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VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 11, 1975

Mr. Norman C. Moseley, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

Serial No. 794
PO&M/JTB:clw

Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Dear Mr. Moseley:

The purpose of this letter is to document discussions which have been held with members of your staff concerning the steam generator tube phenomena referred to as "denting" and to apprise you of the current developments as we know them.

During the current refueling outage on Unit No. 1, inspections were conducted on the steam generator tubes to acquire additional information concerning "denting." As you are probably aware, steam generator tube diameter reductions, i.e. "denting," have been experienced at a number of operating nuclear power stations. Because of the generic nature of the phenomena, the nuclear steam supply system manufacturer, Westinghouse Electric Corporation, is doing extensive investigations and studies into the matter. Enclosed herewith are two "Technical Bulletins" prepared by Westinghouse summarizing the "denting" phenomena.

A meeting between the Regulatory Staff and representatives from Westinghouse was held in Bethesda, Maryland on November 21, 1975 to discuss "denting" on a generic basis. I understand that a representative from Region II was in attendance at that meeting. As a result of the meeting, it was agreed that Westinghouse would prepare a topical report on "denting" and would apprise the Regulatory Staff of their proposed investigative program.

As related to our units, the inspections conducted on the Unit No. 1 steam generator tubes indicate that there is significant "denting." The most severe "denting" has occurred on steam generator "A". The data obtained during the inspections are presently being evaluated to quantify the amount of denting which has occurred. In an attempt to minimize further denting, we have changed our method for controlling feedwater pH by changing from ammonia to cyclohexylamine on both units.

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Both the Station Nuclear Safety and Operating Committee and the System Nuclear Safety and Operating Committee have reviewed the matter of "denting." We are continuing to follow the generic implications of this phenomena and will continue to apprise you and your staff of significant developments.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

Attachments

cc: Mr. Robert W. Reid

Westinghouse Nuclear Service Division

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Technical Bulletin

Received w/Ltr Dated 12-11-75

An advisory notice of a recent technical development pertaining to the installation or operation of Westinghouse-supplied Nuclear Plant equipment. Recipients should evaluate the information and recommendation, and initiate action where appropriate.

P.O. Box 2728, Pittsburgh, PA 15230

Subject	Number
Occurrence of Denting in Steam Generator Tubes	NSD-TB-75-12
System(s): Steam Generator	Date: 10/15/75
Affected Plants: All operating and construction	S.O.(s): 120
References: None	Sheet 1 Of 3

BACKGROUND

During the scheduled refueling outage of the Turkey Point Unit 4 plant in May 1975, eddy current inspections of the three steam generators were performed. The purpose of the inspection was to provide a measure of tube integrity following the steam generator chemistry change to AVT which occurred in September 1974. This insepction revealed a continuation of thinning due to the previous phosphate water chemistry treatment. The corrosion was limited to the area immediately above the tube sheet in the area of sludge accumulation. It has been attributed to residual phosphates retained in this sludge material which surrounds the tubes. These observations are identical to the findings at several other plants with a comparable operating history. The continued corrosion in the sludge pile has provided the basis for increased emphasis on sludge removal by lancing techniques in these units. Each of the steam generators was sludge lanced during the shutdown. Also during the May outage, steam generator mechanical modifications were accomplished.

INFORMATION

Examination of the eddy current tapes revealed additional signals at the intersection of the tube support plates. These signals were interpreted by Zetec and Westinghouse personnel to be due to physical deformation of the tube wall.

The unit was returned to full power in late June and operated until August 3, 1975 when shutdown was initiated to repair a primary to secondary leak which developed in the B steam generator. The leak was identified as a peripheral tube and this tube was plugged. The leak was between the second and sixth tube support plate on the hot leg side. The axial location could not be accurately determined since the standard eddy current probe could not pass beyond these locations.

Additional Information, if Required, may be Obtained from the Originator. Telephone 412 – 256-5413 or (WIN) 236-5413

Originator	Approval
T. E. Bowman, Lead Engineer	F. C. Wellhofer, Manager
Operating Plants Services	Mechanical Technology
T. E. Bowman	F. C. Wellhofer

WESTINGHOUSE FORM 55378

During the August shutdown, a tube sample was removed from the B steam generator for laboratory examination to more fully explore the eddy current denting signals. Examination of the tube is still in progress, and the findings thus far are enumerated below:

1. The tube has shown no significant loss in integrity.

2. Wall thinning to the extent of 4 percent (maximum) was evident on the OD surface adjacent to the tube support plate.

3. Tube wall inward deformation up to the extent of 7 mils on the radius was measured by profilometry. The denting is rather uniform, slightly oval, and is the area adjacent to the 3/4 inch thick tube support plate.

4. No gross "working" of the tube surface is evident from metallography.

5. There is no evidence of intergranular stress corrosion.

Both analytical and laboratory studies are in progress to pinpoint the possible cause for this finding. Eddy current tape reviews are also in progress to define any patterns to the denting which may exist.

Since no gross surface working or metal grain deformation is evident in the metallography, vibration forces such as those which might be induced by flow and turbulence do not appear a likely cause for denting. This is also consistent with analytical predictions of dynamic forces in the tube bundle. The definitive mechanism for denting, therefore, is not as yet readily explainable. It is clear that the denting is not related to steam generator mechanical modifications since the eddy current indications were evident prior to implementation of these modifications.

Since restart of the plant in August, another tube leak was detected in the B steam generator. The plant entered into another shutdown on September 21. The leakage was again identified as a peripheral tube and this tube was plugged in addition to several other tubes in this wedge region. It has not been ascertained as yet whether the tube leakage is related to the denting. A section of tubing from the hot and cold legs was removed during the outage for further investigation into the denting phenomena.

A review of eddy current inspection records is in progress for other plants. Some denting signals are also found in this review to an extent yet to be finalized. This review will cover Beznau I and II, Prairie Island , Ginna, Robinson, San Onofre, Point Beach 2, and others as inspections are performed in the future.

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It is important to note that no loss of tube integrity is indicated by this inspection. The eddy current signal from denting is relatively large compared to the usual amplitude of corrosion related phenomena, and calibration standards are being devised to better quantify the amount of physical deformation.

The specific mechanism involved is being pursued with expeditious effort by responsible division within Westinghouse.

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Subject	Number
Localized Steam Generator Tube Diameter Reductions	NSD-TB-75-16
System(s)	**Date**
Reactor Coolant System	November 17, 1975
Affected Plants	**S.O.(s)**
	120
References	**Sheet 1 Of 4**
NSD Technical Bulletin 75-12	

Information

Localized steam generator tube diameter reductions have been observed at some operating plants. Evaluation of such anomalies with regard to the effects on the operation and safety analyses of such plants have been performed. The evidence available to date indicates that operation of affected units can continue without undue risk to public health and safety. Plans have been made to obtain additional data in order to better understand the causes and extent of such diameter reductions: the intent being to minimize their occurrence in the future.

The following elaborates on what has been observed, the evaluations performed to date, and the plans made to better characterize such localized steam generator tube diameter reductions.

Examination of eddy current inspection data obtained from the Turkey Point Unit 4 steam generators during the scheduled refueling in May, 1975 revealed the presence of anomalous signals at the tube/tube support plate intersections. These signals were interpreted as being a result of localized deformation of the tube wall in the area of the tube support plates.

Following refueling, Turkey Point 4 was returned to power in early June and operated until August 3, 1975 when the plant was shut down to repair a primary-to-secondary leak which developed in the "B" steam generator. The leak was in a peripheral tube on the hot leg side at approximately the 2nd tube support. This tube was plugged and another tube having a similar EC indication on the hot leg was removed for examination.

Additional Information, if Required, may be Obtained from the Originator. Telephone 412 – 256-7183 or (WIN) 236 – 7183

Originator
V. W. Doutt
V. W. Doutt, Manager
Operations Support

Approval
Wellhofer
for F. C. Wellhofer, Manager
Mechanical Service

The plant was restarted in August and operated until September 21 when it was shut down because of a primary-to-secondary leak. The leakage was again identified as being in a peripheral tube at the 2nd support plate, and this tube was plugged. During this shutdown, both a hot leg and cold leg tube exhibiting diameter reduction indications were removed from the unit for examination. Both tube sections extended from the primary face of the tubesheet to beyond the 5th tube support.

Laboratory examination of these tubes is still in progress. The results available to date indicate the following which are essentially similar to the results of the examination of the August tube:

1. Plastic deformation of the tube wall has occurred in the area adjacent to the 3/4" thick tube support plates, resulting in a constriction of the tube diameter in that area. Maximum reduction in diameter in the tubes examined in the laboratory is approximately 20 mils.

2. Metallographic examination and mechanical property tests indicate no loss in tube integrity.

3. Wall thinning to the extent of 4 percent (maximum) was observed on the O.D. of the tube surface adjacent to the tube support plate.

4. There is no evidence of stress corrosion cracking.

5. There is no evidence of extensive deformation of the surface layer within the dent region.

During September, the Surry 1 unit was shut down just prior to a scheduled refueling. At the time of shutdown, primary-to-secondary leakage was experienced in steam generator "A." Three leaking tubes were identified in this steam generator; all were in Row 2, which is one tube removed from the divider lane. The leaks were in the hot leg in the region of the 6th and 7th support plate.

Eddy current examination of this unit is still in progress. Initial results indicate prevalent localized diameter reduction. Plug gaging of the leaking tubes indicates they will not pass a standard EC probe beyond the 2nd tube support. Tubes have been pulled from this unit for detailed laboratory examination. The leaking tubes were plugged and were not removed due to inaccessibility and proximity to the divider plate.

Visual examinations of the tube support plates were made through the tube lane handholes. This examination indicated inplane hour-glassing of the flow slots in the center of the tube support plates. No other physical changes in the tube support plates were observed.

The support plate, tubes, and anti-vibration bars were examined in the U-bend region of the bundle. All components were intact and in the as-shipped condition.

Since the initial evidence of the localized diameter reduction problem, analytical and experimental programs have been underway to establish the mechanism responsible for the phenomena. Examination of affected tubes shows no evidence of surface deformation which would be expected if the problem was caused by flow induced tube vibrations. This is also consistent with analytical predictions of dynamic forces in the tube bundle. Analytical studies to explore the possibility of dia. reduction resulting from thermal ratchetting induced by local thermal/hydraulic effects in the tube support region also indicate that this is not a feasible mechanism. At the present time, it appears that the most likely mechanism is the growth of deposits and/or corrosion product within the tube/tube support plate annulus, giving rise to forces sufficiently high to locally plastically deform the tubes.

Methods and procedures are being developed so that a section of tube within the adjacent tube support plate area can be secured for laboratory examination at some future date.

Additional EC examinations are planned to categorize the extent of the localized diameter reduction.

The eddy current inspections of those units which have converted to AVT after extended phosphate operation have indicated that the phenomenon is localized at the tube support plate locations since the AVT conversion. It has been observed in the past essentially on a random basis and to a minimum extent in these units during phosphate operation.

Eddy current inspections thus far on those plants which operated for a short period with phosphates and switched to AVT indicated no denting was taking place.

Laboratory testing is underway to simulate conditions of the proposed mechanism and to identify corrective measures which can be applied to prevent further wall deformation. This testing includes the exposure of tube/tube support simulations to aqueous environments containing contaminants corrosive to carbon steel. The intent is to determine if corrosion products will accumulate and expand on the tube wall. Also, means for neutralizing or removing deposits in the tube/support plate region are being investigated.

Possible implications of localized tube diameter reduction have been considered with respect to normal plant operation, expected transients, and postulated accident conditions.

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Regarding normal operation with expected transients and based upon experience at several plants which have operated with tubes in this condition, if primary-to-secondary leakage from affected tubes occurs, it can be detected at low, controlled levels and progresses slowly. If necessary, shutdown and repairs can be accomplished in an orderly manner and with no undue risk to the public health and safety.

With respect to postulated design basis accidents, the criterion of importance is that significant tube damage or ruptures will not occur during the course of the accident as a result of tube diameter reduction. Our evaluation, based on analyses previously conducted, indicates that limiting stresses will not be exceeded, and hence, tube failures are not expected to occur. Also, collapse strength of tubes is considered to be adequate even with significant diameter reduction, based upon available test data.

In summary, the assessment of postulated accidents as presented in the safety analysis report is not expected to change as a result of localized tube diameter reduction.