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FILE NUMBER INCIDENT REPORT

TO: Mr. Norman C, Moseley

FROM: Duke Power Co.
Charlotte, N. C. 28242
William O. Parker, Jr.

DATE OF DOCUMENT 06-06-77

DATE RECEIVED 06-24-77

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PLANT NAME: OCONNE UNIT # 1
jcm 06-24-77

ENCLOSURE Consists of Reportable Occurrence Report (RO-269/77-15) on 06-06-77 concerning One channel of borated water storage tank level instrumentation inoperable... 2 pages

Reportable Occurrence Report (RO-269/77-18) on 05-27-77 concerning potential prbolem in the ECCS analysis which could impact all B&W operating reactors.... 2 pages

NOTE: IF PERSONNEL EXPOSURE IS INVOLVED SEND DIRECTLY TO KREGER/J. COLLINS

FOR ACTION/INFORMATION

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NRC FORM 195 (2-76)

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

June 6, 1977

Mr. Norman C. Moseley, Director
U. S. Nuclear Regulatory Commission
Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Reference: Oconee Unit 1
Docket No. 50-269

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Reportable Occurrence Report RO-269/77-15.

Very truly yours,

W. O. Parker, Jr.

William O. Parker, Jr. By

LJB:ge

Attachment

cc: Director, Office of Management Information
and Program Control

DUKE POWER COMPANY
OCONEE UNIT 1

Report No.: RO-269/77-15

Report Date: June 6, 1977

Occurrence Date: May 6, 1977

Facility: Oconee Unit 1, Seneca, South Carolina

Identification of Occurrence: One channel of borated water storage tank level instrumentation inoperable

Conditions Prior to Occurrence: Unit at 100 percent full power

Description of Occurrence:

On May 6, 1977, it was determined that Channel 1 of the Oconee Unit 1 borated water storage tank (BWST) level instrumentation was indicating a lower level than Channel 2 of the instrumentation. Channel 2 indicated that the level of borated water required by technical specifications was being maintained. A work request was promptly issued, the line was cleared, and in less than one hour the Channel 1 level indicator was declared operable.

Apparent Cause of Occurrence:

The cause of this occurrence has not been determined. It is postulated, however, that the Dekoron trace heating controller was failing intermittently resulting in boiling of the borated water in the process line and bubbles of steam.

Analysis of Occurrence:

This condition is considered to be reportable since it constituted operation in a degraded mode permitted by a Limiting Condition for Operation. Technical Specification 3.3.5 makes provision for the removal from service for test or monitoring of any component of the high pressure injection, low pressure injection or Reactor Building spray for a period of 24 hours provided not more than one train of each system is affected. This occurrence resulted in the loss of one of two redundant channels of BWST level indication for less than one hour. During this period, the redundant level transmitter properly indicated the true level of the BWST. The conditions of the BWST required by Technical Specification 3.2 were maintained and the emergency core cooling systems would have performed as required in the unlikely event they were needed.

It is concluded that the health and safety of the public were not affected by this incident.

Corrective Action:

The BWST level transmitter process line was bled and returned to service within one hour of the incident. The Dekoron heat tracing controller was turned off pending installation of a modification which will replace the present Dekoron heat tracing system, which surrounds the process line, with a Nelson heat tracing system as committed in our letter of January 31, 1977, which transmitted Reportable Occurrence Report RO-270/77-1. This modification will be implemented by October 3, 1977. The modification should eliminate problems experienced with the Dekoron heat tracing system.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

June 10, 1977

TELEPHONE: AREA 704
373-4083

Mr. Norman C. Moseley, Director
U. S. Nuclear Regulatory Commission
Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Re: Oconee Unit 1
Docket No. 50-269

Dear Mr. Moseley:

Pursuant to Sections 6.2 and 6.6.2 of the Oconee Nuclear Station Technical Specifications, please find attached Reportable Occurrence Report RO-269/77-18.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

MST:ge
Attachment

cc: Director, Office of Management Information
and Program Control

DUKE POWER COMPANY
OCONEE UNIT 1

Report No.: RO-269/77-18

Report Date: June 10, 1977

Occurrence Date: May 27, 1977

Facility: Oconee Unit 1, Seneca, South Carolina

Identification of Occurrence: Input error discovered in ECCS analysis

Description of Occurrence:

On May 24, 1977, Duke Power Company was notified by NRC/ONRR of a potential problem in the Emergency Core Cooling System (ECCS) analysis which could impact all Babcock & Wilcox operating reactors. The problem, which was first identified during the review of Davis-Besse, Unit 1, is related to the U-baffles on the core barrel at the reactor coolant inlet nozzles. These U-baffles had been modified from the configuration originally assumed in the ECCS analysis, such that, the U-baffle fluid flow characteristics (pressure loss coefficients) used in the ECCS blowdown analysis are not consistent with the actual system configuration. When the appropriate U-baffle pressure loss coefficient was assumed in the case of Davis-Besse 1, the calculated peak cladding temperature (PCT) increased significantly. Certain specific confirmatory analyses were requested to assure continued acceptable results of the Oconee ECCS analysis.

In discussions with B&W on May 27, 1977, a determination was made that this should be a reportable occurrence pursuant to Oconee Technical Specification 6.6.2.1.a(8).

Designation of Apparent Cause of Occurrence:

The apparent cause of this concern is related to the U-baffles on the core barrel at the reactor coolant inlet nozzle. This U-baffle design is different from the configuration assumed in the LOCA analysis presented in BAW-10103, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS." This design difference affects the flow characteristics, and thus, the calculated pressure losses attributed to these baffles during a postulated LOCA.

Analysis of Occurrence:

This occurrence reveals an error in the flow resistance factor associated with the U-baffle in the ECCS analysis. The correction of only this flow resistance factor would be inappropriate in light of the availability of improved design pressure distribution models for the entire Reactor Coolant System based upon actual operating experience. This data, unavailable when the 177 fuel assembly lowered-loop analysis was submitted to the NRC, can now be used to alter and verify system pressure distributions. These improved distributions are the basis upon which to construct future LOCA evaluations.

Based upon the correction of a similar error in other 177 fuel assembly raised-loop NSS, the Babcock and Wilcox Company expects that upgrading these pressure distributions to the new data will result in an insignificant increase or a decrease in calculated peak cladding temperatures for Oconee Nuclear Station.

In consideration of the expected small effect of this error and the margins which exist in the area of ECCS, it is concluded that continued operation while confirmatory analyses are being conducted, will not affect the public health and safety.

Corrective Action:

Confirmatory analyses requested by NRC/ONRR are currently in progress.

U.S.A.E.C.
REGULATORY OPERATIONS
REGION II
ATLANTA, GA.

JUN 13 10 30 AM '77