been added to all Instrumentation and Electrical procedures.

The changes in verification requirements will be presented to all affected personnel with emphasis on the necessity for ensuring proper equipment status for safe operation of the plant.