OMB No.: 31500011
NRCB 88-02

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 5, 1988

NRC BULLETIN NO. 88-02: RAPIDLY PROPAGATING FATIGUE CRACKS IN STEAM GENERATOR TUBES

Addressees:

For Action - All holders of operating licenses or construction permits for Westinghouse (W)-designed nuclear power reactors with steam generators having carbon steel support plates. Steam generators in this category include Westinghouse models 13, 27, 44, 51, D1, D2, D3 and D4, and the Westinghouse model E steam generators at South Texas Unit 1.

For Information - All other holders of operating licenses or construction permits for Westinghouse (W) and Combustion Engineering (CE) designed nuclear power reactors.

Purpose:

The purpose of this bulletin is to request that holders of operating licenses or construction permits for Westinghouse (W)-designed nuclear power reactors with steam generators having carbon steel support plates implement actions specified herein to minimize the potential for a steam generator tube rupture event caused by a rapidly propagating fatigue crack such as occurred at North Anna Unit 1 on July 15, 1987.

Description of Circumstances:

On July 15, 1987, a steam generator tube rupture event occurred at North Anna Unit 1 shortly after the unit reached 100% power. For several days prior to the event, operators had observed erratic air ejector radiation monitor readings. Grab samples were taken prior to the tube rupture for purposes of performing environmental release calculations. Subsequent analysis of this data indicated that increasing primary to secondary leakage had occurred over a 24- to 36-hour period before the tube rupture event. This leakage had been below the limit given in the Technical Specifications. The ruptured tube was located in Row 9 Column 51 in steam generator "C." The rupture location in this model 51 steam generator was at the top support plate on the cold leg side of the tube. The rupture extended circumferentially 360° around the tube.

The cause of the tube rupture has been determined to be high cycle fatigue. The source of the loads is believed to be a combination of a mean stress level in the tube and a superimposed alternating stress. (The mean stress is produced by denting of the tube at the uppermost tube support plate, and the alternating stress is the result of out-of-plane deflection of the U-bend portion of the

tube above the uppermost support plate, caused by flow-induced vibration.) Denting also shifts the maximum tube bending stress to the vicinity of the uppermost tube support plate. These loads are sufficient to produce fatigue in an all volatile treatment (AVT) water chemistry environment.

The specific mechanism for the flow-induced vibration has been determined to be a fluid-elastic instability. The fluid-elastic mechanism has a significant effect on tube response in cases where the fluid-elastic stability ratio equals or exceeds 1.0. The stability ratio, SR, is defined as:

$$SR = V_{eff}/V_c$$

where V_{eff} is the effective crossflow velocity and V_c is the critical velocity beyond which the displacement response to the tube increases rapidly.

The most significant contributors to the occurrence of a high fluid-elastic stability ratio are believed to have been (1) a reduction in damping at the tube-to-tube support plate intersection caused by denting and (2) locally high flow velocities caused by non-uniform antivibration bar (AVB) penetrations into the tube bundle. The presence of an AVB support will restrict tube motion and thus preclude the deflection amplitude required for fatigue. The original design configuration required AVBs to be inserted to Row 11. However, inspections have shown that some AVBs in the North Anna Unit 1 steam generators penetrate to Row 8, exceeding the minimum AVB design depth. However, no AVB support was present for the Row 9 Column 51 tube that ruptured.

Discussion:

Based on available information, the staff concludes that the presence of all the following conditions could lead to a rapidly propagating fatigue failure such as occurred at North Anna:

(1) denting at the upper support plate

(2) a fluid-elastic stability ratio approaching that for the tube that ruptured at North Anna

(3) absence of effective AVB support

Actions Requested:

Within 45 days following receipt of this bulletin, addressees having Westinghouse steam generators with carbon steel support plates shall submit a written report detailing the status of their compliance with the actions specified below for purposes of minimizing the potential for rapidly propagating fatigue failure such as occurred at North Anna 1. The report shall include an appropriate schedule for completion of the analyses described under item C below, if applicable.

A. The most recent steam generator inspection data should be reviewed for evidence of denting at the uppermost tube support plate. Inspection records may be considered adequate for this purpose if at least 3% of the total steam generator tube population was inspected at the uppermost support plate elevation during the last 40 calendar months. "Denting" should be considered to include evidence of upper support plate corrosion and the presence of magnetite in the tube-to-support plate crevices, regardless of whether there is detectable distortion of the tubes. The results of this review shall be included as part of the 45-day report. Where inspection records are not adequate for this purpose, inspections of at least 3% of the total steam generator tube population at the uppermost support plate elevation should be performed at the next refueling outage. The schedule for these inspections shall be included as part of the 45-day report and the results of the inspections shall be submitted within 45 days of their completion. Pending completion of these inspections, an enhanced primary-to-secondary leak rate monitoring program should be implemented in accordance with paragraph C.1. below.

B. For plants where no denting is found at the uppermost support plate, the results of future steam generator tube inspections should be reviewed for evidence of denting at the uppermost support plate. If denting is found in the future, the provisions of item C below should be implemented. Commitments to implement these actions shall be submitted when the results of A above are submitted.

C. For plants where denting is found, the NRC staff requests that the following actions be taken:

1. Pending completion of the NRC staff review and approval of the program described in C.2 below or completion of inspections specified in item A above to confirm that denting does not exist, an enhanced primary-to-secondary leak rate monitoring program should be implemented as an interim compensatory measure within 45 days of the date of receipt of this bulletin.* Implementation of this program shall be documented as part of the 45-day report. The enhanced monitoring program is intended to ensure that if a rapidly propagating fatigue crack occurs under flow-induced vibration, the plant power level would be reduced to 50% power or less at least 5 hours before a tube rupture was predicted to occur. The effectiveness of this program should be evaluated against the assumed time-dependent leakage curve given in Figure 1.

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*While this bulletin was being prepared, licensees for a few plants committed to an enhanced primary-to-secondary leak rate monitoring program at the staff's request. These plants had been identified on a preliminary basis by Westinghouse as being potentially susceptible to rapidly propagating fatigue cracks. These enhanced programs should be upgraded as necessary to comply with this paragraph. However, no relaxation from current commitments should be made without prior approval by the NRC staff.

This program should consider and provide the necessary leakage measurement and trending methods, time intervals between measurements, alarms and alarm setpoints, intermediate actions based on leak rates or receipt of alarms, administrative limits for commencing plant shutdown, and time limitations for (1) reducing power to less than 50% and (2) shutting down to cold shutdown. Appropriate allowances for instrument errors should be considered. Finally, the program should make provision for out of service radiation monitors, including action statements and compensatory measures.

2. A program should be implemented to minimize the probability of a rapidly propagating fatigue failure such as occurred at North Anna Unit 1. The need for long-term corrective actions (e.g., preventive plugging and stabilization of potentially susceptible tubes, hardware, and/or operational changes to reduce stability ratios) and/or long-term compensatory measures (e.g., enhanced leak rate monitoring program) should be assessed and implemented as necessary. An appropriate program would include detailed analyses, as described in subparagraphs (a) and (b) below, to assess the potential for such a failure. Alternative approaches and/or compensatory measures implemented in lieu of the actions in subparagraphs (a) or (b) below should be justified.

   Although the 45-day report shall provide a clear indication of actions proposed by licensees, including their status and schedule, a detailed description of this program and the results of analyses shall be submitted subsequently, but early enough to permit NRC staff review and approval prior to the next scheduled restart from a refueling outage. Where the next such restart is scheduled to take place within 90 days, staff review and approval will not be necessary prior to restart from the current refueling outage. An acceptable schedule for submittal of the above information should be arranged with the NRC plant project manager by all licensees to ensure that the staff will have adequate time and resources to complete its review without adverse impact on the licensee's schedule for restart.

   (a) The analysis would include an assessment of stability ratios (including flow peaking effects) for the most limiting tube locations to assess the potential for rapidly propagating fatigue cracks. This assessment would be conducted such that the stability ratios are directly comparable to that for the tube which ruptured at North Anna.

   (b) The analysis would include an assessment of the depth of penetration of each AVB. The purpose of this assessment is twofold: (1) to establish which tubes are not effectively supported by AVBs and (2) to permit an assessment of flow peaking factors.

(Note: Most steam generators have at least two sets of AVBs. This applies only to the set that penetrates most deeply into the tube bundle.) The methodology used to determine the depth of penetration of each individual AVB shall be described in detail in the written report. The criteria for determining whether a tube is effectively supported by an AVB shall also be identified. (Note: An AVB that penetrates far enough to produce an eddy current signal in a given tube may not penetrate far enough to provide a fully effective lateral support to that tube.)

If addressees cannot perform this suggested approach or meet this suggested schedule, they should justify to the NRC their alternative approaches and schedules.

The written reports shall be submitted to the appropriate Regional Administrator under oath or affirmation under provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, the original of the cover letter and a copy of the reports shall be transmitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555 for reproduction and distribution.

This request for information was approved by the Office of Management and Budget under blanket clearance number 31500011. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C. 20503.

The NRC intends to review the information collected under this bulletin and determine the adequacy of specific actions proposed by each licensee. The information will be analyzed and placed in the NRC Public Document Rooms.

If you have any questions about this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate regional office.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Emmett Murphy, NRR
(301) 492-0945

Keith Wichman, NRR
(301) 492-0908

Attachments:
1. Figure 1 Leak Rate Versus Time Chart
2. List of Recently Issued NRC Bulletins

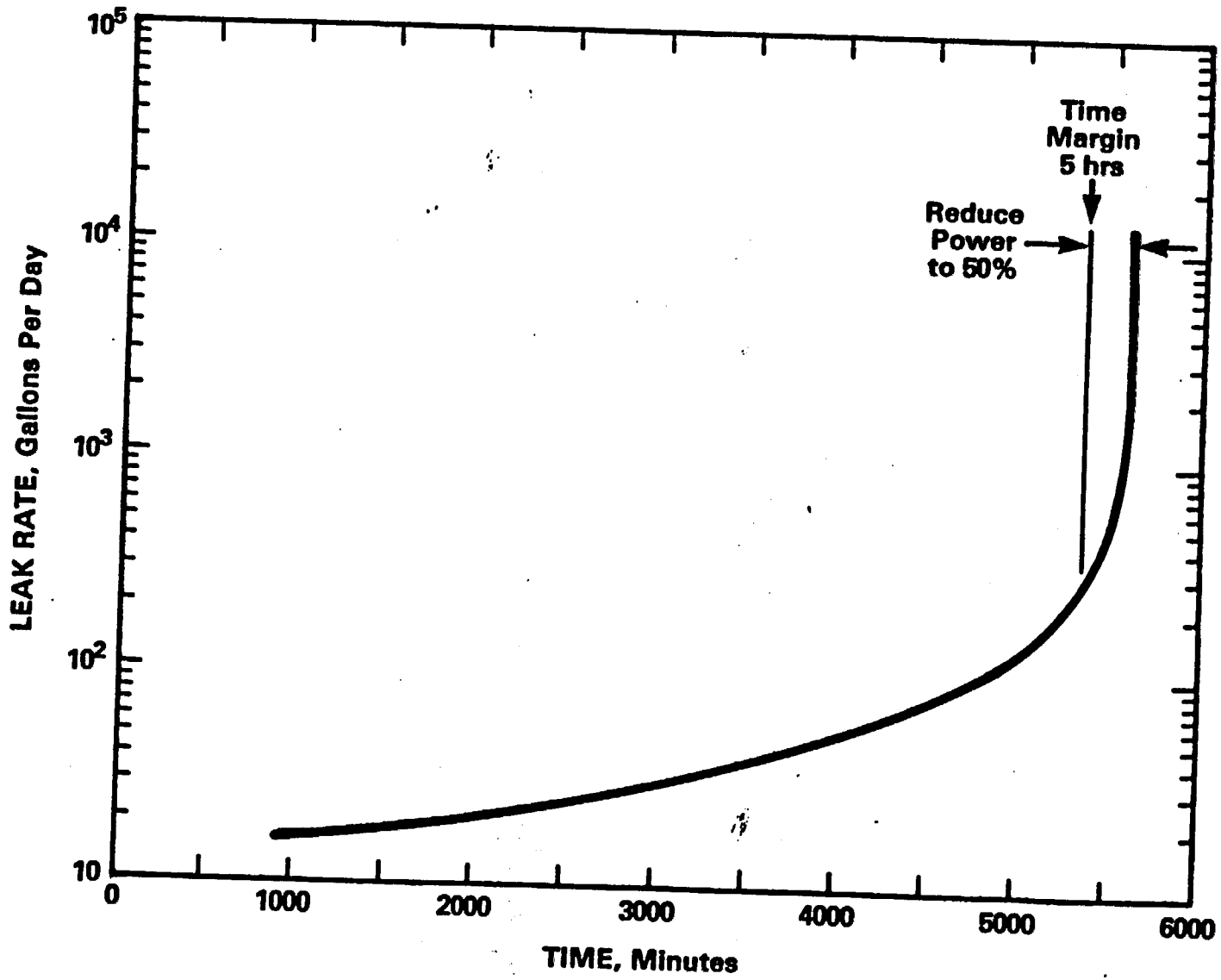

Figure 1 LEAK RATE VERSUS TIME

Attachment 1

Attachment 2
NRCB 88-02
February 5, 1988

LIST OF RECENTLY ISSUED
NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
88-01	Defects in Westinghouse Circuit Breakers	2/5/88	All holders of OLs or CPs for nuclear power reactors.
87-02	Fastener Testing to Determine Conformance with Applicable Material Specifications	11/6/87	All holders of OLs or CPs for nuclear power reactors.
87-01	Thinning of Pipe Walls in Nuclear Power Plants	7/9/87	All licensees for nuclear power plants holding an OL or CP.
86-04	Defective Teletherapy Timer That May Not Terminate Dose	10/29/86	All NRC licensees authorized to use cobalt-60 teletherapy units.
86-03	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	10/8/86	All facilities holding an OL or CP.
86-02	Static "O" Ring Differential Pressure Switches	7/18/86	All power reactor facilities holding an OL or CP.
86-01	Minimum Flow Logic Problems That Could Disable RHR Pumps	5/23/86	All GE BWR facilities holding an OL or CP.
85-03	Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings	11/15/85	All power reactor facilities holding an OL or CP.

OL = Operating License
CP = Construction Permit

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1. Figure 1 Leak Rate Versus Time Chart
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*SEE PREVIOUS CONCURRENCES

		D/DOEA:NRR	*C/OGCB:DOEA:NRR	*EMTB:DEST:NRR
		CERossi	CHBerlinger	CYCheng
		2/2/88	1/05/88	12/08/87
*OGCB:DOEA:NRR	*EMTB:DEST:NRR	*EMTB:DEST:NRR	*EAD/DEST:NRR	*PPMB:ARM
SDMacKay	EMurphy	KWichman	JRichardson	TechEd
12/08/87	12/08/87	12/08/87	12/08/87	12/9/87

Document Name:
NRCB 88-02

Requestor's ID:
KING

Author's Name:
E. Murphy/Sam MacKay

Document Comments:
D9 NRC Bulletin NO. 87-XX

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12/8 /87	12/ /87	12/ /87	12/ /87	12/ /87