Kelmore
Verrelli
Elliott

March 15, 1977

A-3

NOTE TO: K. Goller
G. Lear
R. Reid
A. Schwencer
D. Ziemann

RE: PRAIRIE ISLAND HEARING

The Appeal Board in the Prairie Island proceeding has requested the Staff provide information relating to characteristics and operating history of all domestic PWR's. In reviewing this request with the staff consultants, it appears that the project managers for the individual facilities are probably the best source of this information. It is a massive job if one individual has to do it, however, if the project manager for the individual projects takes some time out of his regular duties to address this matter, I think that they will find that we already have the answer to most of the questions and can readily obtain answers to those we do not have on hand.

The answers should be provided to Marshal Grotenhuis by Monday, March 22. The individual project managers should maintain notes of the sources of the information used to respond to each of the questions for the individual facilities.

The questions are attached.

Joseph F. Scinto

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QUESTIONS

4. In order for us to develop an understanding of the possible relationship between the denting phenomenon and the Prairie Island steam generators, all available operational experience relevant to this matter should be considered. To this end, the staff is directed to provide information which may be of significance for this review[2/] for each PWR in the U.S. and (where the information is available) for foreign PWR's. At least the following items should be included.

a. Significant operational history, e.g., date of initial operation, periods of major down time, etc.;

b. Materials of construction used for major secondary system components, e.g., steam generator (SG) tubes, SG tube sheet, SG tube support plates (TSP), condenser tubes, etc.;

c. Operational history of secondary water treatment, e.g., periods of use of phosphates, AVT, condensate demineralization, etc.;

d. Condenser cooling water -- typical chemical composition;

e. History of significant condenser tube leakage;

f. "Denting" history (date discovered, how discovered, leakage associated with denting);

g. SG tube plugging history (dates and number of tubes);

h. Restrictions imposed on plant operation due to

RECEIVED

PUBLIC SERVICE COMMISSION OF WISCONSIN

FILE NO.

DEPARTMENTAL CORRESPONDENCE

November 7, 1979

Subject: Meeting Between Wisconsin Electric Power Company, Westinghouse Electric Corporation, and Nuclear Regulatory Commission Staff on the Steam Generator Problems at Unit 1 of the Point Beach Nuclear Generating Plant -- November 5, 1979

By: L. L. Smith

On November 5, 1979 I attended a meeting at the Nuclear Regulatory Commission's Bethesda's offices between representatives of Wisconsin Electric Power Company, Westinghouse Electric Corporation, and the NRC staff on the problem of steam generator tube degradation at Unit 1 of the Point Beach plant. A copy of the list of attendees is attached. In summary, the meeting consisted of the following presentations:

1. Wisconsin Electric Power Company reported on the general results of the steam generator tube inspections done during the current refueling outage and an update on the number of additional tubes requiring plugging and the accumulated total of plugged tubes in each of the two steam generators.

2. Westinghouse described the results of detailed physical examinations, metallurgical tests performed on three tube sections removed from steam generator "A" for the purpose of detailed analysis.

3. Westinghouse reported on the general results of its assessment of continued steam generator integrity in their present condition and presented results of verification analysis

that the various safety criteria involving steam generator tube integrity are still met.

4. Wisconsin Electric Power described a number of options under consideration for potential modifications to future operations to address this problem on a continuing basis.

Conclusions

1. Based primarily on the results of testing and reanalysis done by Westinghouse, it was concluded that the various steam generator parameters associated with tube integrity and performance still meet safety criteria and remain within limiting conditions for safe operation.

2. Based primarily on the review by Westinghouse and Wisconsin Electric of existing overall accident analyses, it was concluded that the safety analyses remain valid and are satisfied with the Unit 1 steam generators in their present condition, i.e., with approximately 10% of the tubes plugged. In other words, the original safety analyses done to satisfy NRC licensing requirements included an assumption of up to 10% of the steam generator tubes plugged and since no more than 10% will be plugged on completion of the current repairs, the existing analyses remain satisfied.

3. Although there are a number of operational options under consideration to attempt to further arrest or mitigate continued tube degradation due to corrosion, it was agreed that upon completion of current plugging and other maintenance, Unit 1

can be safely started and operated at full power. The unit is presently scheduled to return to service on November 17 (previous schedule mentioned was November 10). No NRC action or approval is required.

4. It was my perception of the NRC staff's position that it is satisfied with the continued safe operation of Unit 1 in the near term. NRC staff at the meeting, however, expressed concern about the long-term outlook if the tube degradation experience continues. There are already NRC requirements covering the frequency of in-service inspections required based on sample testing of steam generator tubes in the current Point Beach Technical Specifications. Staff indicated that they would continue to watch this aspect of Point Beach operating experience closely in the future both from the specific standpoint of safe operation of Point Beach as well as the more generic steam generator tube problems in this and other similar plants.

The following are "minutes" prepared by me from my notes and recollections of the highlights of the more detailed discussions and follow the presentations outlined in the opening summary above. The meeting was chaired by Mr. Charles Trammell who is the NRC staff project manager from the Operating Reactors Branch who was assigned to the Point Beach Plant, and also by Mr. E. D. Liaw, who is NRC's staff expert on the steam generator corrosion problem. In effect, the meeting was conducted in a very open roundtable style with oral presentations and question and answer exchanges. There were no formal or written presentations and I am not aware whether the NRC will prepare or distribute minutes of the meeting.

Wisconsin Electric Presentation on Current Status

Mr. Fay of the utility reported on the recent steam generator tube history from the August outage caused by a leaking tube which resulted in 100% inspection and plugging of 87 additional tubes. In preparation for the fall refueling outage, utility staff met with Westinghouse in September to plan strategy for tube inspections during the refueling outage scheduled for early October. At that time it was decided to do more than the required amount of inspection using the newly-developed multi-frequency eddy current technique. It was also decided at that time to remove sections from a few tubes in order to obtain specimens for further detailed physical examinations and metallurgical testing.

Mr. Frieling of the utility covered some of the earlier history of plant operations including the switch from phosphate to all volatile secondary water trestment at Unit 1 in September of 1974. The status of plugged tubes prior to the recent, refueling outage from photo-verified counts was 251 tubes plugged in each of the two Unit 1 steam generators. For the October refueling outage, the utility originally scheduled 75 tubes for inspection by the multi-frequency eddy current method.

These were located in the area of the hot leg side where most of the problems have occurred previously. Out of the 75, 18 bad tubes were found; that is, the eddy current tests indicated defects greater than the 40% threshold where plugging would be required. Based on this failure rate (18 out of 75) it was decided to increase the sample inspected to 200 tubes, also concentrating in the region with the history of worst corrosion. Out of these 200 tubes, a total of 75 showed indications of defects in excess of the 40% threshold. As a result of this failure rate (75 out of 200) the inspection was increased to 100%, or all tubes were inspected. The results for steam generator "A" remained at 75 bad tubes which would be plugged. A similar procedure was followed with steam generator "B" and resulted in 65 additional tubes to be plugged. On completion of plugging operations, the accumulated totals will be 326 tubes plugged (exactly 10%) in steam generator "A" and 316 tubes plugged in steam generator "B".

The 75 additional tubes plugged in steam generator "A" as indicated above is really a combination of 73 defective tubes and two good tubes from which sections were removed for further analysis. Sections of three tubes were physically removed for further detailed examination and testing. Sample sections were taken from one bad tube (89% indicated defect but not leaking) from the problem region, from one good tube (no eddy current indication) from the problem region, and from one good tube (no eddy current indication) from the good region around the periphery of the tube sheet.

Examination of all of the three removed tubes showed intergranular stress corrosion attack in the crevice region within the tube sheet (see attached figure). The major portion of the tubes above the tube sheet were reported to be in generally good condition. The type of corrosion is not unique or unexpected and is believed to be caustic assisted corrosion associated with the use of phosphate secondary water treatment and the build-up of sludge and corrosion products in this small crevice area between the tube and the tube sheet on the secondary side. Photomicrographs of the tube wall cross-section in this area show general corrosion deterioration in the outer surface with sharper circumferential stress corrosion cracks penetrating deeper through the wall cross-section. The most severe defect found in the section of bad tube which indicated an 89% defect on eddy current test showed a general intergranular corrosion cracking approximately 20 mils (.020 inches) deep and the deepest stress crack penetration of approximately another 20 mils. The total through wall penetration of the crack, therefore; was 40 mils or approximately 80% through the tube wall which has an original thickness of 50 mils. These specific measurements must be qualified somewhat in that this tube broke during the pulling operation required to remove it for testing. Also, considerable amount of pulling stress in tension was placed on the tube during removal and may have increased the crack depth from its actual undisturbed condition.

The second tube sample was from a tube which was indicated to be good (no eddy current indication) but located in the problem area. Again, examination showed generalized corrosion attack of the outside tube surface in the tube sheet crevice region. A photomicrograph of the tube wall cross-section showed general corrosive deterioration approximately 5 mils deep with the deepest of the stress corrosion cracks pentrating approximately an additional 20 mils deep for a total of 25 mils or 50% of the original tube wall thickness.

Sections of the third tube removed were from a good tube (no eddy current indication) from the peripheral region where no problems have been experienced. Examination also showed the same generalized corrosion attack of the outside surface throughout the tube sheet crevice region. The generalized corrosion deterioration was approximately 5 mils deep with the deepest stress crack extending approximately another 15 mils deep for a total through-wall penetration of 20 mils, or 40%.

Although the general corrosion exhibited by the last two good tubes was not unexpected, there was concern about the measured depth of crack penetration on these two tubes compared to the fact that the eddy current test indicated no recognizable defects. The examiaation results, however, as mentioned above are qualified by the fact that the severe tube cracks may have been worsened by the pulling stress (between about 10,000 and 25,000 pounds) which was required to remove the sample sections from the tube sheet. It was indicated, for example, in one case where the section exhibited 10-13% elongation as a result of tensile stress from the pulling operation.

Westinghouse people described the nondestructive testing techniques used for tube inspection which for steam generator tubes is confined primarily to eddy current techniques using a probe into the interior of the tube from the bottom side of the tube sheet which is accessible through the lower plenum of the steam generator. A new multi-frequency eddy current technique has been developed which offers several advantages over the previously-used single-frequency method. Basically the new technique results in more definitive information and a potential for more positive identification of defects due to the collection of data at four different frequencies and the flexibility of comparing data at one or more of the frequencies in order to get better discrimination and cancel out noise or interference effects. During the August inspection at Point Beach, both methods were used to gain experience and make comparisons between the two although not all tubes were inspected by both techniques. During the 100% inspection in October, the multi-frequency technique was used. Because of the better detection results offered by the new multi-frequency eddy current method, the question was raised as to what extent of the bad tubes found during the October inspection were actually tubes with previously undetected defects which could now be identified by the better method as opposed to the identification of new or increasing defects caused by continued or accelerated corrosion since the last inspection. There is no real way to quantify this, however, Westinghouse estimated that of the original sample tested of 75 tubes where both inspection methods had been used, that about 2/3 of the tubes found to be bad (40% defect or greater) were identified because of better detection methods and 1/3 were the result of continued or accelerated corrosion since previous inspections. Also as mentioned above in the description of the tubes removed for detailed testing, there is still some concern about inspection sensitivity even with the newer method since two of the tubes which had shown no eddy current indications did have significant crack defects on detailed examination.

Westinghouse described further testing done on mechanical properties of the tube material removed and how remaining strength based on these tests indicated that the structural strength remained adequate, both for normal operations and under different assumed accidents. Since in the Point Beach situation all of the tube deterioration is confined to the tube sheet crevice region, the analysis concentrated on this area, however, data was also given for defects outside the tube sheet area. The fact that the deterioration is confined to the tube sheet area is a disadvantage from the standpoint of attempts to arrest the corrosion but is an advantage with respect to the safety performance as a result of tube failure or rupture since the leakage and any movement of the tube is constrained because it is rather tightly bound due to the accumulation of corrosion products and sludge in this crevice region. It was indicated that the limiting condition for failure inside the tube sheet was a minimum thickness of 5 mils or 10% (90% penetration of a circumferential crack) for the most severe condition. For tube rupture at normal operation or several other conditions it was indicated that all that was required is enough ductility in the remaining tube material to allow the tube to expand to contact to tube sheet. The results of tests on removed samples indicated that adequate ductility of the remaining material of the corroded tubes is maintained to meet the limiting condition. In the case of a full circumferential crack within the tube sheet, it was indicated that the resultant leakage will either be constrained if the failure is deep enough toward the bottom of the tube sheet or will blow out if it is near enough the top of the tube sheet such that the resultant leakage would be easily detectable. This has also generally been the pattern of tube leakage detection during operations at Point Beach where relatively small minor leakage was detected and monitored and if increasing at a significant rate or to a significant level action can be taken to shut down for repairs. Westinghouse detailed analysis and testing also indicated that the leak rate will progress to a level requiring plant shut-down prior to the crack or rupture reaching a critical length or size which would lead to a complete break failure. This was indicated to be true even for defects or ruptures above the tube sheet.

There was a brief review and discussion concerning steam generator problems at other Westinghouse plants, particularly those with similar steam generator designs. Nearly all of the operating plants have some history of operations with the phosphate secondary water treatment although Point Beach used phosphate for some of the longest times because of the age of Unit 1. Even though there has been a number of steam generator tube problems at various reactors, Point Beach Unit 1 is experiencing the most severe corrosion problem in the tube sheet crevice region. Differences in the tube corrosion problem evidenced at the various plants are not fully explained or directly comparable. This is due in part to small variations in design and operations, different sizes of units, different water chemistry and other perhaps subtle but important differences or combinations of factors which are not known or not clearly identified. The two existing plants were steam generator replacements are being made or proposed (Surry and Turkey Point) are different in that the serious tube degradation has been caused by corrosion build-up and tube denting up in the tube support plate region rather than problems with tube sheet crevice corrosion such as Point Beach. Problems with condensor tube integrity are also involved in those two locations since they use sea water for condensor cooling and intrusion of chlorides into the secondary system accelerates corrosion. Westinghouse was asked to describe the design and structural modifications taken on newer steam generators to eliminate or mitigate these problems. They believe that the newer designs have adequately addressed most of these problems, however, there is very little operating history with steam generators of the latest design.

Mr. Frieling of Wisconsin Electric described the normal and additional actions which will be taken prior to start-up of the unit. Besides the normal primary and secondary hydrostatic testing, they will perform a newly-developed and novel procedure in an attempt to flush out some of the sludge and corrosion product deposits from the tube sheet crevice. This is a modification of a similar attempt done initially at a Japanese plant. It was reported that the Japanese technique was successful in removal of some of these solids but the overall effect of whether corrosion will be significantly reduced is unknown until additional operating experience is accumulated. Very briefly, the method involves filling the secondary side with approximately 2 feet of water above the tube sheet, heating it up with primary pump and decay heat power to approximately 250^0 F under pressure and then quickly depressurizing the secondary side to initiate boiling in hopes that the boiling in the crevice area will lift out some of the deposited solids. This operation will take several days and it is one of the reasons for the later start-up target date.

Operating Options

Wisconsin Electric described several options under consideration in the basic effort to arrest corrosion or reduce the rate of corrosion and therefore extend tube life and reduce the failure rate.

The first option would be to operate at a primary pressure of 2,000 pounds. This is a reduction from normal primary pressure of 2,250 pounds. There is no reduction in power output, temperature, or efficiency of the unit. The primary benefit is a reduction in the mechanical stress that the tubes are subject to because of the pressure difference between the primary and secondary side of the steam generator. Normal secondary side pressure is 850 pounds; therefore, the differential pressure under normal conditions at full load is 1,400 pounds. With reduced primary pressure, the pressure differential would be reduced to 1,150 pounds.

The second option is to reduce the primary coolant temperature on the hot leg side of the steam generator. The benefit here is that testing and analysis to date indicate that the corrosion and corrosion rate is temperature-dependent. This is also evidenced by the fact that the severe corrosion is only on the hot leg side which operates at approximately 600° F. and there is very little corrosion on the cold leg side which operates at about 540° F. The specific transition temperature or temperature range between high and low corrosion rates is not known but the proposal would be to reduce power output of the unit to approximately 70% of full load. At this level the hot leg temperature would be reduced from 600° F. to approximately 550° F. It is hoped that this would reduce corrosion rates to a level that would significantly extend the life of the tubes. The sacrifice would be at a reduction in capacity (30%) of about 150 megawatts.

The third option also involves reducing temperature of the primary coolant in the hot leg and also involves a reduction in unit output. Unlike the method used for temperature reduction in Option 2, this approach involves keeping all of the pressure and temperature parameters as is and simply backing down power output which also results in a reduction in the temperature difference including a reduction of the hot side temperature. The effect is not as great as Option 2, however, in that a reduction to about 70% power would not reduce the hot side temperature to 550° F. as in Option 2 but to something higher than that.

There was a brief discussion of the ultimate options up to and including the worst case of a total replacement of the entire steam generators. The company indicated that they were presently putting together the economic studies on these options in connection with preparation for the upcoming PSC investigation and hearing. The significant time period indicated (approximately 2 years) to fabricate a replacement steam generator significantly impacts the timing of decision-making and the resultant economic or power supply impacts should replacement be required.

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Attachment 3

Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

November 16, 1979

Mr. J. G. Keppler, Regional Director
Office of Inspection and Enforcement,
Region III
U. S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

DOCKET NO. 50-266
POINT BEACH NUCLEAR PLANT UNIT 1
LICENSEE EVENT REPORT NO. 79-017/01T-0

Enclosed is Licensee Event Report No. 79-017/01T-0 (a 14-day followup report) with an attachment which provides a description of an event reportable in accordance with Technical Specification 15.6.9.2.A.3, "Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.".

We will provide you with an additional summary report including the results of our investigations of the tube samples which were removed for metallurgical analysis upon completion of our review of this data. This matter will be the subject of a meeting with representatives of the Office of Nuclear Reactor Regulation in Bethesda, Maryland, on November 20, 1979.

Very truly yours,

Executive Vice President

Sol Burstein

Enclosure

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