Omaha Public Power District

1623 HARNEY ■ OMAHA, NEBRASKA 68102 ■ TELEPHONE 536-4000 AREA CODE 402

August 8, 1979

Mr. K. V. Seyfrit, Director
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region IV
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

Reference: Docket No. 50-285

Gentlemen:

The Omaha Public Power District hereby requests permission to perform the inspection program specified in IE Bulletin 79-13 during the next scheduled refueling outage at the Fort Calhoun Station in January 1980. The District received IE Bulletin 79-13, dated June 25, 1979, requesting that an inspection program be conducted within 90 days of the date of the Bulletin, to evaluate indications of all feedwater nozzle-to-piping welds and of adjacent pipe and nozzle areas.

Since the receipt of the Commission's letter, the District has evaluated the design and operational characteristics of the Fort Calhoun Station with respect to the potential for weld and pipe failures as described in the Bulletin. In particular, the District examined: steam generator nozzle design, secondary water chemistry, piping and nozzle stress levels, thermal transients, and crack propogation parameters. The results of this examination demonstrate that the Fort Calhoun Station is not susceptable to the kinds of failures described in the Bulletin. A detailed technical discussion is attached in support of this position.

Although it is felt that the Fort Calhoun Station has not experienced any cracking in the vicinity of the feedwater piping-to-nozzle welds, it is recognized that these types of failures have occurred at other facilities, and there is, therefore, sufficient cause for concern. However, as further stated in the attached information, the potential for cracking is sufficiently small at our station so as to warrant an extension in the inspection schedule from September 23, 1979 (90 days after the date of Bulletin 79-13) to January, 1980. Since a reactor shutdown for a minimum of eight days will be required to perform these inspections, extending the schedule to January, 1980, will permit the District to take advantage of a scheduled refueling outage to fulfill the bulletin requirements and, at the same time, assure continued station availability during the summer months without jeopardizing the health and safety of the public. In addition, extending the schedule would result in a savings to the District of approximately $722,000 in net costs for replacement power.

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Considering the foregoing discussion, the Commission is respectfully re quested to grant a schedule extension to January, 1980, for the performance of the subject inspection (designated in items 1a, b, c, of the bulletin) and provide such determination on a timely basis. Should a cold shut down of sufficient duration occur prior to the next scheduled refueling outage, then the exams would be performed at that time. The District's staff is available to discuss this matter, should further information be desired.

Sincerely,

for T. E. Short
Assistant General Manager

TES/KJM/BJH/rh

Attachment

xc: U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

Director of Nuclear Reactor Regulation
Attn: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

REQUEST TO EXTEND THE DEADLINE FOR THE FEEDWATER NOZZLE-TO-PIPE AREA RADIOGRAPHIC EXAMINATION OF IE BULLETIN 79-13

1.0 POTENTIAL FOR CRACK OCCURRENCE

The Fort Calhoun Station steam generators have design features and an operating history that differs from those units in which cracks have been observed. These design features and the plant operating history, which are discussed below should significantly reduce the likelihood of cracks that have been observed in other units.

1.1 Nozzle Design

The steam generator nozzle cracks observed at other units may be largely due to design features that do not exist at the Fort Calhoun Station.

The Fort Calhoun Station nozzle design provides an excellent match to the feedwater piping and is dissimilar from the designs shown in References 1 and 2. Table 1 lists the pertinent Fort Calhoun piping and nozzle parameters.

TABLE 1

	Inside Diameter	Outside Diameter	Wall Thickness
Feedwater Nozzle (safe end)	14.312 in.	16.562 in.	1.125 in.
Feedwater Piping (16 in. Sched. 80)	14.312 in.	16.000 in.	0.844 in.

This connection has the following features:

a) An even alignment of inside diameters;

b) Slight mismatch of outside diameters and wall thickness (standard design practice), but never reduced to less than design wall thickness for schedule 80 pipe; and

c) No backing rings were used to fabricate this connnection.

It can readily be seen that no significant discontinuity stresses on the inside surface should exist at the connection of the piping to the nozzle. Details of the nozzle design may be seen on drawing CE-E-232-485-5, which was submitted to the Commission on June 18, 1979, in response to the Commission's letter of May 25, 1979, on this subject.

1.0 POTENTIAL FOR CRACK OCCURRENCE (Continued)

1.1 Nozzle Design (Continued)

This design must be compared to other designs in which cracking has been observed. In a design prevalent at plants in which cracking has been observed, the nozzle wall thickness is about .656 inches, approximately 22% less than minimum pipe wall thickness for 16-inch schedule 80 pipe.

In such a design, not only is there a stress discontinuity where the piping was machined down to match the nozzle inside diameter, but that portion of the pipe is considerably weaker than the remainder of the piping. This situation does not exist at the Fort Calhoun Station.

1.2 Water Chemistry

Stress assisted corrosion cracking can be caused by improper water chemistry control. The Fort Calhoun Station, however, has had excellent chemistry control. Data taken from January, 1974, through July, 1979, for dissolved oxygen in the feedwater show that 95.23% of the readings show no oxygen at all, as illustrated in Table 2.

The auxiliary feedwater source (the emergency feedwater storage tank) is maintained with a residual level of hydrazine which is checked on a weekly basis and is covered with a nitrogen blanket to keep the water oxygen free.

TABLE 2

Dissolved O_2 Level, Feedwater, in ppm
Readings Accurate to ± 5 ppb
Data Period - January, 1974, through July 5, 1979

DO_2	Percent Occurrence
0.000	95.2
<0.005	98.1
≤0.010	99.3
≤0.015	99.8
≤0.020	100.0

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1.0 PO[illegible]NTIAL FOR CRACK OCCURRENCE (Continued)

1.3 Piping and Nozzle Stress Levels

The feedwater piping is designed to withstand the combination of pressure, dead weight, thermal, and seismic loading. The stress analysis (computer printout enclosed) shows that calculated stress levels as defined in the USAS B 31.7 code are well below the allowable. No seismic or water hammer type events have occurred since piping installation, and the allowable stress levels have never been exceeded.

1.4 Thermal Transients

The feedwater nozzles are not subject to rapid thermal transients during startup, shutdown, or load changes due to the gradualness of the condensate/feedwater system temperature changes. Even during a plant trip, feedwater temperatures will change slowly over several hours with temperatures reducing from a full load value of 440°F to ambient. Sudden introduction of cold auxiliary feedwater flow to a hot line will not produce rapid temperature transients in the feedwater nozzle, as the auxiliary feedwater connection to the main feedwater system is upstream of the feedwater control valves outside of the containment building. This means that the initial auxiliary feedwater flow must travel through a 200 foot long section of a large, already hot line before reaching the steam generators. Thus, there will be no severe thermal transients due to the introduction of auxiliary feedwater flow. The "worst case" heatup and cooldown rates have been used in the enclosed crack growth analysis discussed in Section 2.6 below.

1.5 Operating History

The Fort Calhoun Station has had a relatively stable operating history with the plant generally operating at or near rated load. The plant history includes only 16 cold shutdowns and 38 hot shutdowns. There have been no water hammer or severe vibration problems on the main feedwater piping at Fort Calhoun. Severe water hammers or vibrations have been noted at several of the plants in which feedwater nozzle cracks have been noted. It is important to note that one plant which has had a history of water hammer problems, Calvert Cliffs Unit 1, is another Combustion Engineering plant and has completed the feedwater nozzle examinations directed by the IE Bulletin 79-13. Their examination revealed no cracks. The materials used in fabricating the nozzle and safe end of the Calvert Cliffs steam generators are identical to those used in the Fort Calhoun steam generators. The radiographs of the nozzle-to-pipe welds of the Fort Calhoun steam generators have been re-examined, finding only minor amounts of slag or porosity. Several of the radiographs displayed "No Apparent Defects."

1.0 POTENTIAL FOR CRACK OCCURRENCE (Continued)

1.6 Crack Growth Study

Enclosed is A Study of Crack Growth Rate in the Fort Calhoun Feedwater Nozzle, performed by Pacifica Technology. The study, using historic operating and design data from the Fort Calhoun Station, and methods consistent with the philosophy and practice of the ASME Boiler and Pressure Vessel Code, Sections III and XI, shows that the worst possible crack that could have hypothetically occurred thus far, due to primary and secondary stresses at the Fort Calhoun nozzle, would easily meet the Section XI requirements for continued operation without repair.

2.0 EFFECT OF FEEDWATER NOZZLE CRACKING

Cracking or rupture in the Fort Calhoun Station feedwater nozzles will not preclude safe shutdown of the plant.

There are two possible adverse effects of a pipe line crack or rupture. One is the loss of the line's function. The other is damage to essential equipment caused by the ruptured line.

Analysis, reported in Reference 3, has been performed that shows the plant can be safely shut down following the postulated rupture of a feedwater line. Furthermore, the loss of main feedwater lines will not prevent supplying auxiliary feedwater to the steam generators since each steam generator has a separate emergency feedwater nozzle that is connected to the auxiliary feedwater system. In the event of a feedwater line rupture, the affected steam generator would be isolated from the feedwater system while maintaining the ability to feed the intact steam generator.

Restraints and other protective devices have been provided to prevent damage to essential equipment following the postulated rupture of a feedwater line.

Any leakage from the secondary system inside of containment will be detected by the containment humidity detectors and the containment sump level detectors. The information can be correlated with both primary and secondary system inventories to determine if and from where a leak is occurring.

3.0 REFERENCES

3.1 IE Bulletin 79-13, USNRC.

3.2 Minutes of USNRC Public Meeting "Briefing on Feedwater Nozzle Cracks in Westinghouse Reactors."

3.3 Fort Calhoun Station Unit 1 Final Safety Analysis Report.