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Portland General Electric Company

David W. Cockfield Vice President, Nuclear

March 11, 1988

Trojan Nuclear Plant
Docket 50-344
License NPF-1

Mr. John B. Martin
Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek CA 94596-5368

Dear Mr. Martin:

Nuclear Regulatory Commission (NRC) Bulletin 88-02,
Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

On February 8, 1988, Portland General Electric Company (PGE) received NRC Bulletin 88-02 regarding the failure mechanism observed for the North Anna steam generator tube rupture event. The Bulletin is applicable to the Trojan Nuclear Plant which has Westinghouse Model 51 steam generators with carbon steel support plates. Similar to North Anna, an all-volatile water chemistry treatment is used at Trojan.

PGE has placed a high priority over the years on the maintenance of Trojan's steam generators. Some of the actions we have taken in this regard include being the first commercial plant to implement the Electric Power Research Institute Steam Generator Owner's Group guidelines for steam generator water chemistry, the removal of copper-bearing alloys from the Plant's secondary system, and an aggressive steam generator inservice inspection program. Our eddy-current inspections have indicated the presence of some very minor tube deformation from denting. This deformation is barely detected. Over the past several years, the denting in these tubes has not progressed nor have new dents been identified. By comparison, North Anna has experienced a much more severe denting problem than Trojan. We feel the aggressive actions we have taken to implement our commitment to maintain our steam generators reduces the likelihood of a North Anna-type tube rupture at Trojan. Nonetheless, action has been initiated in accordance with the Bulletin in order to further reduce the possibility of a similar event occurring at Trojan.

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In this regard, PGE has an enhanced primary-to-secondary leak rate monitoring program in place as an interim measure. This program was implemented on January 29, 1988 in the form of a night order to shift operating personnel. The program has since been incorporated into a Plant Operating Manual procedure. It is our intention to keep the program in place until the corrective actions identified by the detailed analyses have been implemented. A description of the program is provided in Attachment A.

In accordance with Paragraph A of the Bulletin, on February 19, 1988, PGE directed its steam generator eddy current contractor to review eddy current data for all four steam generators for tubes in Rows 8-12. These are the rows potentially susceptible to a North Anna-type failure. The purpose of this review is to identify the presence or absence of denting at the upper tube support plate and to map the anti-vibration bar installations. Attachment B describes the scope of the review and contains some preliminary data which we expect to be corroborated and further quantified by the formal review.

The Bulletin requires steam generator inspection data to be less than 40 months old. The data for "A", "B", and "C" steam generators meets this criteria. The eddy current data for the "D" steam generator is approximately 45 months old. The "D" steam generator is scheduled for eddy current examination during the 1988 refueling outage. The inspections will be performed between May 13 and May 21, 1988. The previously obtained inspection data will be initially evaluated and once the new inspection data are available, any differences will be factored into the detailed analyses.

PGE has initiated action to perform the detailed analyses described in Paragraph C.2 of the Bulletin. The analyses will include an assessment of stability ratios for the most limiting tube locations. The Bulletin calls for these analyses to be complete early enough to permit NRC review and approval prior to the restart from the next refueling outage if the next restart is greater than 90 days from receipt of the Bulletin. The next restart for the Trojan Nuclear Plant is 113 days from receipt of the Bulletin (May 31, 1988). Our current schedule calls for the detailed analyses to be complete by May 14, 1988. The "D" steam generator inspection data will be available on May 21, 1988 at the earliest. We then intend to submit this data to the contractor performing the analyses for resolution and the issuance of final analytical results and corrective action recommendations. After an internal review by PGE, a final report would be prepared and submitted to the NRC for review. We do not believe it is practical to perform these activities as well as receive NRC approval by May 31, 1988. Additionally, since the scheduled date for completion of all work in Containment is May 21, 1988, corrective actions could not be implemented in 1988 without extending the outage. Such an extension would cost $150,000 per day for replacement power.

Portland General Electric Company

PGE requests an extension to the schedule for responding to NRC Bulletin 88-02. We propose submitting the results of the steam generator tube analyses and our proposed corrective actions by July 31, 1988. This schedule allows us to evaluate the results of the analyses, develop corrective actions, and submit the response. If the analytical results demonstrate there are tubes which are susceptible to the North Anna-type failure, then the enhanced primary-to-secondary leak rate monitoring program will remain in place until corrective actions (ie, tube plugging, stabilization, etc) are implemented.

We request your review and approval of our proposed schedule. Your response to this request would be appreciated no later than March 31, 1988 in order for us to finalize our outage schedule.

Sincerely,

Attachments

c: U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk

Mr. William Dixon
State of Oregon
Department of Energy

Mr. R. C. Barr
NRC Resident Inspector
Trojan Nuclear Plant

Subscribed and sworn to before me this 11th day of March 1988.

Betty J. Simich
Notary Public of Oregon

My Commission Expires: 10-24-90

Betty J. Simich
BETTY J. SIMICH
NOTARY PUBLIC - OREGON
My Commission Expires 10-24-90

DESCRIPTION OF ENHANCED MONITORING PROGRAM FOR PRIMARY-TO-SECONDARY LEAKAGE

In order to predict a North Anna-type steam generator tube failure in time to take action prior to the tube actually rupturing, an enhanced primary-to-secondary leak rate monitoring procedure has been developed. The intent of the procedure is to identify primary-to-secondary leakage initially through sampling and/or installed instrumentation, accelerate the monitoring frequency once leakage is identified, trend the leak rate, and take appropriate action to arrest the increasing leak rate prior to a tube rupture.

Primary-to-secondary leakage has been observed previously at the Trojan Nuclear Plant due to U-bend cracking in tubes in Rows 1 and 2. Portland General Electric Company (PGE) has experience with detecting, characterizing, and trending such leakage.

PGE has developed an enhanced monitoring program to detect the onset of a North Anna-type tube rupture. Due to the possibility that further Row 2 (all Row 1 tubes are plugged) U-bend cracking could occur, the procedure has been developed to enable differentiation between a North Anna-type event and a U-bend crack. The nature of the U-bend crack leak rate is an initial rapid rise in leakage followed by a rapid decrease in the leak rate as the crack fills with corrosion products. The North Anna-type failure is characterized by a steadily increasing leak rate with a very rapid rise immediately prior to tube rupture. The procedure accounts for the differences in failure characteristics by requiring accelerated monitoring and basing corrective actions in the event of a North Anna-type failure on a leak rate increase of 50 gallons per day (gpd) over an 8-hour period. This criterion was selected based on the NRC Bulletin 88-02 leak rate versus time curve and will result in the initiation of action to prevent a tube rupture approximately 13 hours before a tube rupture would be expected. A description of the procedure is as follows:

Initial Leak Rate Identification

The primary-to-secondary leak rate is normally calculated weekly.

Process Radiation Monitors (PRM) 6 (condenser air ejector off-gas monitor), 10 (steam generator blowdown) and 16 (main steam line monitors) are set to alarm when the count rate exceeds 1.5 times the background level. For PRM 6, this would currently represent an alarm if the leak rate were to reach approximately 4 to 5 gpd. Off-Normal Instruction 12, "High Activity Radiation Monitoring", has been revised to require the notification of the on-shift Chemistry technician whenever PRM-6 alarms.

If PRM-6 is out-of-service, the leak rate will be calculated daily.

Accelerated Monitoring

If the leak rate increases to 10 gpd, the enhanced monitoring requirements are invoked in accordance with Chemistry Manual Procedure (CMP) 1, "Primary-to-Secondary Leak Rate". The leak rate will then be calculated at least daily. The frequency of leak rate calculations thereafter is dependent upon the leak rate itself and the rate of increase of the leak rate. If the leak rate reaches 50 gpd and is increasing, the leak rate is calculated at least every 8 hours until the leak rate stabilizes or it is determined power should be reduced or the Plant shut down. If the leak rate continues to increase and the rate of change of the leak rate reaches 50 gpd or greater in an 8-hour period, recommendations will be made to Plant management to reduce power or to shut the Plant down.

This monitoring scheme allows for a step increase and subsequent decrease or stabilization in the leak rate (characteristic of a U-bend crack) without initiation of a power reduction or Plant shutdown. If the leak rate shows a continuous increasing trend, then monitoring frequency increases enabling the identification of a North Anna-type failure. When the rate of increase of the leak rate reaches 50 gpd in 8 hours, corrective actions are recommended to Plant Management. This rate of increase corresponds to a time approximately 13 hours before a tube rupture would be expected as derived from the NRC Bulletin 88-02 leak rate curve.

STEAM GENERATOR
INSPECTION DATA REVIEW

A review of steam generator tube inspection data is being performed for all steam generators commencing with 1984 baseline data and proceeding to the most recent data. For the "D" steam generator, the 1984 data is the most current with the exception of 68 Row 2 tubes which were examined in 1986. Since the results of the review are not yet available, we will provide them with the July 31, 1988 submittal. The formal review of steam generator inspection data will include:

1. The identification and characterization of crevice corrosion at the upper tube support plate, emphasizing tubes in Rows 8-12.

2. The magnitude of steam generator tube deformation from denting, if any.

3. Mapping of anti-vibration bar locations.

4. Signal pictures (100 kHz and 400 kHz) for use in the detailed analyses.

From previous inspections, it is known there is some corrosion of the tube support plates at the tube-to-support plate intersections. Additionally, it is known that there are tubes which have experienced physical deformation due to denting. The denting, however, has been minor and has not affected tube integrity. From the results of the past several inspections, the denting does not appear to be progressing. The extent of the denting problems at the upper tube support plate for tubes in Rows 8-12 is being quantified as part of the formal data review.

Anti-vibration bars are present in all four steam generators and were originally specified to be inserted into Row 12. Preliminary mapping results show that in some cases the bars penetrate to Row 7. The table below indicates the number of tubes in Rows 8-12 in each steam generator which have anti-vibration bar support. As can be seen, anti-vibration bar support exists for a majority of the tubes in these rows, thus reducing Trojan's susceptibility to a North Anna-type tube rupture. This mapping will be verified during the formal data review.

	Steam Generator				
	A	B	C	D	Total Tubes in Row
Row 8	46	44	21	41	92
Row 9	68	86	56	79	92
Row 10	90	88	90	91	92
Row 11	91	92	89	92	92
Row 12	92	92	91	92	92

The results of the formal data review will be utilized in the detailed analyses and will be provided to the Nuclear Regulatory Commission in the July 31, 1988 report.

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