NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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May 25, 1984

Docket No. 50-336

A03719

Director of Nuclear Reactor Regulation
Attn: Mr. James R. Miller, Chief
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) K. L. Heitner letter to W. G. Counsil, dated December 30, 1983.

(2) W. G. Counsil letter to R. A. Clark, dated June 3, 1983.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Resolution of Open Items
Amendment 89 to DPR-65

Amendment No. 89 to DPR-65 for Millstone Unit No. 2 was issued by the NRC Staff in Reference (1). The amendment allows repair of degraded steam generator tubes by installing metal sleeves together with the current repair method which involves plugging the tube.

The Reference (1) transmittal letter and safety evaluation report identified two items for which the Staff requested additional confirmatory documentation. Specifically, the Staff requested NNECO to justify the 40% plugging criterion for degraded sleeves proposed in Reference (2) and to provide a status report on the efforts to more accurately determine primary-to-secondary leakage.

The information requested by the Staff is provided in the attachment hereto. It is concluded that the 40% plugging limit for degraded sleeves is appropriate

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based on the evaluation conducted in accordance with Regulatory Guide 1.121. We trust you find it responsive to your requests. My Staff remains available to assist you in your review of this matter.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Docket No. 50-336

Attachment

Millstone Nuclear Power Station, Unit No. 2

Resolution of Open Items

Amendment 89 to DPR-65

May, 1984

Confirmation of Sleeve Plugging Limit

Section 4.2 of the NRC Safety Evaluation Report[1] supporting Amendment No. 89 to Operating License No. DPR-65 for Millstone Unit No. 2 discusses allowable steam generator sleeve degradation. Northeast Nuclear Energy Company (NNECO) proposed[2] that steam generator tubes with sleeves degraded forty percent (40%) through wall would be plugged. The NRC Staff requested NNECO to establish equivalency in bending strengths for a tube and sleeve degraded forty percent through wall to support the proposed plugging limit for sleeves.

NNECO conducted reviews in accordance with the guidance of Regulatory Guide 1.121 to determine the appropriate plugging limit for the sleeves. Bending stresses were determined for both tubes and sleeves under normal and accident conditions. The following discussion addresses the issue of bending strength of steam generator tubes and tube sleeves as it relates to the proposed plugging criterion for sleeves.

Millstone Unit No. 2 employs a Combustion Engineering design nuclear steam supply system. Each of the two steam generators contain 8519 tubes in a U-bend configuration. The tubes are arranged in a triangular pitch. The tube sheet is 21.5 inches thick and 14 feet in diameter. Since the pressure is normally higher on the primary side of the tube sheet than on the secondary side, the tube sheet becomes concave upward. The tube sheet flexes further during a main steamline break due to the larger pressure differential between the primary and secondary sides of the steam generator. Design features have been incorporated into the steam generator to limit the tube sheet deflection. The most notable feature of the Millstone Unit No. 2 steam generator design is the stay cylinder depicted in Figure 4.3-2 of the FSAR included herein.

The loading conditions imposed on the tubes and sleeves during a main steam line break (MSLB) include the following. The pressure differential across the tube sheet during a MSLB causes the tube sheet to become concave upward. This flexure induces an end rotation on the sleeves and tubes. Crossflow conditions from the fluid flow from the downcomer into the tube bundle will impart a transverse load to the peripheral tubes. However, this load is not of concern to sleeves since the peripheral tubes (first three rows) are not sleeved due to the geometry of the steam generator channel head (sleeves cannot be inserted into peripheral tubes). Internal tubes containing sleeves will be shielded from the cross flow by the peripheral tubes and the sleeves themselves will be shielded by their parent tube. As such the cross flow loads will not be imparted to sleeves. Uplift forces on the tube bundle will exist due to the large steam flow resulting from the MSLB. These forces translate to an axial pull-out force in the area of the tube bundle containing the sleeves. Moments imparted to the tubes as a result of the uplift forces will be mitigated by the multiple tube supports between the top of the tube bundle and the tube sheet. Differential thermal expansion between the tubes/sleeves and the steam generator shell due to material property differences will impose an axial load on the tubes and sleeves.

In addition to the loads described above, the tubes and sleeves have a differential pressure across their boundary which produces an additional load.

The tube sheet will deflect due to the pressure differential regardless of the presence of a sleeve in a degraded steam generator tube. This deflection will rotate the steam generator tubes protruding from the top of the tube sheet from vertical. This tube rotation imparts a bending stress in the tube or tube/sleeve arrangement. The ability of the steam generator to perform its function under both normal and accident conditions is influenced by the capability of the tubes and/or sleeves to accommodate the bending imposed by the tube sheet deflection. In this regard the stiffness of the tube and sleeve material will determine what level of bending stress is imposed on the tube and/or sleeve by the tube sheet deflection.

The sleeve design is such that it is not as stiff as the parent tube. As such, the sleeve will accommodate the tube sheet deflection better than the original tube. This is demonstrated in Section 6.3.3 of Reference (2) where for normal operation, the bending stress for a sleeve is less than for a tube. This relationship holds true for accident conditions also.

NNECO has evaluated the stresses which will be imposed on a tube or a sleeve under the main steam line break accident. Under these conditions, stresses will result from the tube sheet rotation (bending), axial compression (assuming the tube is fixed at the first egg crate support), and thermal growth differences due to the different materials used for tubes, sleeves and the steam generator shell. The maximum stress in a degraded sleeve under these conditions is less than the yield strength of the sleeve by more than a factor of three.

The fact that the tubesheet deflection occurs regardless of the material properties or design of the tubes or sleeves renders a bending strength comparison immaterial. Margin to burst is the appropriate measure assuring steam generator integrity when evaluating a plugging limit for steam generator tubes and sleeves. It has been demonstrated[(2)] that a sleeve degraded 40% through wall has a factor of safety of 3 or more against burst under all conditions specified in Regulatory Guide 1.121. As such, the proposed plugging limit has been appropriately supported.

<u>Primary-to-Secondary Leak Rate Determinations</u>

The accompanying letter to Amendment No. 89 to DPR-65 requested NNECO to provide a status report of actions taken to more accurately determine primary-to-secondary leakage.

NNECO initiated actions to improve primary-to-secondary leakage rate determinations following the conclusion in early 1983 that leakage rates measured by isotopic analysis of the blowdown system effluents were non-conservative.[(3),(4)] These actions included the use of the steam jet air ejector radiation monitors and the reactor coolant system mass balance determination.

Leak rates utilizing the steam jet air ejector discharge flow have been performed since the plant returned to service in March, 1983. This technique was included in the leak rate procedures during January, 1984. Experience has been gained in utilizing this more sensitive measurement technique with leak rates calculated from the isotopes of Xenon 133 and 135. In addition, the reactor coolant system (RCS) leak rate program has been reviewed and operational techniques for identifying sources of leakage improved. This has resulted in increased confidence in the ability to identify sources of RCS leakage, including possible primary-to-secondary leakage.

The increased ability to more accurately determine primary-to-secondary leakage rates has not been quantified due to the improved performance of the steam generators since the March, 1983 outage. The lack of a primary-to-secondary leak during the remainder of Cycle 5 and the start of Cycle 6 make it impossible to verify any improvements. However, the confidence in the ability to measure leakage has increased due to the two refinements discussed above.

During future periods of significant steam generator leakage (greater than 0.1 gallons per minute) the steam generator blowdown calculation and the steam jet air ejector calculation will provide measurement of the leakage. The steam jet air ejector calculation will provide a total leakage number which will be matched against the sum of the individual blowdown leak rates. As an additional check, if the above leak rates are not compatible, the primary-to-secondary leak rates can be compared to the RCS leak rate with other known leakage subtracted.

This program has been formalized in Inservice Test T84-11 and will be implemented when leakage exceeds 0.1 gallons per minute. The results will be utilized to determine the most effective and accurate measurement technique.

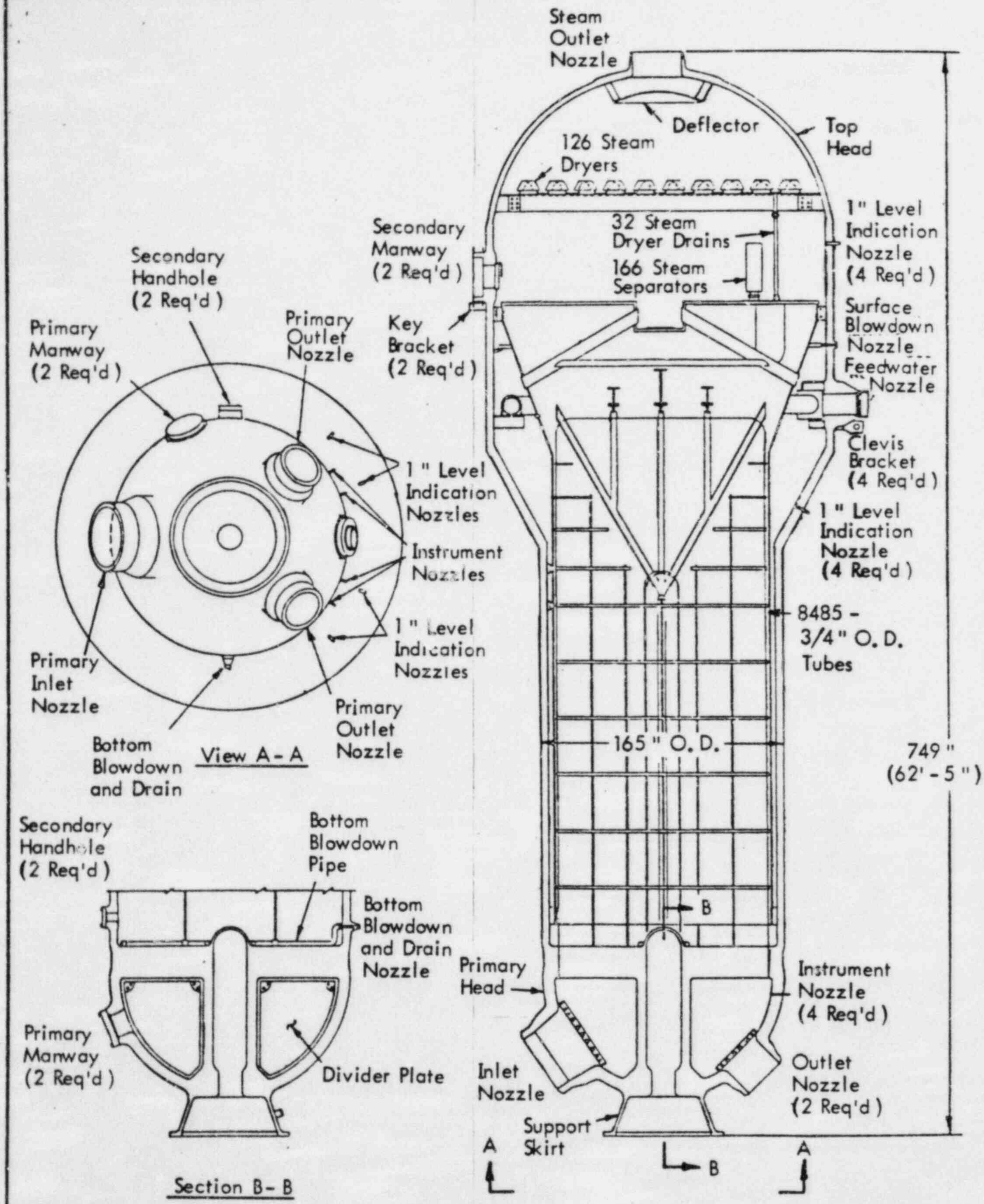
<u>References</u>

(1) K. L. Heitner letter to W. G. Counsil, dated December 30, 1983.

(2) W. G. Counsil letter to R. A. Clark, dated June 3, 1983.

(3) E. J. Mroczka letter to R. C. Haynes, dated March 31, 1983.

(4) W. G. Counsil letter to R. A. Clark, dated March 14, 1983.



Millstone Nuclear Power Station Unit No. 2

Steam Generator

Figure 4.3-2