JUN 22 1984

MEMORANDUM FOR: D. Eisenhut, Director Division of Licensing, NRR

FROM: R. Denise, Director, Division of Reactor Safety and Projects

SUBJECT: REGION IV SER INPUT ON THE FORT CALHOUN STATION STEAM GENERATOR TUBE RUPTURE INCIDENT

Attached is our input to NRC's safety evaluation report on the Fort Calhoun Station Steam Generator tube rupture incident. Our input covers all portions of Section 5, "Operation Related Activities." Additionally, we have attached comments from one of our inspectors who witnessed the steam generator inspections. His comments should augment those from the technical review branch which is providing SER input on the licensee's report Section 4, "Visual Inspection and Laboratory Analysis."

From our perspective, we believe that the Omaha Public Power District (OPPD) is following a prudent course of action to determine the causative failure mechanism, responsible chemical species involved, steam generator integrity, and other pertinent factors associated with the incident and plant restart. As a result of our review of OPPD's May 31, June 18, and June 19, 1984, submittals, we have the following two comments.

1. The licensee believes that condenser in leakage creates a concentration of caustics in the secondary chemistry. We do not understand why such caustics would have concentrated on a short portion of one tube and not throughout the steam-blanketed areas of all steam generator tubes. We expect the licensee's on-going evaluation to explore this concern.

2. The licensee's June 19, 1984, report states that the failed steam generator tube was corrosion-constrained at support plates and that such constraint leads to additional stress at the axial location of the failure, which was between support locations. It is not clear to us why the geometric shape of the deformation at the failure site does not appear to reflect a buckling or bowing type of failure that would result from thermal expansion of a constrained tube. But rather, the deformation appears to have resulted from a squeezing of the tube, which might have resulted from clamping the tube in a vise grip or pliers prior to installation in the steam generator. We expect the licensee to pursue this concern in their continuing investigation.

SPES dP	SPES RBI	PSA/1 JJ	RPB1	DRSP RD	RA-RIV JTC
DPowers/jj	RIreland	JJaudon	EJohnson	RDenise	JTCollins
06/21/84	06/21/84	06/21/84	06/21/84	06/21/84	06/22/84

FRPS
BNicholas
6/22/84

DMB-SE04
IE01
1/1

We have concluded that OPPD has been quite responsive to our requirements and conditions of restart as specified in our confirmatory action letter of June 5, 1984. Consequently, we shall lift those restrictions on Fort Calhoun Station operation after NRR's issuance of the SER.

Original Signed By:
Richard P. Denise

R. P. Denise, Director
Division of Reactor Safety and
Projects

Attachments:
As stated

cc: (with attachment)
H. Denton, NRR
G. Lainas, NRR
J. Miller, NRR
E. Tourigny, NRR
J. Axelrad, IE
T. Westerman, EO
D. Powers, RPB 2
D. Tomlinson, SRI
E. Johnson, RPB2
L. Yandell, SRI
R. Ireland, RPB2
RIV File
B. Nicholas, RIV
P. Check, DRA

ATTACHMENT 1

5.0 Operation Related Activities

5.1 Leakage Detection Improvements

The licensee had investigated laboratory capabilities for determining small primary-to-secondary leak rates when a small leak existed during several weeks of operation prior to the end of Cycle 8. Using typical reactor coolant boron and radionuclide concentrations and typical steam generator blowdown rates, the licensee has determined that the smallest leak rate detectable, using boron in hot shutdown, is 0.03 gpm; and using Cs-137 in hot shutdown after refueling is 0.002 gpm. Licensee Procedure CMP-4.68, Revision 0, June 12, 1984, has been issued to provide instructions to calculate the leak rate for each steam generator by gamma isotopic and boron analysis.

The NRC staff has independently reviewed and verified the licensee's laboratory capabilities to determine primary-to-secondary leak rates by reproducing the licensee's leak rate equations and mathematical calculations. The results obtained were compared to those obtained using other industry accepted methods and were found acceptable. The NRC staff also verified that the licensee's calculations were based on realistic and obtainable sensitivity levels and that their analytical procedures and their gamma isotopic analysis equipment were capable of accurately detecting and identifying a small primary-to-secondary leak rate using typical reactor coolant boron and radionuclide concentrations in conjunction with typical steam generator blowdown rates. The NRC staff's calculations verified that the licensee, during Mode 4 operation and continuing into Mode 1 operation, would be able to detect a primary-to-secondary leak rate of 0.03 gpm using their approved boron analytical procedure and a leak rate of 0.002 gpm using a radionuclide measurement of typical operating fission products such as Cs-137.

The NRC staff has reviewed Fort Calhoun Station Special Order No. 35, Revision 0, June 12, 1984, in which the licensee has reduced the maximum allowable primary-to-secondary leak rate through the steam generator tubes from 1 gpm total for both steam generators to 0.3 gpm. In conjunction with this, the licensee has revised ST-RLT-3, "Reactor Coolant system Leak Rate Calculations," to incorporate this additional acceptance criterion into the daily leak rate determination. Anytime an unknown leakage of $\geq$ 0.3 gpm is calculated, the shift chemist will be directed to perform analyses per Procedure CMP-4.68 to determine the primary-to-secondary leak rate. The licensee has committed to applying the action statement of Technical Specification 2.1.4(3) when the primary-to-secondary leak rate is found to exceed 0.3 gpm total for both steam generators.

The NRC staff concludes that the licensee's method of analysis is capable of detecting primary-to-secondary leak rates significantly below the revised limit of 0.3 gpm for both steam generators, and that sufficient administrative instructions have been implemented to ensure adequate steam generator leak rate sampling and plant operational restrictions.

5.2 Sampling Frequency Improvements

To provide early detection of low leakage rates into the steam generators, the licensee has increased the frequency for gamma isotopic analysis of steam generator blowdown from weekly to daily. Boron analysis of steam generator blowdown will be performed once per shift beginning when the plant reaches Mode 4 and continuing until 10 days after reaching Mode 1. Steam generator blowdown monitors RM-054A and B will continue to provide continuous monitoring and automatic blowdown isolation for all but the smallest leaks.

The NRC staff concludes that the licensee's sampling frequency is sufficient to ensure early detection of low leakage rates.

5.3 Procedure Reviews

OPPD Letter LIC-84-160 of May 31, 1984, from W. C. Jones to J. T. Collins, Region IV Administrator, committed the licensee to review the steam generator tube rupture emergency procedures to reconfirm their adequacy. The review team for this effort included the Reactor Engineer (SRO), a training coordinator (SRO), and two licensed operators (one SRO, one RO). This review incorporated the licensee's experience from the May 16, 1984, incident and the applicable lessons learned from the Ginna tube rupture of January 25, 1982. Guidance for the latter review was provided by NUREG 0909, "NRC Report on the January 25, 1982, Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant," Sections 9.0 and 10.0; and NUREG 0916, "Safety Evaluation Report related to the restart of R. E. Ginna Nuclear Power Plant," Sections 1.4.1, 1.4.2, 4.2, 4.3 and 7.4. Specific items analyzed and addressed are documented in Fort Calhoun Station memorandum FC-989-84 dated June 11, 1984. On the basis of this review, the licensee found the existing procedures to be adequate, but has revised Emergency Procedures EP-30, "Steam Generator Tube Leak/Rupture (PPLS Unblocked)," Revision 28 dated June 19, 1984, and EP-30A, "Steam Generator Tube Rupture (PPLS Blocked)," Revision 16, dated June 19, 1984, to clarify and improve the format. Other procedures reviewed included OI-RC-11, "RCS Natural Circulation Cooldown"; OP-6, "Hot Standby to Cold Shutdown"; and EP-35, "Reset of Engineered Safeguards."

The NRC staff has reviewed the revised emergency procedures and concludes that they provide the necessary information and guidance to enable Fort Calhoun Plant operators to take proper action in the event of a steam generator tube leak or rupture.

5.4 Licensed Operator Refresher Training

The licensee has committed to providing all licensed operator personnel with refresher training on the revised emergency procedures, EP-30 and EP-30A, prior to returning the plant to power operation. This training has commenced and will continue until all licensed personnel have been trained.

The Senior Resident Inspector (SRI) has attended one of the training sessions to ensure that the revised procedures were covered in detail, that reasons for changes were explained, and that lessons from the May 16, 1984, incident and the Ginna tube rupture incident were emphasized. The SRI will continue to monitor this training effort to verify that all licensed personnel are trained prior to standing shift while the plant is at power operation.

The NRC staff concludes that the licensee's refresher training effort is satisfactory and that, when all licensed operators have received this training, they will be adequately prepared to act properly in the event of a steam generator tube leak or rupture.

ATTACHMENT 2

4.0 Visual Inspection and Laboratory Analysis

A Region IV NRC inspector observed the actual probing of the Fort Calhoun steam generator tubes on site. Additionally, the NRC inspector reviewed the certifications for the licensee's inspection personnel who conducted the 1982 and 1984 inspections. The NRC inspector also reviewed and confirmed the independent verification of the data gathered for each tube. The NRC inspector has also reviewed the licensee's June 19, 1984, report and concludes that it is comprehensive and that it satisfactorily analyzes the problem. Details of this inspection will be issued in NRC Inspection Report 50-285/84-14.