JUN 22 1984

In Reply Refer To:
Docket: 50-285

Omaha Public Power District
ATTN: W. C. Jones, Division Manager
Production Operations
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

SUBJECT: RESTART OF THE FORT CALHOUN STATION, UNIT 1

By letter dated May 22, 1984, you notified us that a steam generator tube failure occurred at the Fort Calhoun Station, Unit 1 on May 16, 1984, during the hydrostatic pressure test of the reactor coolant system. By letter dated May 31, 1984, you provided specific details of the event, a description of past steam generator inspections, and the results of the failed tube visual inspections and laboratory analyses. By letter dated June 19, 1984, you submitted an update of the report you provided to us on May 31, 1984.

The purpose of this letter is to provide you our safety evaluation which is based upon your submittals described above. Our safety evaluation, which is enclosed, contains the following summary of conclusions and recommendations:

We conclude that the operators had full control of the station during the event and subsequent to it, and have acted responsibly.

We accept your conclusions that the tube failure was due to outside diameter initiated stress corrosion cracking, and since there is no evidence to the contrary at this time, caustic is a reasonable first candidate as the causative agent. However, 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 your ongoing evaluation to explore this concern. The preliminary profilometry data indicates that tube ovalization/denting is occurring in those tubes at the outer areas of the tube bundle that pass through all three vertical support straps, with maximum deformation occurring at the strap on the hot-leg side of the generator. This is consistent with ovalization and location of the failed tube in the generator and provides the evidence as to the source of the stress component of the observed stress corrosion cracking. However, there is not sufficient supporting data to fully explain the tube degradation that you have experienced. Your June 19, 1984, report states that the failed 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

SPES dlP	SPES RBI	PSA/I	RPB1	DRSP	RA-RIV
DPowers/lk	REIreland	JPJaudon	EHJohnson	RPDenise	JTCollins
6/22/84	6/22/84	6/22/84	6/22/84	6/22/84	6/22/84

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S PDR

DMB
IE31
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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 you to pursue this concern in your continuing investigation. Therefore, unless you can provide additional justification that a mid-cycle inspection is not warranted, all tubes presently with dent indications at the vertical support locations must be examined, in addition to eddy-current, with profilometry 9 months following initial power operation (Mode 1) to measure ovality/denting so that in the event denting is not arrested we can establish a strain criteria for preventive plugging in the future. We also request that you complete the analysis of the profilometry data and reduce the tube diametric values to percent of permanent strain so that a baseline can be established for future profilometry tests. The additional justification schedule should be on a compatible schedule with strain results of the profilometry data and other planned corrective actions, as contained in Section 6.0 of the June 19, 1984, submittal, and should be submitted for NRC staff review no later than 7 months into full power operation.

We conclude that the lower primary-to-secondary leak rate limit of 0.3 gpm total for both steam generators is appropriate. This administrative limit should remain in effect until further notice. Your proposed future operation-related activities are also acceptable. This includes the method of analysis for detecting leakage, sampling frequency, emergency procedures revisions, and operator training refresher.

Based on the above, we conclude that you have met the requirements of our June 5, 1984, letter and that the Fort Calhoun Station, Unit 1 can safely return to power operations. On this basis, you are authorized to return the Fort Calhoun Station, Unit 1 to service.

Sincerely,

Original signed by
John T. Collins

John T. Collins
Regional Administrator

Enclosure:
As stated

cc:
W. G. Gates, Manager
Fort Calhoun Station
P.O. Box 399
Fort Calhoun, Nebraska 68023

Harry H. Voight, Esq.
LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, NW
Washington, DC 20036

bcc to DMB (IE31)

bcc distrib. by RIV:

RPB2	Resident Inspector	J. Collins, RA
TPB	Section Chief (RPB2/A)	
RIV File	R. Denise, DRS&P	

KANSAS STATE DEPT. HEALTH
NEBRASKA STATE DEPT. HEALTH

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 1000
ARLINGTON, TEXAS 76011

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