SUMMARY OF PRESENTATION

ON FEEDWATER LINE CRACKING

AT H. B. ROBINSON MADE

BY CP&L ON JUNE 27, 1979

TO THE NRC STAFF

7907170026

Section No.	Topic
1.0	Feedwater Piping Arrangement
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3.0	Feedwater Line Support Inspection
4.0	Nozzle and Reducer Repair
5.0	Metallurgical Evaluation of Feedwater Line Cracks
6.0	Stress Analysis of Feedwater Line Piping
7.0	Feedwater Line Instrumentation Program
8.0	Summary and Corrective Action

1.0 Piping Arrangement

In each of the three steam generator feedwater lines a vertical, 16-inch carbon steel pipe is welded to a 90-degree, long radius elbow which in turn is welded to a 16-inch by 18-inch reducer. The 18-inch end of this reducer is welded to an 18-inch low alloy nozzle on the steam generator. All 16-inch piping is fabricated of Schedule 100 A-106 GR B Carbon Steel; the elbows are A234 WPB each being a 24-inch long radius; the reducers are Schedule 100 A-106 GR B Carbon Steel.

2.0 Inspection of Feedwater Reducers and Nozzles

On June 9, 1979, H. B. Robinson Unit No. 2 proceeded from hot shutdown (operating temperature and pressure) to cold shutdown to facilitate the repair of a CRDM canopy seal weld leak. During this repair period, a parallel effort was directed toward the radiographic inspection of the feedwater nozzle to reducer welds to determine if H. B. Robinson had incurred the cracks previously discovered at the D. C. Cook and San Onofre Plants.

The radiographic inspection of the nozzle to reducer welds (18-inch O.D.) revealed circumferential cracking in the machined area of each reducer. This cracking was most predominant in the 3 o'clock and 9 o'clock positions with reducers "B" and "C" showing the most severe indications. No crack indications were found in the 16-inch end of the reducers.

Prior to removing the reducers, the centerline of each nozzle to reducer weld was located to ensure preservation of the area of indications. The nozzles were then removed by grinding through the centerline of the welds.

After the reducers were removed, the inside surfaces of the steam generator nozzles were buffed and PT inspected. The PT inspection revealed several linear, circumferential crack indications in the area of the nozzle lip. These indications were most predominant in the lower half of each nozzle. Moderate pitting was also discovered in the lower half of each nozzle. The pitting extended from the nozzle lip to the taper at the thermal sleeve and in the case of "A" and "B" nozzles, the pitting appeared to continue below the thermal sleeve.

To properly inspect the area around the "A" thermal sleeve a 0.5-inch ring of thermal sleeve was removed from the entire circumference of "A" thermal sleeve. This verified that the pitting was limited to a small area about the 6 o'clock position on the nozzle. A one by three-inch half moon section of thermal sleeve was then removed from the 6 o'clock position to gain access to the pitting. The fitup of "B" thermal sleeve to "B" nozzle was not as tight as "A" and therefore did not require the removal of a 0.5-inch length along the circumference to gain access for inspection. However, the pitting under the thermal sleeve of "B" nozzle was more extensive and required a 1.75-inch by 7.5-inch half moon section of thermal sleeve to be removed prior to reaching the termination of pitting.

3.0 Feedwater Line Support Inspection

An inspection of feedwater line supports from the containment vessel wall to the steam generator nozzles was performed by EBASCO Services and Carolina Power & Light Company during the 1979 outage. This inspection showed all piping and piping supports behaving as designed. There were no signs of stress on any supports. Three spring hangers, each at the base of the vertical run of pipe preceeding the reducer to nozzle section were observed to be off scale while the plant was in a cold condition. These supports were identified as being off scale in a previous ISI inspection but were observed to be on scale when the plant was returned to a hot condition. EBASCO Services inspected these supports and judged them to be satisfactory.

At the beginning of the 1979 outage two thermal restraints in "A" feedwater line were found to have spalled concrete adjacent to the embedded plates on which they were mounted. In one case this condition was caused by the embedded plates being installed too close to an existing penetration in the supporting polar crane wall. In the other case, insufficient clearance was found between the sliding and fixed members of the support. Both restraints have been repaired through modifications.

4.0 Nozzle and Reducer Repair

All linear indications were removed from the nozzles by careful grinding. Pitting which exceeded the Westinghouse acceptance criteria was also removed by grinding. Weld repairs to the nozzle base material were performed in all cases where the ground area exceed the following criteria:

a. Any local grind-out in excess of 0.050-inch in the nozzle counterbore or 15° taper.

b. Any local grind-out in excess of 0.075-inch in the area more than 1.5-inch from the face of the nozzle weld prep.

Areas which did not exceed the specification for weld repair were ground with a minimum 4:1 taper to the nozzle ID and a minimum 0.5-inch radius.

The sections removed from "A" thermal sleeve were minimal and had no impact on the performance of the thermal sleeve. Therefore, the removed sections of "A" thermal sleeve will not be replaced. The half moon section which was removed from "B" thermal sleeve was much larger and was repaired by welding a half moon shape of compatible material to the thermal sleeve. This weld was made with a "V" groove and an integral backing ring.

Three new schedule 100, 16-inch by 18-inch, A 106 GR B, carbon steel reducers were purchased as replacements. The counterbore on the 18-inch end of the reducer, necessary to match the schedule 100 reducer to the schedule 60 nozzle, will be 1.25 inches in length. This will ensure that discontinuity, caused by the taper, is removed from the area of the weld. The taper will be blended to the reducer ID with a minimum 4:1 slope and generous 1.0-inch radius.

The backing ring welding technique, used in initial construction, will be replaced by a consumable insert technique. These modifications are designed to minimize the stresses in the areas adjacent to the nozzle to reducer weld and eliminate the crevice where corrosion products or chemicals would tend to collect.

To minimize the time in which the weld area will be under stress and the nozzle susceptible to Hydrogen absorption, the nozzle to reducer weld area will not be cooled for radiography of the root; the weld will be completed and post weld heat treatment begun immediately.

5.0 Metallurgical Evaluation of Feedwater Line Cracks

A 360° specimen from Loop C was examined to determine the extent and metallurgical cause of the cracks. Radiography had previously indicated cracks located approximately 1/2" from the weld certerline on the feedwater line side of the nozzle weld and extending circumferentially around the reducer. The 360° specimen was removed by cutting along the certerline of the weld. An ultrasonic inspection from the cup face showed cracking 360° around the reducer except at two regions located between 10:45 to 11:30 and 1:00 to 2:00. The deepest crack penetration was at 9:00.

A specimen was removed from the 9:00 position for more detailed metallography. This inspection revealed a 0.75" deep crack (about 1/2 way through the wall at the transition between the counterbore and the taper of the machined region in the reducer). On a macroscopic scale, the crack was very straight, but on a microscopic scale, some branching and oxide protrusions could be seen. Beach marks were observed and electron microscropy revealed some striations at the crack tip. The location and morphology of the crack were almost identical to those previously observed on other cracks that were caused by corrosion fatigue.

Small secondary cracks, parallel to the main crack, were observed between the weld and the main crack. These branching cracks were located in a crevice beneath the backing ring, although there were no cracks in the backing ring itself. The morphology and location of these cracks suggest that stress corrosion was taking place. The cracks were all less than 0.1" deep and grew progressively smaller as the distance from the tip of the crevice increased.

As a result of this investigation, it was concluded that the cracking was caused primarily by corrosion fatigue with stress corrosion playing a secondary role.

6.0 Stress Analysis of the Feedwater Piping

Stress analysis was performed on the feedwater line configuration in an effort to determine the mechanism causing the observed cracking. This analysis was broken into three parts:

1. Structural analysis of the feedwater line including the effects of thermal, deadweight and pressure.

2. 2D finite element fatigue analysis of the feedwater nozzle/elbow configuration.

3. Frequency analyses of the feedwater line and steam generator.

The structural analysis was performed using a 2D finite element model of the feedwater line with anchors included at the steam generator (SG) and containment penetration and the vertical and horizontal thermal growth of the SG applied at the feedwater nozzle. The WESTIA717 computer code was used for the analysis. The geometry consists of the feedwater nozzle with an 18" Schedule 60 and prep which is connected to an 18" X 16" schedule 100 reducer, then to a 16" schedule 100 elbow and a vertical pipe run of approximately 20'. This vertical run is followed by from 5 to 7 pipe segments, varying in length from 50' for Loop A to 128' for Loop B, running to the containment penetration. Supports consisted of deadweight spring hangers and several horizontal rigid supports.

Two thermal conditions were run. The first with the SG at 547°F and the feedwater line at 450°F representing normal operation. The second with the SG at 547°F and the feedwater line cold representing the hot shutdown condition. The analyses results show a maximum thermal stress of approximately 16 ksi at the nozzle to pipe junction. The maximum deadweight and pressure stresses were 1.5 ksi and 5.0 ksi respectively. These stresses are well below code allowable values.

The second analysis performed was a detailed 2D finite element fatigue analysis of the most severe thermal transient, which occurred during hot shutdown, in the region of the feedwater nozzle to reducer junction. The analysis used the KECAN computer code and the rules of ASME Section III, MB-3200. The model used constant strain quadralateral elements with a minimum of 8 model through the wall and ran from the SG shell to 10" beyond the nozzle to reducer weld. The transient analyzed consisted of a ramp change in temperature from 547°F to 60°F in 9 seconds followed by a period of constant 60°F operation, flow velocity of .38 ft/sec. This represents the injection of auxiliary feedwater into the feedwater nozzle/elbow junction, which has been heated by the SG during the hot standby condition. When the auxiliary feedwater is terminated, a step change in temperature is assumed from 60°F to 547°F. This represents the conservative assumption of a leaking check valve in the feedwater system, which allows water to flow from the SG to the feedwater line and assumes no mixing of auxiliary feedwater with water in the SG. The maximum peak stress range obtained from this transient was 70 ksi which is then multiplied by a conservative factor of 1.7 to account for the detailed affect of the "notch" at the reducer counterbore. This peak stress range of 120 ksi yields an allowable 2000 cycles using the ASME Section III S/N curves. It should be noted that if the assumption of a leaking check valve were not used (while still using a stress intensification factor of 1.7) the allowable number of cycles using Class 1 piping rules would be greated than 9000 cycles. The design transients given in the SG E-Spec has shown acceptable values of usage factors for the feedwater nozzle. Corresponsingly, analysis of the nozzle to elbow junction will have an acceptable value of usage factor since the thermal transient stresses are lower at this junction than in the nozzle. The final analysis performed was a frequency analysis of the feedwater line. A total of 20 nodes were found with frequencies in the range of 1 to 10 Hz for all three lines. These frequencies are in the same range as those found for the SG in the reactor coolant loop analysis and those observed in the line during normal operation. Westinghouse testing of other plants has shown that the SG vibrates its fundamental nodes due to flow in the reactor coolant loop. This yields the possibility that the feedwater line could be in resonance with the SG and could cause high enough stress in the nozzle to elbow junction to cause the observed cracks. Final resolution of this possibility must await results of feedwater line instrumentation to determine if resonant vibration exists. This is necessary due to the degree of uncertainty in determining the feedwater line and SG frequencies and the closeness of those frequencies that must be demonstrated to show resonance.

The resulting stresses are to be within the design allowable per the ASME Section III, Division I, Figure I-9.1 (S/N) curves.

Temperature

The number of incidents of differential temperature between the inside pipe surface temperature and outside pipe surface temperature shall not exceed the number of design transients for which the reducer has been designed.

Off-Site Acceptance Criteria

The measured parameters of pressure temperature, strain, and acceleration will be reduced to evaluate total piping system performance with respect to design. The criterion for acceptability of the piping system performance is that the stress intensities obtained from the measured data for each operating condition are less than the allowable design stress intensities specified in ASME Code, Section III, Subsection NC-3650.

7.0 Feedwater Line Instrumentation Program

The B and C feedwater piping inside the containment building will be instrumented prior to startup and monitored during plant operation. The purpose of such monitoring is to obtain data to characterize the dynamic response of the feedwater piping due to mechanical and flow induced motions. The data will be used to determine if the insulated piping system is consistent with the design assumptions and is acceptable for continued system operation.

The test objective will be achieved by monitoring the system performance of the feedwater piping at pre-selected locations using electronic transducers. The system performance will be characterized by monitoring parameters comprising:

1. Pressure (2 locations).

2. Temperature (21 locations).

3. Acceleration (13 locations).

4. Strain (15 locations).

5. Displacement (3 locations).

The measured parameters will be recorded during various modes of system operation and will be reduced in order to evaluate system performance with respect to the acceptance criteria.

ACCEPTANCE CRITERIA

Two acceptance criteria are established: 1) on-site criteria to be used as an initial screening tool for piping system acceptability and 2) off-site criteria to evaluate total system performance with respect to design conditions.

On-Site Acceptance Criteria

Stress Acceptance Criteria

The strain gauge data will be evaluated to determine the magnitude and frequency of the dynamic stress due to mechanical and flow induced vibrations.

8.0 Summary and Corrective Action

The metallurgical investigation showed the cause of cracking to be primarily corrosion fatigue with stress corrosion playing a secondary role.

It is suspected that disolved oxygen may have been a contributing factor in the feedwater line cracking. Therefore, Carolina Power & Light Company is reviewing all possible means of reducing the oxygen content below the presently accepted limits. The backing ring welding technique will be replaced by a consumable insert technique to eliminate the crevice where corrosion products and chemicals could collect.

Stress risers in the weld area have been eliminated by buffing out all irregularities on the reducer I.D. and using a minimum 4:1 slope with a generous radius to blend the reducer counterbore to the reducer I.D.

The instrumentation test program will monitor the dynamic response of the feedwater piping during start-up, hot standby and transients. This program is designed to identify the source of high cycle fatigue by monitoring the pressure, temperature, acceleration, strain and displacement of "B" and "C" feedwater lines.

An ultrasonic inspection of each reducer to nozzle weld area will be performed after the post weld heat treatment is completed. This inspection will be used as a baseline for future inspections of these welds.

Docket 50-261

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 12, 1978

ALL POWER REACTOR LICENSEES

Gentlemen:

This letter is being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with application for a license to operate a power reactor (FSAR docketed).

The NRC recently notified you that it had scheduled a series of meetings to discuss implementation of upgraded guard qualification, training and contingency planning requirements. The Region I meeting scheduled to be held on October 11, 1978 has been rescheduled to October 13, 1978. A revised schedule is enclosed.

For further information or comments, please contact Tom McKenna of my staff on (301) 492-7846.

Sincerely,

Jack W Roe
for

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosure:
Meeting Schedule & Locations

cc: Service List

Misc App
cp

REVISED SCHEDULE

9:30 to 3:30

Region II	September 27, 1978	Stadium Hotel* 450 Capitol Ave., SE Atlanta, GA 30312 (404) 688-1900
Region III	October 3, 1978	Ramada O'Hare 6600 North Mannheim Rd. Des Plains, IL 60018 (312) 827-5131
Region IV & V	October 5, 1978	San Francisco Airport Hilton P. O. 8355 San Francisco, CA 94128 (415) 589-0770
Region I	October 13, 1978	Valley Forge Holiday Inn 260 Goddard Blvd. King of Prussia, PA 19406 (215) 265-7500

*Special rate for reservations received before September 15, 1978.

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, D.C. 20036

Hartsville Memorial Library
Home and Fifth Avenues
Hartsville, South Carolina 29550

Docket
50-261

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 25, 1978

ALL POWER REACTOR LICENSEES

Gentlemen:

This letter is being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with application for a license to operate a power reactor (FSAR docketed).

The NRC has scheduled regional meetings to discuss the upgraded guard qualification and training requirements published in the Federal Register on August 23, 1978 and the guidance on this requirement contained in NUREG 0219 as well as the contingency planning requirements published in the Federal Register on March 23, 1978. An agenda, the dates and locations of the meetings, and a registration form are enclosed. Please complete the registration form and return it to Mr. Frank G. Pagano, Jr., Chief, Reactor Safeguards Development Branch, Nuclear Regulatory Commission, Washington, D.C. 20555. Hotel arrangements are the responsibility of each attendee.

For further information or comments, please contact Tom McKenna of my staff on (301) 492-7846.

Sincerely,

James R. Miller

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosures:
1. Meeting Agenda
2. Meeting Schedule & Locations
3. Registration Form

CC: Service List

ENCLOSURE 1

IMPLEMENTATION OF 10 CFR 73 APPENDICES B AND C GUARD TRAINING AND CONTINGENCY PLANNING

Meeting Agenda

Sept. 27 - Atlanta; Oct. 3 - Chicago
Oct. 5 - San Francisco; Oct. 11 - Philadelphia

TIME	SPEAKER	SUBJECT
9:00 - 9:10	J. Miller	Introduction
9:10 - 9:20	V. Stello	NRC Safeguards Responsibility
9:20 - 9:30	F. Pagano	Why We Adopted This Approach
9:30 - 9:45		Coffee Break
9:45 - 10:30	T. McKenna	The Approach
10:30 - 11:15	T. McKenna	10 CFR 73 Appendix B Guard Training
11:15 - 11:45	J. Roe	Contingency Plans
11:45 - 12:00	R. Clark	NRR Staff Reviews
12:00 - 1:30		Lunch
1:30 - 3:00		Question/Answer
3:00 - 3:30	J. Miller/ V. Stello	Closing Remarks

ENCLOSURE 2

SCHEDULES AND LOCATIONS

Region II	September 27, 1978	Stadium Hotel* 450 Capitol Ave., SE Atlanta, GA 30312 (404) 688-1900
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*Special rate for reservations received before September 15, 1978.

ENCLOSURE 3

Registration Form

Implementation of 10 CFR 73 Appendix B
Security Personnel Qualification Training and
Equipment Requirements by Commercial Nuclear
Power Reactors

Regional Meeting

Date ______________________________

Place ______________________________

Utility Represented ______________________________

Individuals Attending:

Name ______________________________

Title ______________________________

Office Phone ______________________________

Name ______________________________

Title ______________________________

Office Phone ______________________________

Name ______________________________

Title ______________________________

Office Phone ______________________________

Name ______________________________

Title ______________________________

Office Phone ______________________________

RETURN THIS FORM BY SEPTEMBER 22, 1978 TO:

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
ATTN: Frank G. Pagano, Jr.
Washington, D. C. 20555

cc: G. F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, NW
Washington, D.C. 20036

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Hartsville, South Carolina 29550