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

July 5, 1991

NRC INFORMATION NOTICE NO. 91-43: RECENT INCIDENTS INVOLVING RAPID INCREASES IN PRIMARY-TO-SECONDARY LEAK RATE

Addressees:

All holders of operating licenses or construction permits for pressurized-water reactors (PWRs).

Purpose:

This information notice is intended to inform addressees of recent incidents involving very rapid increases in the primary-to-secondary leak rate. One of these incidents was followed by a steam generator tube rupture (SGTR). The leakage during these incidents increased at rates that were significantly higher than would be predicted on the basis of Figure 1 of Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to minimize the probability of SGTR events. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Mihama Unit 2 (Japan)

Mihama Unit 2 is a 19-year old PWR built by Mitsubishi and based on a Westinghouse Electric Corporation two-loop design. The Mihama Unit 2 steam generators (SGs) are based on the Westinghouse Model 44 design. At 12:24 hours on February 9, 1991, plant personnel received an "attention" signal from the SG blowdown monitor (R-19). The attention signal setpoint was at 60 counts per minute (cpm), compared to the normal reading of 35 cpm. At 12:33 hours plant personnel received an attention signal from the air ejector monitor (R-15). The attention signal setpoint for the R-15 was at 900 cpm, compared to a normal reading of 800 cpm. At 13:00 hours, plant personnel sampled the blowdown from SGs A and B. Results were obtained at 13:20 hours indicating a radioactivity concentration only slightly higher than normal in SG A, and no detectable concentration in SG B.

At 13:40 hours, an R-15 "counting rate alarm" (alarm setpoint: 2000 cpm) was sounded. At 13:45 hours, the R-19 counting rate alarm (alarm setpoint: 400 cpm) was sounded. At this time, plant personnel manually started a third charging pump because of decreased pressure and water level in the pressurizer. At 13:48 hours, personnel began to manually reduce reactor power at a rate of 4.2 percent per minute. At 13:50 hours, the R-15 "counting rate high" alarm (alarm

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setpoint: 1 X 10^6 cpm) was sounded, followed by reactor trip on "low pressurizer water level," turbine trip, generator trip, and actuation of safety injection on low pressure and low water level in the pressurizer. Leakage from the primary to the secondary was essentially terminated at 14:48 hours. Plant personnel brought the plant to cold shutdown at 02:30 hours on February 10, 1991.

Following this SGTR event*, the utility investigated the rupture and found that it was a complete circumferential failure of tube R14C45 in SG A, at the uppermost support plate. The utility found that the failure mechanism was high cycle fatigue caused by fluid-elastic vibration. By design, all tubes in rows 11 and greater are supposed to be supported by anti-vibration bars (AVBs). However, the subject tube was not found to be so supported because of a reported "incorrect insertion" of the adjacent AVBs.

Maine Yankee Atomic Power Station

Maine Yankee is a PWR designed by Combustion Engineering, Incorporated, and was licensed in 1973. Between 14:00 hours on December 12, 1990, and 00:57 hours on December 17, 1990, the rate of primary-to-secondary leakage gradually increased from 0.0006 gallons per minute (gpm) to 0.008 gpm as determined from grab samples from the condenser air ejector. During this period, the licensee took grab samples at approximately 4-hour intervals. The licensee analyzed a grab sample taken at 02:34 hours on December 17, 1990, and found that leakage in SG 1 had jumped to 0.017 gpm, with a corresponding reading of 75,000 cpm on the air ejector radiation monitor. At 03:40 hours, the licensee began reducing power at a rate of 5 percent per hour. At 04:50 hours, the radiation monitor reading increased from 75,000 to over 400,000 cpm in less than 1 minute. Using this and the previous leak rate, the licensee quickly estimated a leak rate of 0.11 gpm. This estimate exceeded the licensee's administrative limit of 0.07 gpm, and the licensee increased the rate of power reduction to 50 percent per hour. The licensee analyzed a grab sample taken at 05:21 hours and confirmed a leakage rate of 0.105 gpm. At 06:07 hours, the reading from the air ejector radiation monitor jumped to 600,000 cpm. Based on a grab sample taken at 06:36 hours, the calculated leak rate was 1.4 gpm, which exceeded the technical specification leak rate limit of 0.15 gpm. The plant reached hot standby status at 06:53 hours and cold shutdown status at 03:10 hours on December 18, 1990.

Subsequent investigation established the source of the leak to be a 4-inch long axial crack at the apex of the row 6, line 43 U-bend. The licensee described this location as a "steam blanketed region" where the batwing supports restrict flow permitting a steam void to form and contaminants to be deposited on the tube surface. On the basis of the size of the crack found, the staff believes that the early identification and response to the rapidly increasing leak rate was the key factor in averting an SGTR event before plant shutdown.

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*An SGTR event is defined in this information notice as a primary-to-secondary leak exceeding the normal charging pump capacity of the primary system.

Three Mile Island Unit 1 (TMI-1)

Three Mile Island Unit 1 is a PWR designed by Babcock and Wilcox (B&W) with once-through type steam generators and was licensed in 1974. On March 6, 1990, TMI-1 was operating at 75 percent steady power, in a 30-hour hold for restart physics testing. The licensee was first alerted to the onset of primary-to-secondary leakage by an alarm from the condenser air ejector radiation monitors at 08:23 hours. A post-incident review of the radiation monitor data indicates that the activity actually began to increase above its normal steady value at 08:01 hours. At around 08:50 hours, the radiation monitor reading had increased from its initial value of 50 cpm to about 50,000 cpm, 5 times the alarm setpoint. At 09:00 hours, the licensee's preliminary estimates of leak rate were 0.5 gpm based on mass balance estimates for the reactor coolant system (RCS) and between 0.5 and 0.75 gpm based on the decreased level in the make-up tank. These estimates were below the plant technical specification limit of 1.0 gpm. At 09:12 hours, a plant shutdown was commenced at the rate of 2 percent per minute, and the radiation monitor readings began to decrease. The plant reached hot shutdown status at 10:42 hours. On March 7, 1990, the plant reached cold shutdown status by 07:30 hours.

The licensee later determined from activity measurements that leakage had reached 1.1-1.8 gpm before plant shutdown was commenced. However, approximately 2.5-3 hours are needed to obtain results from this method, and, thus, this information was not available to the operators prior to the decision to commence plant shutdown.

After shutting down the plant, the licensee found the source of the leak to be a 360 degree circumferential crack in tube 1 of row 77 in the "lane-wedge" region at the lower face of the upper tubesheet. The staff believes that the early identification and response to the rapidly increasing leak rate was key to preventing a much larger leak (if not an SGTR event) before plant shutdown. Similar fatigue cracks in other B&W steam generators have caused larger leaks, but not SGTRs, because these cracks were confined within the tubesheets or support plates.

Discussion:

Earlier incidents involving rapidly increasing primary-to-secondary leak rates were the subject of NRC Bulletin 88-02 and NRC Information Notice 88-99, "Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage." Bulletin 88-02 was issued in response to the July 15, 1987, SGTR event at the North Anna Power Station Unit 1. The NRC staff requested, in item C.1 of Bulletin 88-02, that an enhanced primary-to-secondary leak rate monitoring program be implemented at certain PWRs (i.e., PWRs with Westinghouse steam generators, carbon steel support plates, and denting corrosion) until a potential fatigue problem at these plants could be resolved. The staff requested this enhanced leak rate monitoring program to ensure that licensees could detect and respond to a rapidly increasing leak rate caused by high cycle fatigue before an SGTR occurs. The effectiveness of this program was to be evaluated against the assumed time-dependent leak rate curve given in Figure 1 of the bulletin. This curve was based on the estimated rate of increase in leakage before the SGTR event at North Anna Unit 1. This curve yields an estimated 63 hours for leak rates to increase from 20 gallons per day (gpd) to 500 gpd.

During discussions with numerous industry representatives, the staff has found that Figure 1 of Bulletin 88-02 is widely used throughout the industry as a benchmark and/or performance measure for plant-specific leak rate monitoring programs. This is true even at plants which were not subject to the actions requested by the bulletin.

However, there have been several incidents since issuance of Bulletin 88-02 where the rate of leakage increase occurred more rapidly than would be predicted on the basis of Figure 1 of the bulletin. The leak rates at Mihama Unit 2, Maine Yankee, and Three Mile Island Unit 1 escalated from very low levels to more than 500 gpd over time periods ranging between one hour and six hours. An earlier leakage incident at Indian Point Unit 3, described in NRC Information Notice 88-99, developed over a similar time span. Thus, these incidents are indicative of the limitations of Figure 1 of the bulletin as a benchmark and/or performance measure for plant-specific leak rate monitoring programs.

Leak rate monitoring programs can provide for early detection and response to rapidly increasing leak rates and, thus, can be an effective approach for minimizing the frequency of steam generator tube ruptures. This can be achieved by having, as close as possible, real time information on leak rate and rate of increase of leak rate on which to act. Data from the air ejector radiation monitors, for example, are displayed continuously in the control room and have been shown to provide a relatively good time response to rapidly increasing leakage. Use of these data, in conjunction with appropriate alarm setpoints, can quickly alert the operators to a rapid increase in leak rate and the need for confirmatory leakage measurements and/or the need to shut down the plant.

Nitrogen-16 (N-16) monitors on the steamlines are coming into increasing use in the U.S. industry as a supplemental method for monitoring primary-to-secondary leakage. These monitors also exhibit good time response to changes in the leakage rate. Data from the N-16 monitors can be continuously displayed in the control room directly in terms of leakage rate and can be alarmed.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the technical contact listed below or the appropriate NRR project manager.

Charles E. Rossi

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Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contact: E. Murphy, NRR
(301) 492-0710

Attachment: List of Recently Issued Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
91-42	Plant Outage Events Involving Poor Coordination Between Operations and Maintenance Personnel During Valve Testing and Manipulations	06/27/91	All holders of OLs or CPs for nuclear power reactors.
91-41	Potential Problems with The Use of Freeze Seals	06/27/91	All holders of OLs or CPs for nuclear power reactors.
88-63, Supp. 2	High Radiation Hazards from Irradiated Incore Detectors and Cables	06/25/91	All holders of OLs or CPs for nuclear power reactors, research reactors, and test reactors.
91-40	Contamination of Non-radioactive System and Resulting Possibility for Unmonitored, Uncontrolled Release to the Environment	06/19/91	All holders of OLs or CPs for nuclear power reactors.
91-39	Compliance with 10 CFR Part 21, "Reporting of Defects and Noncompliance"	06/17/91	All Nuclear Regulatory Commission (NRC) material licensees.
91-38	Thermal Stratification in Feedwater System Piping	06/13/91	All holders of OLs or CPs for nuclear power reactors.
91-37	Compressed Gas Cylinder Missile Hazards	06/10/91	All holders of OLs or CPs for nuclear power reactors.
91-36	Nuclear Plant Staff Working Hours	06/10/91	All holders of OLs or CPs for nuclear power reactors.
91-35	Labeling Requirements for Transporting Multi-Hazard Radioactive Materials	06/07/91	All U.S. Nuclear Regulatory Commission (NRC) licensees.

OL = Operating License
CP = Construction Permit

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6/18/91	6/18/91	6/19/91	6/20/91	6/21/91	06/25/91	06/28/91

DOCUMENT NAME: IN 91-43

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OK for CHB subject to concerns over taking a position

*see previous concurrence

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