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

July 21, 1993

NRC INFORMATION NOTICE 93-55: POTENTIAL PROBLEM WITH MAIN STEAMLINE BREAK ANALYSIS FOR MAIN STEAM VAULTS/TUNNELS

Addressees

All holders of operating licenses or construction permits for pressurized water reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to a potential inadequacy in the main steamline break analysis which could place some pressurized-water reactor (PWR) plants outside their current structural design basis for the main steam valve vaults or main steam tunnels. The plants of concern are those that must postulate a double-ended rupture of a main steamline in these areas. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

During the Watts Bar Calculation Reconstitution Program, Tennessee Valley Authority (TVA) discovered that Westinghouse had supplied data for the main steamline break analysis which assumed perfect moisture separation. TVA used this data to determine the mass and energy released into the main steam valve vault.

TVA had requested Westinghouse to reevaluate the 1975 Westinghouse mass and energy release data used in the Watts Bar analysis for these valve vaults, and to advise TVA if the data were still applicable. On June 23, 1992, Westinghouse advised TVA that the 1975 mass and energy release data were not considered conservative, and were not applicable for a pressure transient evaluation of the vented main steam valve vaults. The mass and energy release did not account for liquid entrainment in the blowdown, and resulted in a slower mass and energy release rate in the 1975 data.

Westinghouse then provided a bounding analysis based on ANSI/ANS Standard 58.2 (1980) methodology which included liquid entrainment in the blowdown. This new analysis indicated that the valve vault structural design pressure would be exceeded in the event of a steamline break within the vaults.

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On August 17, 1992, for Sequoyah Nuclear Plant (LER 50-327/92-013), and on October 13, 1992, for Watts Bar Nuclear Plant (CDR 50-390/92-09), TVA reported the use of nonconservative Westinghouse data for the main steamline break analysis which could result in the valve vault structural design pressure being exceeded. TVA had determined that the mass and energy release data for Watts Bar were also applicable to Sequoyah. TVA prepared a justification for continued operation (JCO) for Sequoyah. This JCO will be in effect until the startup from refueling Cycle 6 for both Sequoyah Unit 1 (Fall 1993) and Unit 2 (early 1994).

DISCUSSION

The 1975 mass and energy release data supplied by Westinghouse was based upon the largest steam generator depressurization rate consistent with a high-quality steam discharge. The Westinghouse data was applicable for a postulated double-ended main steamline break in the turbine building, assuming flow in both the forward and reverse direction. Apparently, TVA applied this data without verifying its applicability to vented compartments, such as the main steam valve vaults. A dry steam release in a vented compartment such as the main steam valve vault may not be conservative, because of moisture entrainment within the discharge.

For a main steamline break analysis, the limiting plant conditions for the steam generator mass inventory and secondary system pressure are often at hot standby/shutdown plant conditions (0 percent power level, primary plant at operating temperature and pressure). Due to the high flow rates associated with the main steamline break, frothing in the steam generator raises the water level rapidly, which decreases the quality of fluid expelled from the steam generator. Although the enthalpy of this low-quality fluid is less than the enthalpy of dry steam, the critical mass flow is 4 to 5 times higher, resulting in a net increase in the energy release rate from the break. This may be the limiting case for determining maximum pressure in vented (blowout panels) compartments.

Westinghouse recommended to TVA that the methodology outlined in ANSI/ANS Standard 58.2 (1980), Appendix E be used to generate the Watts Bar bounding mass and energy release rates, which would determine the pressure inside the valve vaults. This mass and energy release data would include the entrainment of water and bound the analyses that could be conducted for this type of event. Westinghouse performed the Watts Bar analysis with the ANSI/ANS 58.2 methodology and informed TVA of a significant increase in the mass and energy releases generated over those of the original analysis.

TVA determined that the increased Watts Bar mass and energy release rates produced pressures that exceeded the present structural design margins, and challenged the structural adequacy of the walls and slabs of the main steam valve vaults. TVA calculations showed that the peak pressures in the valve vaults could increase by about one-third when moisture entrainment was considered. Failure of the valve vault walls or slabs could damage such equipment as main steam system, main feedwater system, and auxiliary feedwater system components and piping. This equipment damage could result in the inability (or reduced ability) to feed the intact steam generators, or in the blowdown of more than one steam generator.

Upon consultation with Westinghouse, TVA determined that the analysis data for the Sequoyah main steam valve vault rooms were also nonconservative. A JCO has been prepared for Sequoyah. The JCO is based on the Sequoyah main steam system piping design in the valve vaults meeting most of the break exclusion provisions of the Standard Review Plan (SRP) Branch Technical Position (BTP) MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." A postulated one-square-foot break was analyzed for the JCO interim period. The revised calculated pressures (using the ANSI/ANS 58.2 methodology) were bounded by the original design pressure of the vaults. This JCO will be in effect until the next Sequoyah, Units 1 and 2 refueling outages (Cycle 6 for both units). TVA will make plant modifications to bring the plant into compliance with the original design basis. The modifications will involve modifying each of the fluid head anchor-sleeve openings to decrease the flow area in the event of a postulated break, thereby limiting the mass and energy release rate into the valve vaults. The flow area will be sized to limit the pressure in the main steam valve vaults to less than the original design basis of the floor and walls.

Combustion Engineering and Babcock & Wilcox designed PWRs may also be affected by this issue if vented compartments have been analyzed nonconservatively, assuming dry steam. Therefore, this information notice is being sent to all PWR licensees and holders of PWR construction permits.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation project manager.

Brian Grimes

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contacts: J. B. Brady, RII
(404) 331-0339

W. T. Lefave, NRR
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Information Notice No.	Subject	Date of Issuance	Issued to
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93-53	Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned	07/20/93	All holders of OLs or CPs for nuclear power reactors.
93-52	Draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes"	07/14/93	All holders of OLs or CPs for pressurized water reactor (PWRs).
93-51	Repetitive Overspeed Tripping of Turbine-Driven Auxiliary Feed-water Pumps	07/09/93	All holders of OLs or CPs for nuclear power reactors.
93-50	Extended Storage of Sealed Sources	07/08/93	All licensees authorized to possess sealed sources.
93-49	Improper Integration of Software into Operating Practices	07/08/93	All holders of OLs or CPs for nuclear power reactors.
93-48	Failure of Turbine-Driven Main Feedwater Pump to Trip Because of Contaminated Oil	7/6/93	All holders of OLs or CPs for nuclear power reactors.
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orig /s/'d by BKGrimes

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Combustion Engineering and Babcock & Wilcox designed PWRs may also be affected by this issue. The pressure effects on their vented compartments may have been analyzed nonconservatively, assuming dry steam. Therefore, this information notice is being sent to all PWR licensees and holders of PWR construction permits.

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release rate. The flow area will be sized to limit the pressure in the MSVVs to less than the original design basis of the floor and walls.

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the design basis for the Sequoyah / Watts Bar main steam valve vault rooms, and is evaluating the modifications that may by required.

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