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Director of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

RESPONSE TO NRC COMPLIANCE BULLETIN 88-02

The steam generator tubing at Prairie Island Units 1 and 2 has been evaluated for susceptibility to high cycle fatigue cracking at the 7th tube support plate for rows 8 through 12. This evaluation is documented in the Westinghouse proprietary document WCAP 11787, "Prairie Island Units 1 and 2 Evaluation for Tube Vibration Induced Fatigue", which is now being finalized. WCAP 11787 will be forwarded to you under separate letter by April 29, 1988.

CONCLUSION

The tube fatigue evaluation in WCAP 11787 shows that no modification, no preventive tube plugging, or other measures are necessary in the Prairie Island steam generators, even if a tube is dented at the 7th tube support plate.

PRAIRIE ISLAND STEAM GENERATORS

The Prairie Island Steam Generators are Westinghouse Model 51 steam generators with drilled carbon steel support plates, Alloy 600 tubing and open tubesheet crevices (2½ inch lower hard roll). There are 123 tubes plugged in Unit 1 (1.8%) and 189 tubes plugged in Unit 2 (2.8%). The major reasons for plugging are thinning at the lower cold leg tube support plates, fretting wear at the old anti-vibration bar locations, and stress corrosion cracking within the tubesheet crevice (so far, only in 12 steam generator). There are tubes with denting indications, but no tubes are plugged due to dents. Since 1982, 100% of the steam generator tubing (with minor exceptions) has been examined by eddy current each refueling outage.

IE11

The Westinghouse new design AVBs are installed in all steam generators at Prairie Island.

WESTINGHOUSE EVALUATION CRITERIA

Because dent indications do exist in the region of interest and because new anti-vibration bars were installed in Unit 1 in March 1986, and in Unit 2 in October 1985, Westinghouse was contacted in November, 1987, to evaluate the susceptibility of the Prairie Island steam generators to vibration induced fatigue cracking. The criteria established for assuring more than a 40 year fatigue life is a 10% stability ratio reduction that provides at least a 58% reduction in stress amplitude (to $\leq$ 4.0 ksi) for a Row 9 tube in the North Anna 1 steam generators. This same criteria were applied to the evaluation of Prairie Island 1 & 2.

A tube is acceptable if it meets one of the following criteria:

1. No denting at the 7th tube support plate: This includes both the deformed dented tube condition and the clamped tube condition.

2. Tube has AVB support: The criterion for establishing the tube has support from an AVB is at least one-sided support is present to the tube centerline.

3. Stability Ratio is 90% or less of the North Anna failed tube (R9C51): Tubes with different radius U-Bends will have different stiffness and frequency and therefore different stresses and fatigue usage per year than the Row 9 tube. A stress ratio is used to account for these effects. The stress ratio, also, was formulated so that a stress ratio of 1.0 or less assures acceptable stress amplitudes and fatigue usages for Prairie Island Units 1 and 2.

DENTING EVALUATION

A total of fifteen tubes were identified with dent indications at the 7th tube support plate in the region of interest. In addition, the eddy current data was evaluated for evidence of support plate corrosion. About one-half of the examined tubes in Row 11 show indications of support plate corrosion. All Row 11 tubes were evaluated from 10-86 and 1-88 data with similar results. Denting indications are stable with no apparent growth.

AVB INSERTION DEPTHS

AVB locations were determined using eddy current data. If both legs of an AVB assembly are identified on one or both legs of the tube, AVB support at the tube centerline is assured. If

only a single signal is seen, AVB support can not be assured since the lower hinge plate of the AVB assembly can be seen by the eddy current probe starting above tube centerline. For those cases, extrapolation techniques were used to determine the lowest penetration of each AVB hinge plate.

Overall the evaluation shows that all Row 12 tubes are supported by AVBs. Row 12 is the design depth of insertion for the modified AVBs except for a special AVB at each side of the tube bundle which penetrates to about Row 10. The evaluation also shows that the AVBs have very uniform insertion depth with the bottom hinge plate located between Rows 11 and 12.

FLOW PEAKING FACTORS

Tests were performed to determine the flow peaking factors for Prairie Island 1 and 2 AVB configurations relative to the North Anna R9C51 peaking factor. The modified AVB design was used in the tests for the Prairie Island values and the original AVB design was used for R9C51. Since the Prairie Island AVBs show uniform AVB insertion to less than one tube pitch, the tests focused on sensitivity to uncertainties in determining the AVB depth of insertion. The test results were used to define an upper bound of the ratio for "uniform AVB insertion" related to the R9C51 configuration. It was found that the results are enveloped by the peaking factor of 1.47 determined for R9C51 relative to uniform AVB insertion for the original AVB design.

VIBRATION EVALUATION

The calculation of relative stability ratios for Prairie Island makes use of detailed tube bundle flow field information computed by the ATHOS steam generator thermal/hydraulic analysis code. Code output includes three-dimensional distributions of secondary side velocity, density, and void fraction, along with primary fluid and tube wall temperatures. Distributions of these parameters have been generated for every tube in the Prairie Island tube bundle based on a recent full power operating condition. This information was factored into the tube vibration analysis leading to the relative stability ratios for each tube in Rows 8 through 12 relative to North Anna/tube R9C51.

The uniform AVB insertion depths allow the relative stability ratios to be divided by the North Anna 1, R9C51 flow peaking factor. All tubes are less than the 0.9, 10% criteria. In addition, all stress ratios were found to be less than 1.0. Cumulative usage calculations for the entire duty cycle show that the 40 year fatigue usage is less than 1.0, even when assuming the tubes are dented.

WCAP 11787 includes additional detail and supporting test data.

PRIMARY TO SECONDARY LEAK RATE MONITORING

At Prairie Island, primary-to-secondary leaks are monitored by the condenser air ejector monitor, the steam generator blowdown monitor, and by the daily reactor coolant system leak rate calculation. The air ejector monitor has been very reliable since a dry well modification was done 2 years ago. It is a continuous on-line monitor which alarms in the Control Room. In addition, the air ejector monitor and steam generator blowdown monitor readings are available on the Emergency Response Computer System. Thus, they can be trended continuously by Control Room operators on the Emergency Responsible Computer System monitors.

Prairie Island Technical Specification 3.9 requires grab samples to be taken once per 24 hours when an air ejector monitor is out of service. Administratively, grab samples are now required to be taken once per 8 hours when an air ejector monitor is out of service. This was implemented January 13, 1988.

Leak rate monitoring and trending requirements will be increased. This is being done to fulfill the interim requirements of NRC Bulletin 88-02 until WCAP 11787 is reviewed and also to formalize experience gained and actions taken from several previous small ($\leq$ 1 GPM) steam generator tube leaks.

A. If an air ejector or steam generator blowdown radiation monitor is above the normal alarm setpoint, radiochemistry analysis for that monitor shall be done at least once per 24 hours.

B. For greater than 0.01 GPM:

B.1 Radiation Protection Group shall sample the condenser air ejector, determine the leak rate, and report the results to the shift supervisor once per 8 hours, if the air ejector monitor is out-of-service.

B.2 The shift supervisor shall trend the leak rate at least once every 8 hours when the leak is greater than 0.01 GPM using reports provided by the Radiation Protection Group, or by the air ejector monitor readings.

C. For greater than 0.03 GPM:

C.1 Radiation Protection Group shall sample the air ejector, determine the leak rate, and report the results to the shift supervisor every 4 hours, if the aid ejector monitor is out-of-service.

C.2 The Shift Supervisor shall trend the leak rates every 4 hours using the Radiation Protection Group reports or the air ejector monitor readings.

D. For greater than 0.1 GPM:

Reduce power to less than 50% in less than 2 hours.

E. For greater than 0.3 GPM:

Shut down the unit in a manner appropriate to the rate of change of the leak rate.

Three methods of trending are available:

1. Emergency Response Computer System (On-line)

2. Chemistry History and Records Management System

3. Manual Plots

Leak rate determinations using radiochemistry are based on tritium in steam generator blowdown water for very low leaks (about 10^{-4} GPM) and Xenon at the air ejector for higher leak rates (about 10^{-3} GPM). Expected accuracy is about 20% for these methods. Two hours are allowed to decrease power from 100% to 50%.

The air ejector monitor alarm is normally set at 500 CPM with normal readings (no leakage) at less than 100 CPM. Radiation monitor readings are taken manually once per day and compared to the previous day's readings.

Please contact us if you have any questions related to the information we nave provided.

DM Musolf FOR
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Vice President, Nuclear Generation

c: Regional Administrator, Region III, NRC
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UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT DOCKET NO. 50-282
50-306

RESPONSE TO NRC COMPLIANCE BULLETIN 88-02

Northern States Power Company, a Minnesota corporation, with this letter is submitting information requested by NRC Compliance Bulletin 88-02.

This letter contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By DMMusolf
D M Musolf
Manager Nuclear Support Services

On this 24 day of March 1988 before me a notary public in and for said County, personally appeared D M Musolf, Manager Nuclear Support Services, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.

Judy L Klapperick

JUDY L. KLAPPERICK
NOTARY PUBLIC—MINNESOTA
ANOKA COUNTY
My Commission Expires Sept. 29, 1991