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September 3, 1982

Peter B. Bloch, Chairman
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Jerry R. Kline
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Hugh C. Paxton
Administrative Judge
1229 - 41st Street
Los Alamos, New Mexico 87544

In the Matter of
WISCONSIN ELECTRIC POWER COMPANY
(Point Beach Nuclear Plant, Units 1 and 2)
Docket Nos. 50-266 and 50-301
(Repair to Steam Generator Tube)

Dear Administrative Judges:

Approximately a week ago, Judge Bloch telephoned me to request information concerning the 1981 NRC Annual Report, specifically, the section entitled "PWR Steam Generator Tube Integrity" on pages 16 and 17. (enclosed) The section states that a NUREG concerning Unresolved Safety Issues, Tasks A-3, 4 and 5, relating to steam generator tube degradation was soon to be published.

After consulting with the NRC Project Manager in the proceeding, I telephoned Judge Bloch on August 19, 1982. I informed him that the section in the Annual Report had been superseded. USI Tasks A-3, 4 and 5 have been "folded into" the proposed requirements program described in my August 12, 1982 letter to this Board.

Sincerely,

Richard G. Bachmann
Counsel for NRC Staff

Enclosure as stated

cc (w/ encl.):
Service List

OFC	:OELD	:OELD	:	:	:	:	:
NAME	:Bachman:sl	:STreby	:	:	:	:	:
DATE	:8/19/82	:8/30/82	:	:	:	:	:

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A summary of the status of Unresolved Safety Issues is presented quarterly in NUREG-0606. Other generic safety and environmental issues are covered in the Generic Issues Tracking Systems, except that TMI Action Plan items are treated separately in an Action Plan Tracking System.

PROGRESS REPORTS

Given below are progress reports on each of the Unresolved Safety Issues under active consideration. For background on earlier phases of some of these issues, see the *1980 NRC Annual Report*, pp. 45-57.

Water Hammer

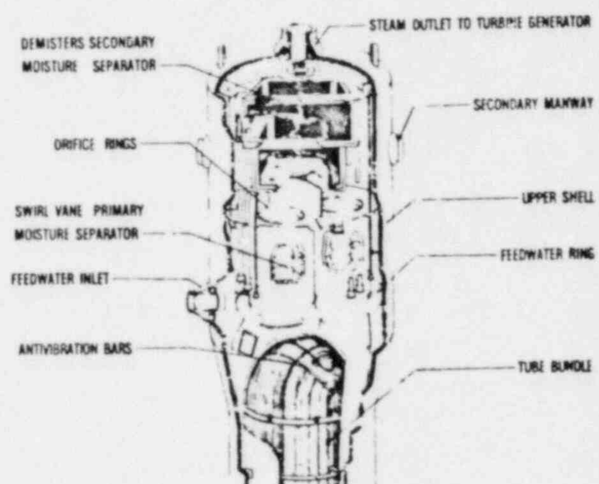
Water hammer events are high pressure pulses experienced by fluid systems. Water hammers can be induced by phenomena such as rapid valve closures, steam condensation or pump startup into empty lines. Commonly experienced water hammer phenomena are pipe rattle when water faucets are rapidly closed and steam heating system thumping from steam condensation effects. Water hammer is commonly experienced in chemical process industries and power plant piping which carries steam or water. Most water hammers are attributed to rapid condensation of steam, steam-driven slugs of water, pump startup into empty lines and operations which result in rapid valve closure. Since 1968, almost 150 water hammer events in nuclear power reactors have been reported. None of these has resulted in any release of radioactivity external to the plant, and for the great majority of events, damage has been confined to pipe supports and snubbers. The principal concern of this safety issue is the rather low probability that a water hammer event would result in failure of the reactor coolant system or would disable safety systems or a system which is needed for safe reactor shutdown and cooling following an initiating accident or malfunction of a different system or component.

The work on this task has been directed at the analysis of water hammer in several specific systems, including steam generator feedwater systems, and several technical reports have been issued summarizing this work. In 1981, Task A-1 was reassessed and a new resolution plan developed which consists of a comprehensive review of fluid systems design and a review of system operating procedures. Design factors and operational procedures which can result in system conditions which are conducive to water hammer events will be identified. As a result of this review, specific recommendations will be developed to reduce the number of water hammers and to minimize the severity of water hammer events. Completion of Task A-1 with publication of the final report is scheduled for January 1983.

PWR Steam Generator Tube Integrity

In plants employing pressurized water reactors, the primary coolant is kept under pressure sufficient to prevent boiling. This high-pressure water passes through tubes around which water circulates in a secondary system to produce steam to drive the turbine generator. The assembly in which the heat transfer takes place and steam is produced is the steam generator. The tubes within the steam generator are an integral part of the primary coolant boundary, keeping the radioactive primary coolant in a closed system isolated from the environment. Maintenance of steam generator tube integrity is a primary concern, both during normal operation or during an accident. Discussions of specific problems associated with steam generator tube integrity occurring at operating reactors were provided in two reports: "Operating Experience with Recirculation Steam Generators" (NUREG-0523, January 1979) and "Operating Experience with Once Through Steam Generators" (NUREG-0571, March 1980).

In order to assure steam generator tube integrity plant technical specifications require routine inservic



SERIES 51 STEAM GENERATOR

inspection of steam generators to be performed every 12 to 24 months. On plants where steam generator tubes have been extensively degraded, the NRC has imposed license conditions to increase the required frequency of inspection and to have severely damaged tubes removed from service. The conditions also require that, following inspection of steam generators and completion of any necessary repair programs by the licensees, the NRC must approve or concur in the restart of the facility. Safe operation is assured by the imposition of strict operating conditions, including the plugging of affected tubes and restricting allowable leak rates during normal operation.

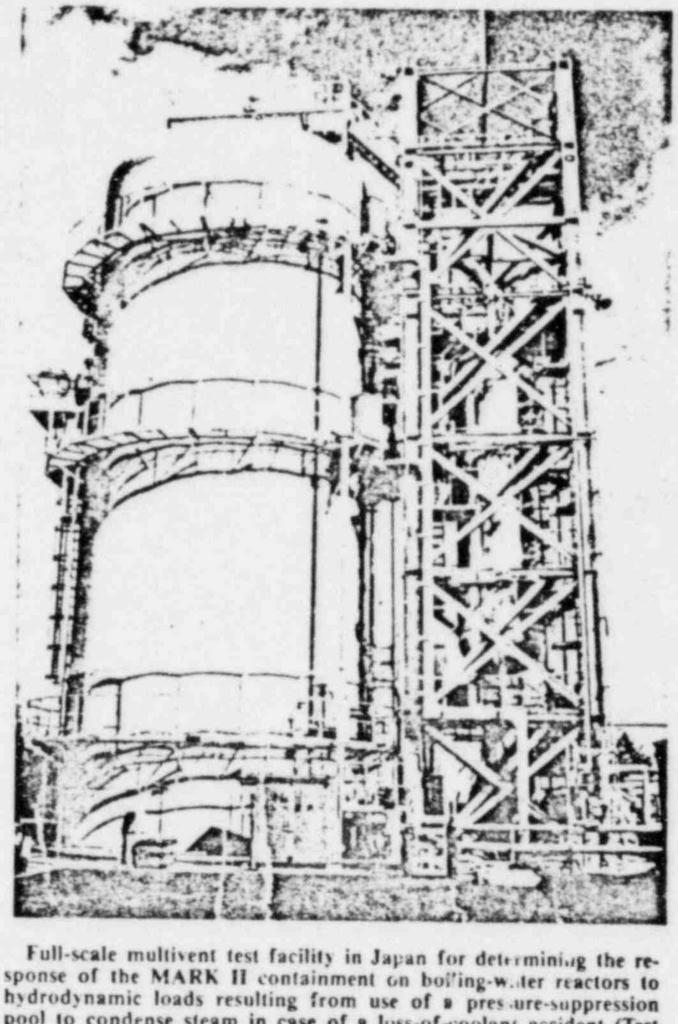
USI Tasks A-3, 4 and 5 were established to address tube degradation problems that have arisen in Westinghouse, Combustion Engineering and Babcock & Wilcox steam generators. A NUREG report presenting the results of the Generic Tasks was prepared and is expected to be published for public comment. The report presented an update of operating experiences and the results of technical studies in the areas of systems analyses, inservice inspection and tube integrity. Based on review of operating experience and results of the technical studies, the report establishes either the adequacy of existing criteria or improved criteria for ensuring safe and reliable steam generator operation. The new criteria will be implemented following incorporation of appropriate public comments. Implementation strategy and impact of new requirements also are discussed in NUREG-0844.

Steam generator tube degradation already occurring in operating plants will be difficult to completely arrest and some degradation is likely to continue to occur. Implementation of the requirements developed in the Generic Tasks A-3, A-4 and A-5 will not bring an end to steam generator tube degradation but will ensure safe steam generator operation with improved reliability.

(See discussion under "Steam Generators," later in this chapter.)

BWR MARK I and MARK II Pressure Suppression Containments

Boiling water reactor (BWR) pressure-suppression containment systems, designed by the General Electric Company are engineered to utilize a large mass of water (suppression pool) as a heat sink which will condense the steam and absorb the energy released from the reactor primary system in the event of postulated accidents or transients. The absorption of excessive energy by the stored water reduces the pressure in the containment and that, in turn, reduces the driving force that might lead to a release of fission products to the environment that may have escaped into the containment building from the primary system.



Full-scale multivent test facility in Japan for determining the response of the MARK II containment on boiling-water reactors to hydrodynamic loads resulting from use of a pressure-suppression pool to condense steam in case of a loss-of-coolant accident (Test A-8). At the left is a mockup of the containment, and at the right is a source of steam for use in these tests.

During the course of large-scale testing for an advanced design pressure-suppression containment (Mark III) and during in-plant testing of facilities with the Mark I containment design, new suppression pool hydrodynamic loads were identified which had not been considered in the original design basis for Mark I and Mark II plants. These additional loads result from the dynamic effect of air, or noncondensible gas, and steam being rapidly forced into the suppression pool during a loss-of-coolant accident (LOCA) or a safety relief valve discharge from the primary system.

The NRC staff has identified and initiated a number of generic tasks to review and evaluate the results of the industry programs and to develop criteria for licensing actions on individual plants using the Mark I and Mark II containment designs. The staff efforts involving Mark I containments have been concluded. Task A-6 was completed with the issuance of the "Mark I Containment Short-Term Program Safety Evaluation Report" (NUREG-0408, December 1977). Task A-7 was concluded with the issuance of "Mark I

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 10, 1982

Peter R. Bloch, Chairman
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Washington, DC 20555

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In the Matter of
Wisconsin Electric Power Company
(Point Beach Nuclear Plant, Units 1 and 2)
Docket Nos. 50-266 and 50-301
(Repair to Steam Generator Tubes)

Dear Administrative Judges:

On May 4, 1982 Chairman Bloch telephoned me and requested answers to two questions relating to LER 82-007 for Point Beach, Unit 1:

Considering the leakage detected in the sleeved tube from which explosive plugs were removed.

1. Did any other tubes sleeved during the demonstration sleeving program for Point Beach Unit 1 have explosive plugs removed from the cold leg?

2, If so, in light of the information presented in the subject LER, does the Staff feel that there is any immediate danger to the public health and safety?

The enclosed preliminary report from the NRC Staff is hereby provided in answer to those questions. The gist of this report was read to Chairman Bloch on the telephone on May 7, 1982.

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Any further evaluation of this subject will be contained in the Staff's Safety Evaluation Report in this proceeding.

Sincerely,

Richard G. Bachmann
Counsel for NRC Staff

Enclosure:
Preliminary Staff Report
cc: Service List

PRELIMINARY STAFF REPORT ON LER 82-007/01T-0
FOR POINT BEACH, UNIT 1

In answer to question 1, there was one other tube sleeved during the demonstration sleeving program which had explosive plugs removed from the hot and cold legs. This tube was sleeved in both the hot and cold legs.

In answer to question 2, the Staff has reviewed the information contained in the subject LER just recently received by the Staff and has had telephone conversations with the Licensee Wisconsin Electric Power Company concerning the information. It is the Staff's preliminary finding that there is no danger to the public health and safety in light of the information presented in the LER.

The tube discovered to be leaking on the cold leg side was leaking at a rate of 20 drops per minute. This is equivalent to about 15 gallons per day. This is approximately 6% of the total allowable leakage of 250 gallons per day (gpd) for the Unit 1 steam generator as allowed by the Confirmatory Orders for Modification of License issued in November, 1979 for Point Beach Unit 1. This total leakage limit of 250 gpd is twice as restrictive as the allowable total primary to secondary leakage limit for the Point Beach Unit 2 and other PWR steam generators.

Though the Licensee has not identified the cause, type or location of the defect which caused the leak, it may well have been due to the plug removal process used during the sleeving outage. Hydrostatic testing has not revealed a similar leakage problem on the other tube from which explosive plugs were removed.

The Licensee has plugged the leaking tube R25C27 from which explosive plugs were removed. Even assuming that the remaining tube from which explosive plugs were removed was to leak due to damage from the plug removal process, the leak would take place well within the tube sheet (total length about 22 inches) since explosive plugs are only about 6-8 inches long.

Appendix A of the Staff's November 30, 1979 Safety Evaluation Report attached to the November, 1979 Confirmatory Orders for Modification of License for Point Beach Unit 1 calculates tube leakage from defects within the tubesheet for both the LOCA and MSLB conditions. The maximum calculated primary to secondary leakage for MSLB accident conditions is 9.5 gpm for a defect located 10 inches below the top of the tubesheet due to the narrow .008" gap between the tube and tubesheet wall. A defect associated with this explosive plug removal would presumably be further from the top of the tubesheet and the leakage rate would be further constrained. The maximum calculated in-leakage for the case of a nominal crevice gap of .008 inches was 5.5 gpm for LOCA accident conditions assuming a differential pressure of 800 psid. This leakage rate will not have any effect on ECCS performance. The necessary in-leakage identified in Appendix A of the Staff's SER to induce a steam binding effect which would retard ECCS performance is 1300 gpm.

Though the Staff does not have any immediate safety concerns regarding the one sleeved tube in Unit 1 from which explosive plugs were removed, it does note that the Licensee was not able to locate the leaking flaw with eddy current testing. Nor did previous inspection results show that a potential problem area existed. Therefore, the staff would expect further assurance from the Licensee that explosive plugs could be removed without causing or contributing to tube defects prior to approving sleeving of tubes that had been previously plugged with explosive plugs.

However, the Licensee has in fact indicated that it does not intend to attempt explosive plug removal in the future.

Timothy G. Colburn

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 30, 1982

Peter B. Bloch, Chairman
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In the Matter of
Wisconsin Electric Power Company
(Point Beach Nuclear Plant, Units 1 and 2)
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(Repair to Steam Generator Tube)

Dear Administrative Judges:

This is in response to the Licensing Board's request for an indication of a need for a hearing. (Tr. at 1126). The NRC Staff has reviewed the submittals of Wisconsin's Environmental Decade dated February 23, 1982 and March 11, 1982, and those of Westinghouse dated March 23, 1982. Based on this review, it is the Staff's opinion that a hearing on the public disclosure of information claimed to be proprietary by Westinghouse is not necessary.

Sincerely,

Richard G. Bachmann
Counsel for NRC Staff

cc: Service List