UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
WISCONSIN ELECTRIC POWER COMPANY	)	Docket No. 50-266
(Point Beach Nuclear Plant,	)	
Unit 1)	)	
	)	

ORDER MODIFYING CONFIRMATORY ORDER OF NOVEMBER 30, 1979

I

Wisconsin Electric Power Company (the Licensee) is the holder of Facility Operating License No. DPR-24 which authorizes the Licensee to operate the Point Beach Nuclear Plant, Unit 1, located in Two Creeks, Wisconsin, under certain specified conditions. License No. DPR-24 was issued by the Atomic Energy Commission on October 5, 1970, and is due to expire on July 25, 2008.

II

Inservice inspections of the Point Beach Unit 1 steam generators performed during August 1979 and October 1979 outages have indicated extensive general intergranular attack and caustic stress corrosion cracking on certain of the external surfaces of the steam generator tubes. The NRC Staff determined in November 1979 that additional operating conditions would be required to assure safe operation prior to resumption of operation of Point Beach Unit 1 from a refueling outage. Such conditions were imposed by Confirmatory Order for Modification of License dated November 30, 1979. In addition to those conditions, the Staff has now determined that additional conditions are required to provide continued assurance that Point Beach Unit 1 can be operated safely.

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These additional conditions are analyzed in a Staff Safety Evaluation Report, dated this date, which is attached to this Order. The Licensee has agreed to this condition by letter dated December 31, 1979.

III

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Part 2 and Part 50, IT IS HEREBY ORDERED THAT License No. DPR-24 be amended, in the manner hereafter provided, to include the following conditions in addition to those conditions listed in the Confirmatory Order of November 30, 1979:

1. Unit 1 will be operated at a reactor coolant pressure of 2000 psia with the associated parameters (i.e., overtemperature ΔT and low pressurizer pressure trip point) with the limits indicated in the Safety Evaluation Report appended to this Order.
2. The licensee shall develop and follow the necessary procedures for operating Unit 1 at the conditions described in condition 1 above.

IV

In view of the above, this amendment of License No. DPR-24 is made immediately effective. Accordingly, within 48 hours of receipt of this Order, the Point Beach Unit 1 facility shall be operated at a reactor coolant system pressure of 2000 psia within the parameters described above.

V

Any person whose interest may be affected by this Order may within twenty days of the date of this Order request a hearing with respect to this Order. Any such request shall not stay the effectiveness of this Order. Any request for a hearing shall be addressed to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555.

In the event a hearing is requested, the issues to be considered at such hearing shall be:

1) Whether the facts stated in Section II of this Order are correct; and

2) Whether this Order should be sustained.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold Denton

Harold Denton, Director
Office of Nuclear Reactor Regulation

Attachment:
Staff Safety Evaluation Report, dated January 3, 1980

Effective date: January 3, 1980
Bethesda, Maryland

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE POINT BEACH UNIT 1 STEAM GENERATOR

TUBE DEGRADATION DUE TO DEEP CREVICE CORROSION

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-266

INTRODUCTION

Wisconsin Electric Power Company (the licensee) has requested changes to the Technical Specifications of Point Beach Units 1 and 2 to allow operation at either 2000 or 2250 psia (Reference 1). These changes include (1) defining over-temperature – ΔT-trip equation for each operating pressure, and (2) redefining the low pressure trip to allow adequate operating margin when operating at the lower pressure (2000 psia).

Although 2250 psia is the design operating pressure, both units have been previously operated at the lower pressure. A brief history of the previous operation of Point Beach Units 1 and 2 is given by the licensee in References 1 and 7 outlining the reasons for changing the pressure, the dates at which these changes were made and providing the references to the various Amendment requests for NRC and the subsequent Staff Safety Evaluation Reports. Presently both units are operating at 2250 psia. The licensee requested the change to permit operation at 2000 psia to reduce stress on the steam generator tubes.

This change to a lower pressure adversely affects the departure from nucleate boiling ratio (DNBR) and requires justification that the reactor is still adequately protected. The proposed change in the over temperature - ΔT(OTΔT) trip provides this protection for some cases. For situations where the OTΔT trip does not operate, adequate protection must be shown by other analysis. The loss of flow and rod drop events are two events in which DNBR protection is provided by means other than the OTΔT trip.

Modification of the reactor low pressurizer pressure trip to provide more margin between the lower operating pressure and this trip also requires justification that the applicable criteria for transient and accident anslyses are still satisfied.

Background

In the Confirmatory Order for Modification of License dated November 30, 1979 (Order) certain requirements were made pertaining to the operation of Point Beach, Unit 1. In the Safety Evaluation appended to that Order certain remedial actions were discussed. Among these remedial actions we noted that the licensee planned to operate the facility at the reactor coolant pressure of 2000 psia rather than at 2250 psia to reduce the internal pressure stresses of operation by about 15% during operation (Action No. 3, p. 15). This action was to be initiated upon NRC approval of an amendment request dated November 2, 1979 which requested permission to operate at that pressure. In the same Safety Evaluation we discussed "Measures for Reducing the Rate of Degradation" on pp 22 and 23. We indicated that the acceptability of this proposed operation would be addressed separately. That Safety Evaluation is incorporated into this Safety Evaluation by reference.

The Order of November 30, 1979 was based on information resulting from the steam generator tube inspection of October 1979. On December 11, 1979 another steam generator leak occurred. An eddy current test was performed on both steam generators which resulted in eddy current indications below the tube sheet (in the tube crevice) in both steam generators. Twenty tubes were plugged in steam generator A and fifteen tubes were plugged in steam generator B. Since there appears to be evidence of continuing intergranular corrosion attack the NRC Staff has now found that is not only desirable, but prudent and necessary, to take immediate action to require the reactor coolant pressure to be reduced from 2250 psia to 2000 psia since this will have the effect of substantially reducing the differential pressure across all tubes in both steam generators.

As explained below, operation of Unit 1 at a reactor coolant pressure of 2000 psia is acceptable from an accident analysis point of view. The applicable criteria for transient and accident analysis are still satisfied.

The licensee has withdrawn the amendment request and has made a commitment to operate the unit at a reactor coolant system pressure of 2000 psia only (Reference 8).

Evaluation

Overtemperature T Trip and Low Pressurizer Pressure Trip

For condition 1 (normal operation) and condition 2 (anticipated transients) events (i.e. where overtemperature trip is required) the fuel rods must be protected from overheating by maintaining the departure from nucleate boiling ratio (DNBR) above the safety limit of 1.3. The primary method of doing this is by means of the overtemperature - ΔT trip. This trip is a function of pressure and is also a function of the value assumed for the low pressurizer pressure trip as explained in Reference 2. Reducing the low pressurizer pressure trip for 2000 psia operation from 1865 psig to 1790 psig would allow more operating margin between the lower operating pressure (2000 psia) and the low pressurizer pressure trip. The licensee provided an equation for the overtemperature - ΔT trip applicable to operation at 2000 psia.

In 1973, the licensee proposed operation at 2000 psia. Justification for this was presented in Reference 3. The staff approved operation at 2000 psia and the corresponding overtemperature - ΔT equation in Reference 4. As can be seen from Table 1 the currently proposed 2000 psia equation for the overtemperature ΔT does not result in a significant decrease in margin to DNB when compared to the previously approved equation for 2000 psia.

Also, as shown in Table 1, the values of the trip* are almost the same at 2250 psia. This results in a gain in DNB margin at the higher pressure since the trip values remain almost the same while the pressure increased 250 psia, from 2000 psia to 2250 psia. Increasing pressure under PWR conditions results in increased margin to DNB. Therefore, even though the higher pressure would have justified a high trip value, the value was kept the same.

*The values shown in the table are normalized to full power delta - T.

TABLE 1

T_{AVG} (°F)	WCAP 8151 (Reference 3) 2000 psi	Present Tech Spec 2250 psi	Present Evaluation 2000 psi
550	1.47	1.48	1.465
560	1.33	1.33	1.315
570	1.16	1.18	1.165
578	1.029	1.06	1.045
580	0.9978	1.03	1.015
590	0.839	0.88	0.87

As discussed in the next section, the licensee also reviewed the Condition 2 events which trip on the overtemperature ΔT trip and found that the DNBR=1.3 safety limit is not exceeded with the new overtemperature ΔT equations.

Based on the fact that the proposed overtemperature ΔT trip equation at 2000 psia gives values which have not changed significantly from the values previously approved by the staff for operation at 2000 psia and the fact that a review of Condition 1 and Condition 2 events (the only events to which the overtemperature ΔT trip applies) shows that the DNBR=1.3 safety limit is not exceeded, we find the new overtemperature ΔT equation to be acceptable.

Transient and Accident Analyses Affected by Lower Operating Pressure

The licensee has also reviewed the postulated accident events in the FSAR using the methods described in Reference 5, known as the Westinghouse Reload Methodology, to determine the effect of reduced pressure operation on the plant transients and accidents. This review determined that several of these events needed to be reanalyzed. These events are listed in Table 2.

TABLE 2

Accidents Re-Analyzed For Low Pressure Operation

Rod Ejection

Loss of Flow

Locked Rotor

Rod Withdrawal at Power

WCAP 8151 (Reference 3) gives a qualitative discussion of the impact of 2000 psi operation on the transient and accident analyses. These conclusions are, in general, still valid.

The low pressurizer pressure trip is important in the small break LOCA. The value assumed for this trip in the analysis is 1795 psig which is above the low pressure trip being proposed for 2000 psia operation. The licensee stated that the analysi is still conservative because the reduction in pressure from 2250 psia to 2000 psia more than offsets the slight (5 psi) change in low pressurizer pressure setpoint. For example, the licensee states that at 2250 psi and 1795 psi low pressurizer pressure trip there would be 3.8 full power seconds before trip while at the lower operating pressure of 2000 psia whit the corresponding low pressurizer pressure trip of 1790 psig, only 0.8 full power seconds would result in the case of the worst small break.

The Large Break Loss of Coolant Accidnet (LOCA) was also reanalyzed at 2000 psia (and 18% steam generator tube plugging) to justify operation at the lower pressure (Reference 5). Only the limiting break size (a DECLG, C_D=0.4) was reanalyzed. This is acceptable since the change in peak cladding temperature is relatively small and the reactor pressure would not be expected phenomenologically to have a large effect.

The results of the LOCA analysis for both 2000 psia and 2250 psia are given in Table 3.

TABLE 3

Results of LOCA Analysis for Point Beach Unit 1 for 2000 psia and 2250 psia

	2000 psia	2250 psia
Peak Clad Temperature (°F)	2062	2053
Maximum Local Clad/Water Reaction (%)	5.11	5.3
Total Core Clad/Water Reaction (%)	$<$0.3	$<$0.3

The overpower - ΔT trip which provides protection against fuel centerline melting is derived in such a way that it is not a function of reactor coolant system (RCS) pressure of the low pressurizer pressure reactor trip (it is a function of the high pressurizer pressure reactor trip). It is therefore unaffected by the change in pressure.

Summary

The staff has reviewed the change to operate Unit 1 at a reactor coolant pressure of 2000 psia and finds it acceptable based on two points. The first is that the licensee, using the standard Westinghouse reload methods (Reference 6), has verified that Point Beach Unit 1 would still meet the applicable safety criteria. The second point is that no significant reduction in margin has been made in the overtemperature - ΔT set point over that previously approved by the staff. While this second point was not essential to acceptability of the proposed change, it does provide additional assurance of safe operation.

The Safety Evaluation appended to the November 30, 1979 Confirmatory Order for Modification of License considered the reduction of reactor coolant pressure to 2000 psia as one of the licensee's proposed actions to reduce the rate of steam generator tube degradation (p. 15 and p. 22). The staff indicated that the acceptability of this proposal would be addressed separately (p. 23) and further discussed the other components that could be affected. The staff concluded that the remedial actions proposed by the licensee will mitigate the effects of postulated accidents and retard the rate of corrosion (p. 24).

We have now completed the review of the licensee proposal to operate at 2000 psia and find; 1) from the view of the inter-related operating considerations the reduction in pressure is acceptable, 2) from the view of steam generator tube degradation it is prudent to reduce that degradation as much as possible. The reduction of the reactor coolant pressure was one of the licensee proposed actions to reduce steam generator tube degradation and was postponed only to permit a complete review of the interrelation of other systems. Now that we have concluded that the reduction in pressure produces no problem in other operating parameters or systems, it is prudent and necessary that this reduction in pressure be accomplished as soon as possible.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) operation at a reactor coolant system pressure of 2000 psia is required to provide continued assurance that the health and safety of the public will not be endangered, and does not involve a significant hazards consideration, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this requirement will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 3, 1980

References

1. Letter from S. Burstein, Wisconsin Electric Power Company to H. Denton U. S. Nuclear Regulatory Commission, dated November 2, 1979.

2. Ellenberger, S. L., et. al., "Design Basis for Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions, Westinghouse Electric Corporation." WCAP 9745, March 1977.

3. "Fuel Densification: Point Beach Nuclear Plant Unit No. 2 Low Pressure Analysis," Westinghouse Electric Corporation, WCAP 8151, June 1973.

4. (a) "Safety Evaluation by the Directorate of Licensing, Amendment No. 3 to Facility Operating License No. DPR-24, (Change No. 8 to Appendix A of Technical Specifications), Wisconsin Michigan and Wisconsin Electric Power Company, Point Beach Nuclear Plant Unit No. 1, Docket No. 50-266," May 23, 1974, transmitted by letter; Dennis L. Ziemann for Karl R. Goller to Mr. Sol Burstein, May 23, 1974.

   (b) "Safety Evaluation by the Directorate of Licensing, Supporting Amendment No. 5 to License No. DPR-27, Change No. 11 to the Technical Specifications, Wisconsin Electric Power Company and Michigan Power Company." September 30, 1974, transmitted by letter; Karl R. Goller to Sol Burstein, September 30, 1974.

5. Letter from C. W. Fay, Wisconsin Electric Power Company to H. Denton, U. S. Nuclear Regulatory Commission, dated November 27, 1979.

6. Bordelon, F., et. al., "Westinghouse Reload Safety Evaluation Methodology," Westinghouse Electric Corporation, WCAP 9272, March 1978.

7. Letter from S. Burstein, Wisconsin Electric Power Company to H. Denton, U. S. Nuclear Regulatory Commission, December 19, 1979.

8. Letter from S. Burstein, Wisconsin Electric Power Company to H. Denton U. S. Nuclear Regulatory Commission, December 31, 1979.

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

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USNRC
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Office of the Secretary
Docketing & Service
Branch

April 4, 1980

Docket No. 50-266

Mr. Sol Burstein
Executive Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

ATTACHMENT 5

Dear Mr. Burstein:

Enclosed is a signed original Order dated April 4, 1980, issued by the Commission for Point Beach Nuclear Plant Unit No. 1. The Order requires that testing be performed within 90 effective full power days.

With these additional limits, we have concluded that there is reasonable assurance that the public health and safety will not be endangered by the continued operation of Point Beach Unit No. 1. The basis for this conclusion is contained in our Safety Evaluation Report which is appended to the Order.

A copy of the Order is being filed with the Office of the Federal Register for publication.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosure:
Confirmatory Order

cc: w/enclosure
See next page

Copies to Messrs Miller
Rosber / Hinkens
Fay
Peter V

4-9-80

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cc: Mr. Bruce Churchill, Esquire
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of WISCONSIN ELECTRIC POWER COMPANY (Point Beach Nuclear Plant, Unit 1)	)))))	Docket No. 50-266

MODIFICATION OF NOVEMBER 30, 1979 ORDER

I.

Wisconsin Electric Power Company (the licensee) is the holder of Facility Operating License No. DPR-24 which authorizes the licensee to operate the Point Beach Nuclear Plant, Unit 1, located in Two Creeks, Wisconsin, under certain specified conditions. License No. DPR-24 was issued by the Atomic Energy Commission on October 5, 1970 and is due to expire on July 25, 2008.

II.

Inservice inspections of the Point Beach Unit 1 steam generators performed during the August 1979 and October 1979 outages indicated extensive general intergranular attack and caustic stress corrosion cracking on certain of the external surfaces of the steam generator tubes. As a result of information provided in discussions with the licensee and its representatives, which is documented in a letter dated November 23, 1979 from S. Burstein to H. R. Denton, and the Staff's Safety Evaluation Report, dated November 30, 1979, on Point Beach Unit 1, Steam

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Generator Tube Degradation due to Deep Crevice Corrosion, it was determined that additional operating conditions would be required to assure safe operation prior to resumption of operation of Unit 1 from the 1979 refueling outage.

III.

The licensee in letters dated November 29, 1979 and November 30, 1979 agreed to additional conditions which were necessary to provide reasonable assurance for safe operation of Unit 1. On November 30, 1979, an Order was issued to impose limiting conditions on continued operation of Unit 1 for a period of 60 effective full power days, at which time the licensee was required to shut down until the Director of Nuclear Reactor Regulation determined in writing in accordance with condition 6 of the Order that the results of the eddy current tests required by the Order were acceptable. On February 28, 1980, Unit 1 was taken out of service for the tests required by the Order. On March 28, 1980, the licensee provided the results of such tests to the NRC.

In accordance with condition 6 of the November 30, 1979 Order the NRC staff has reviewed the licensee's March 28, 1980 submittal and has assessed whether continued operation of the facility would be safe. I have found for the reasons given in the attached Safety Evaluation that the public health, safety and interest requires that Unit 1 be shut down and certain tests be conducted within 90 effective full power days of operation after the date of this Order. The licensee has agreed to this condition. Subject to this condition and with continuation of the other conditions set forth in the November 30, 1979 Confirmatory Order and the January 3, 1980 Modification of the Order, I have concluded that there is reasonable assurance that the public health and safety will not be endangered by the continued operation of Point Beach Unit 1.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT the November 30, 1979 Confirmatory Order for Modification of License be amended, effective immediately, to delete condition 1 of Section IV of that Order and replace such condition with the following condition.

1. Within 90 effective full power days from the date of this Order, a 2000 psia primary-to-secondary hydrostatic test and a 800 psia secondary-to-primary hydrostatic test shall be performed. Also during this plant outage, an eddy current examination shall be performed on tubes in each steam generator. The program shall be submitted to the NRC for staff review and require examination of about 1000 tubes in the central region of the hot leg and three (3) percent of all tubes outside this central region and 3% of all cold leg tubes. The central region shall encompass all areas where deep crevice corrosion has previously been observed.

All other conditions of the November 30, 1979 Confirmatory Order and the January 3, 1980 modification of that Order, including condition 6 requiring that the licensee not resume operation after the required eddy current examinations until the Director, Office of Nuclear Reactor Regulation, determines in writing that the results of such tests are acceptable, remain in effect in accordance with their terms.

V.

Copies of the above referenced documents are available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. 20555, and are being placed in the Commission's local public document room at the Document Department, University of Wisconsin, Steven's Point Library, Stevens Point, Wisconsin 54451.

VI.

Any person whose interest may be affected by this Order may within twenty days of the date of this Order request a hearing with respect to this Order. Any request for a hearing shall be addressed to the Director of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555 with a copy to the Executive Legal Director at the above address. If a hearing is requested by a person who has an interest affected by the order, the Commission will issue an order designating the time and place of hearing. Any such request SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

In the event a hearing is held, the issues to be considered at such hearing shall be:

1. Whether the facts stated in Sections II and III of this Order provide an adequate basis for actions ordered; and

2. Whether the license should be modified to include the conditions set forth in Part IV of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Attachment:
Staff Safety Evaluation Report
dated April 4, 1980

Effective Date: April 4, 1980
Bethesda, Maryland

SAFETY EVALUATION REPORT RELATED TO POINT BEACH UNIT 1 STEAM GENERATOR TUBE DEGRADATION DUE TO DEEP CREVICE CORROSION

April 4, 1980

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INTRODUCTION

In accordance with the Confirmatory Order dated November 30, 1979, Point Beach Unit 1 was shutdown on February 29, 1980 for steam generator hydrostatic testing and eddy current inspection after having completed the authorized operating period of sixty (60) effective full power days (EFPD's) since the restart subsequent to the October 1979 steam generator inspection. The evaluation herein provides an update of the SER issued in support of the Confirmatory Order to reflect the operating experience at Unit 1 since the Order was issued, and the results of the steam generator inspection obtained during the February 29, 1979 outage. The background information and results of two consecutive inspections (August and October, 1979) as discussed in the November 30, 1979 SER are incorporated into this evaluation by reference.

BACKGROUND

CONFIRMATORY ORDER DATED NOVEMBER 30, 1979

Inservice inspections of the Point Beach Unit 1 steam generators performed during the August and October 1979 outages indicated extensive general intergranular attack (IGA) and stress corrosion cracking on the external surfaces of the steam generator tubes within the thickness of the tubesheet (generally referred to as "deep crevice corrosion"). In view of these findings and of the apparent high rate at which this corrosion phenomenon was developing, the licensee agreed to certain conditions to assure safe operation of Unit 1 for a period of sixty (60) effective full power days. This commitment was formalized by a Confirmatory Order dated November 30, 1979, amending the Operating License to include, in part, the following conditions:

1. a) Hydrostatic testing to be performed within 30 EFPD's.

   b) Hydrostatic testing and eddy current inspection within 60 EFPD's. Submittal of the proposed eddy current inspection program for NRC staff review. Eddy current inspection results also to be submitted, with no resumption of power until the Director, Office of Nuclear Reactor Regulation determines in writing that the results are acceptable.

2. More restrictive limits on primary to secondary steam generator leakage.

3. More restrictive limits on primary coolant activity.

4. Unit 1 not to be operated with more than 18% of tubes plugged in either of the steam generators.

While not covered under terms of the Confirmatory Order, the licensee implemented additional measures in an attempt to retard further tube degradation. These measures included 1) a crevice flushing program to remove harmful chemicals from the tubesheet crevices, 2) reduced operating temperature and pressure, 3) continued close surveillance of feedwater chemistry and condenser tube leakage, and 4) sludge lancing to be performed within 12 months of the return to power.

DEFECTS AT OR ABOVE TUBESHEET

The Safety Evaluation issued in support of the November 30, 1979 Confirmatory Order reflected the staff's understanding that the extensive degradation observed during the August and October 1979 inspections involved general intergranular attack and cracking within the tubesheet crevices, exclusively. Subsequent to the Confirmatory Order, however, the staff became aware of five (5) tubes with defect indications at or above the tubesheet which had not been addressed in the November 30 SER.

In response to our request, the licensee submitted by letter dated December 21, 1979 additional details regarding the defects in these five tubes and an evaluation of their significance. The licensee reviewed the single frequency eddy current test results since 1975 for the subject five tubes and compared the signals of these past inspections to the same frequency signal obtained during the multi-frequency inspection in October 1979. This comparison showed that the signals have not changed through three or four inspections since 1975. On the basis of this review the licensee concluded that the defects observed in October 1979 at or above the tubesheet have remained essentially unchanged since at least 1975 and occurred as a result of earlier thinning or cracking rather than to the intergranular attack phenomenon currently being experienced in the tubesheet crevice area and which was only first observed in November, 1977.

In response to our request, the licensee submitted by letter dated December 21, 1979 additional details regarding the defects in these five tubes and an evaluation of their significance.

Based upon our review of this submittal and a subsequent conference call with the licensee on December 22, 1979, we concluded that (1) the eddy current indications at or above the tubesheet, which were observed during the October 1979 inspection, are old defects, possibly due to wastage or stress corrosion cracking, which were active mechanisms in 1975 and earlier, (2) these indications are not related to the active phenomenon of general intergranular attack and cracking currently being experienced in the tubesheet crevices, and (3) the staff conclusions set forth in the November 30, 1979 SER remained valid and that the unit could continue to be safety operated under terms of the Confirmatory Order. Nonetheless, we have continued our investigation into the significance of the defects found at or above the tubesheet, particularly with regards to eddy current capabilities to detect these defects and their safety significance. This matter is addressed in further detail in this evaluation.

OPERATING EXPERIENCE SUBSEQUENT TO THE CONFIRMATORY ORDER

Following the issuance of the Confirmatory Order, Point Beach Unit 1 was returned to power on December 1, 1979. On December 11, 1979, Unit 1 experienced a rapid increase in primary to secondary leak rate, to 260 gpd, and was forced to shutdown under terms of the Confirmatory Order. The source of the leak was identified as one leaking tube and two leaking plugs in steam generator B. Although not required by either the Technical Specifications or the Confirmatory Order, the licensee performed multifrequency eddy current examinations in both the A and B steam generators. A total of approximately 1900 tubes were inspected. The inspection bounded all areas of previously observed deep crevice corrosion by at least one row and column of tubes. The inspection boundaries were expanded when new indications were observed near the boundary. A set of randomly selected tubes outside the boundaries were also inspected. Representatives from the NRC staff and consultants were at the site on December 16, 1979 to observe the inspection in progress. As a result of this inspection, twenty (20) tubes were plugged in steam generator A and fifteen (15) tubes were plugged in steam generator B. None of the observed indications occurred at or above the top of the tubesheet. The inspection program and results were formally documented in Licensee Event Report 79-021/OIT-0 dated December 22, 1979.

Prior to resuming power operation, 2000 psid primary to secondary and 800 psid secondary to primary hydrostatic tests were performed. No tube failures or additional leakage resulted from these tests.

Based upon our review of the December 11 tube leak occurrence and the inspection results we concluded that the conclusions reached in the November 30, 1979, SER remained valid and that the operating restrictions imposed by the Confirmatory Order continued to provide adequate assurance of safe operation.

Point Beach Unit 1 was returned to power on December 22, 1979 and operated to the completion of its authorized 60 EFPD operating period (on February 24, 1980) with only a very minor, but equivalent to a constant 30 gpd primary to secondary leak. This was within the trace amount of equivalent leakage normally experienced at this unit.

MARCH 1980 INSPECTION RESULTS

FIELD EDDY CURRENT TESTING

The eddy current testing (ECT) program implemented during the March 1980 steam generator inspection was submitted for NRC staff review by letter dated February 26, 1980. This program was modified to incorporate NRC staff comments. ECT of 100% of the tubes in regions of previously observed deep crevice corrosion activity (including the kidney shaped central bundle region) was performed within boundaries bounding previously observed defects by at least one tube row and column. Where defects were observed to occur at the boundary, the inspection was expanded to bound these defectives by one tube row and column. An additional 3% random sample was inspected on the cold leg side and also among tubes on the hot leg side in areas not being 100% inspected. Representatives of the NRC staff were on site during the inspection to monitor the inspection as it proceeded, and to facilitate timely decisions from NRC/NRR regarding the need for additional inspection or tube pulling for laboratory examination.

Multifrequency eddy current testing (ECT) conducted in accordance with the approved program revealed 18 defect indications on the hot leg side in steam generator A and 24 defect indications on the hot leg side in steam generator B. In addition, 3 tubes in S.G. B and 6 tubes in S.G. A were found with undefinable indications within the tubesheet. On March 31, a hydrostatic test conducted after the ECT inspection revealed two tubes leaking at approximately 2 drips/minute and two wet plugs in S.G. B. Following plugging of these tubes and repair of the wet plugs a second hydrotest revealed another leaking tube in S.G. B which was plugged. Table I summarizes the ECT indicated defect depths in the two steam generators. Table II summarizes the elevation of the defect indications above the lower, primary surface of the tubesheet which is about 23 inches thick. Some defects affected several inches of tube length and one tube had indications running from the tube expansion at the primary surface of the tubesheet to approximately one inch below the upper, secondary tubesheet surface. The elevations indicated in Table II are the highest elevations reached by each defect.

TABLE I ECT INDICATED DEFECT DEPTHS		
DEFECT DEPTH IN PERCENT OF TUBE WALL	NUMBER OF TUBES	
	S.G. A	S.G. B
90 to 100	5	3
80 to 89	7	7
70 to 79	2	7
60 to 69	3	3
50 to 59	-	2
40 to 49	1	2

TABLE II ELEVATION OF ECT DEFECT INDICATIONS		
DISTANCE ABOVE THE PRIMARY TUBESHEET SURFACE (INCHES)	NUMBER OF TUBES	
	S.G. A	S.G. B
0-4	-	1
5-9	-	2
10-14	2	2
15-19	8	6
20-21	8	12
1/2" ABOVE SECONDARY T.S. SURFACE	-	1

No defective tubes were discovered outside of the central bundle region on the hot leg side nor anywhere on the cold leg side of either steam generator.

Tables I and II in Appendix I provide a tube by tube evaluation of ECT indicated defect depths and elevations and results of re-evaluations of ECT tapes from previous inspections for each defective tube. Study of these tables reveals that 15 tubes in steam generator A and 4 tubes in steam generator B had the same ECT indications but were overlooked in either the December or the December and October 1979 inspections. All of the tubes with defect indications were plugged except those that were removed for laboratory examination. All the ECT indications were of small amplitude and indicate very small volume defects.

TUBE PULLING AND LABORATORY EXAMINATIONS

In their February 26, 1980 submittal the licensee committed to remove a tube from the Unit 1 steam generators if one was found with an eddy current testing indicated defect at or above the top of the tubesheet, such as were observed in five tubes during the October 1979 inspection. The primary interest in removing this type of tube was two fold: (1) to determine if the intergranular attack occurring within the tubesheet crevices is resulting in tube degradation at or above the upper secondary surface of the tubesheet and (2) to correlate field ECT with laboratory examination of the defects. As indicated in Table II one tube was discovered in steam generator B with an indication approximately 1/2" above the top of the tubesheet. This was tube R19-C37 and the indication was 58% deep. In accordance with their commitment, this tube was removed from the steam generator for laboratory examination. In addition, the NRC (after a review of the ECT results) required removal of two other tubes for laboratory examination. These were tubes R30-C41 which had a 47% indication approximately 21" above the primary face of the tubesheet and tube R26-C53 which had a 86% indication approximately 18" above the primary face of the tubesheet. Removal of these tubes was intended to provide additional data regarding the extent and magnitude of IGA and the accuracy of ECT. The tube removal procedures extended the outage time approximately six days and resulted in approximately an additional 155 manrem exposure.

LABORATORY RADIOGRAPHY AND EDDY-CURRENT TESTING

Radiography and ECT were performed on all three of the removed tube specimens by Westinghouse at their Pittsburgh R&D facility.

As a result of the pulling process the original 22.1/2" length of tube R30-C41 within the tubesheet was elongated to approximately 24-3/4". This measurement was based on the ring left on the tube at the top of the tubesheet. Radiography of the removed tube revealed many defect indications in the region up to 23-1/4" from the tube end. Many ECT indications existed up to 23-1/2" from the tube end. No radiographic or ECT indications existed at or above the ring marking the top of the tubesheet.

The laboratory ECT examination indicated an approximately 70 to 80% defect based on evaluation of the single frequency (400 KHZ) signal, located 23-1/2" from the tube end. Based on the elongation caused in the tube removal process, 23-1/2" corresponds to approximately 21.3" from the tube end in the unstrained tube.

The field ECT indicated a 47% defect at 400 KHZ approximately 21" from the tube end. Field evaluation of the defect based on the multi-frequency signal estimated the defect depth in the same 70% to 80% range as obtained in the laboratory (at 400 KHZ) in the absence of tubesheet interference effects. Defect depths are reported based on the single frequency signal when possible since it is the technique currently approved by the ASME Code.

The pulling of tube R26-C53 elongated the original 22.5" of tube in the tubesheet crevice to approximately 25-7/16". Radiography of the removed tube revealed many defect indications in the region up to approximately 19.8" from the tube end as well as a single defect 25" above the tube end. Eddy current testing revealed many defect indications up to 19.8" from the tube end. Eddy current testing also revealed two 90% defects located approximately 7/16" and 2-7/16" below the tubesheet ring. No radiographic or ECT indications existed at or above the ring marking the top of the tubesheet.

None of the above laboratory ECT indications for tube R26-C53 were specifically identified in the field. Some of the indicated defects may have been introduced or made worse during the tube pulling operation. "Squirrel" indications (minor disturbances in the ECT signal of underterminable origin) were observed in the field over the full length of tube within the tubesheet. It was not possible to verify through laboratory ECT the 86% ECT indication observed in the field 18" above the tube end, since this corresponded to one of the locations where the tube broke during pulling. However, this field ECT indication will be compared with the results of the fractography analysis of the fracture surface as part of a detailed report which the licensee has committed to submit by April 30, 1980.

Tube R19-C37 was of particular interest because of the field ECT indication of a 58% defect located approximately 1/2" above the tubesheet. Unfortunately, when the tube was examined there was no ring clearly indicating the top of the tubesheet as there was on the other two tubes which were removed. Since the section of tube within the tubesheet experiences a different load and elongation during the removal process than the section of tube above the tubesheet, the exact location of the top of the tubesheet relative to the tube cannot be directly quantified.

Radiography and ECT of the removed tube revealed many defect indications in the region up to 23.75" from the tube end. Radiograpny also showed crack like indications approximately 24-3/8" above the tube end and ECT indicated an approximate 60% defect 24-1/2" above the tube end. No ECT indications were observed above the 60% indication.

Although the 60% laboratory ECT indication corresponds well with the 58% field ECT indication, its elevation cannot be directly correlated to the field indications because the location of the top of the tubesheet is not identifiable. Calculations based on strains in the other tubes which were removed indicate that this defect would have been inside the tubesheet. Nonetheless, it is the defect with the highest elevation in the tube, its depth corresponds well to the field ECT depth and it could be the defect of interest given the non-uniform straining of the tubes during removal.

Metallographic Examinations

Metallographic examination consisted primarily of photomicrographs (PM) to determine at what elevation IGA existed in the tubes.

For tube R30-C41 PMs were prepared for sections centered on the top of the tubesheet and approximately 0.35" below and 0.45" above the top of the tubesheet. In each of these regions PMs of 50 and 200 power magnification were made. The 200 power PMs were centered on the region in the 50 power photomicrographs indicating the greatest surface irregularities. For the section of tube below the top of the tubesheet the PMs showed shallow grain boundary separation on the order of 0.0025" maximum. At the top of the tubesheet, shallow surface separation was observed affecting grain boundaries to just over 0.001" in depth. Similarly above the top of the tubesheet surface separation of the grain boundaries was observed to a depth of approximately 0.001 inches. Extensive general IGA as is occurring deeper in the tubesheet crevice was not observed in any of these regions.

Photomicrographs were also prepared for tube R26-C53. Again the PMs were centered about the top of the tubesheet and approximately 0.4" below and 0.2" above the top of the tubesheet. The section below the top of the tubesheet showed shallow grain boundary separation penetrating approximately 0.002" maximum.

The region centered about the top of the tubesheet showed no grain boundary separation although some surface irregularities penetrating less than 0.001" existed. Above the top of the tubesheet some areas of grain boundary separation penetrating approximately 0.003" were observed. Extensive general IGA as is occurring deeper in the tubesheet crevice was not observed in any of these regions.

Five photomicrographs were made of tube R15-C39. One was centered on the 60% defect described earlier while the other four were centered approximately 1-5/8" and 3/4" below and 1" and 1-3/4" above the defect. The two sections below the defect showed IGA penetrating to depths of nearly 0.004". Photographs of the tube surface at the defect show a crack running less than approximately 1/2" longitudinally then turning and running less than approximately 1/4" circumferentially. Photomicrographs of a section made through the defect show a crack penetrating approximately 0.017" surrounded by localized IGA. The longitudinal section made for the PM may not have included the deepest section of the crack. Section D above the defect indicates one localized area of grain boundary separation approximately 0.001" deep and section E above the defect shows no grain boundary separation but some shallow surface irregularities less than 0.001" in depth.

PROPOSED CONDITIONS FOR CONTINUED OPERATION

The licensee has proposed the following conditions to allow continued operation of Point Beach Unit 1.

1. Within 90 EFPD, a 2,000 psid primary-to-secondary hydrostatic test and a 800 psid secondary-to-primary hydrostatic test will be performed. An eddy current examination consisting of about 1,000 tubes in the central region of the hot leg in each steam generator and 3% of the remaining tubes outside this area will be performed.

2. Primary coolant activity for Point Beach Unit 1 will be limited in accordance with the provisions of Sections 3.4.8 and 4.4.8 of the Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 2, July 1979, rather than Technical Specification 15.3.1.C.

3. Close surveillance of primary-to-secondary leakage will be continued and the reactor will be shutdown for tube plugging on confirmation of any of the following conditions:

a. Primary-to-secondary leakage of 150 gpd (0.1 gpm) in either steam generator;

b. Any primary-to-secondary leakage in excess of 250 gpd (0.17 gpm) in either steam generator; or

c. An upward trend (average over a three-day period) in primary-to-secondary leakage in either steam generator in excess of 15 gpd (0.01 gpm) per day, when measured primary-to-secondary leakage is above 150 gpd in that steam generator.

4. The reactor will be shutdown, any leaking steam generator tubes plugged, and an eddy current examin tion as described in Item 1., above, will be performed if leakage due to crevice corrosion in either steam generator exceeds the limits stated in Technical Specifications 15.3.1.D.

5. Unit 1 will be operated at a reactor coolant pressure of 2,000 psia with the associated parameters (i.e., overtemperature ΔT and low pressurizer pressure trip point) with the limits indicated in the Safety Evaluation Report appended to your letter of January 3, 1980.

On return to power operation, the licensee proposes to continue the following program to assist in retarding further tube degradation:

a. Unit 1 will be operated at a reduced reactor coolant system hot leg temperature.

b. Continue close surveillance of feedwater chemistry conditions and condenser tube leakage.

c. Perform sludge lancing within nine months of returning to power.

EVALUATION

ECT PROGRAM, RESULTS, AND CAPABILITIES

Members of the NRC staff and their consultant from Oak Ridge National Laboratory were on site during the inspection to review the testing and evaluation techniques. Eddy current testing examinations were conducted in accordance with the program proposed in the licensee's February 26, 1980 submittal and approved, with comment, by the NRC. This program bounded the areas where deep crevice corrosion was previously observed and was expanded in any areas where new indications were found. The random inspection of peripheral hot leg tubes and cold leg tubes revealed no deep crevice corrosion. Therefore, the inspection performed is adequate to ensure that the great majority of tubes with deep crevice corrosion have been removed from service by plugging.

The March 1980 ECT results show a marked reduction in the number of tubes with indicated defects compared to the August and October 1979 inspections. In addition, fifteen of the 24 ECT indicated defects in steam generator B and 6 of the 18 ECT indicated defects in steam generator A were shown to exist previously through reexamination of the ECT tapes from previous inspections. Thus, the number of new defects discovered in this inspection is smaller than the raw data indicates. The inspection results suggest that some of the remedial actions taken by the licensee following the October 1979 inspection, particularly the lower temperature operation, may be succeeding in retarding the rate of further deep crevice corrosion, especially since the time of the December 1979 outage.

As discussed in our November 30, 1979 SER the accuracy of the eddy current technique is somewhat diminished in the tubesheet region and cannot be fully relied upon to detect every tube degraded by deep crevice corrosion. This appears to be particularly true for tubes subject to general IGA, but which do not contain cracks. Partially through wall cracks of significance are generally detectable, even in the tubesheet region, with ECT. As experience has shown, however, very small volume defects which in turn produce very small amplitude ECT signals may be easily overlooked (as was the case with the 19 tubes above). Our evaluation of the safety significance of IGA and stress corrosion cracking occurring within the thickness of the tubesheet is discussed in our November 30, 1979 SER which is incorporated into this SER by reference.

With regard to the tubes observed during the October and March inspections to contain defects at or slightly above the top of the tubesheet, we have concluded that multifrequency ECT can detect defects of a significant size to threaten tube integrity during normal or postulated accident conditions. All of the defects discovered at or above the top of the tubesheet are small amplitude, small volume defects. Assuming the defects at or above the tubesheet to be wall thinning (wastage related), rough estimates of the size of the defects were made by the staff based on comparison with the ECT signatures from the ASME Code calibration standard. These estimates show that if these defects are wastage related, the volumes of these defects are very small compared to what is necessary to burst or collapse the tube under postulated accident conditions, as determined by independent tests sponsored by NRC (NUREG/CR-0718).

In the case of tube R19-C37 which exhibited a field ECT indication of 58% approximately 1/2 inch above the tubesheet, the laboratory examination indicates that the defect indication observed in the field is most likely a crack. NRC sponsored burst and collapse tests (NUREG/CR-0718) have been performed on free standing tubes with EDM notches (simulating a crack) of up to 85-90% (through wall) in depth. The results indicate the lower bound burst strength to exceed the maximum primary to secondary pressure differentials during normal operation or postulated accidents for notches (cracks) ranging to about 1 inch in length. It should be noted that the burst strength of a tube containing a crack defect slightly above or below the top of the tubesheet is considerably higher than for free standing tubes, because of the restraint against radial expansion of the tube provided by the tubesheet. The above tests indicated a collapse failure to be a much less limiting failure mode than a burst failure mode for free standing tubes during postulated accidents. Cracks of sufficient size to cause a burst or collapse failure under postulated accidents are considered by the staff to be well within the detectable capability of the multifrequency eddy current technique, regardless of the location of the crack relative to the top of the tubesheet.

Tube Removal and Laboratory Exam

Laboratory radiography and ECT confirm the position taken by the staff that general IGA may not be detectable in the crevice of the tubesheet until it is severe enough for preferential crack growth to occur. Detection of defects below the top of the tubesheet by laboratory examinations is due partly to increased capability of ECT without the influence of the tubesheet and partly to the creation of new or the opening of old defects during the removal process. Laboratory radiography and ECT confirmed the absence of defects above the tubesheet in tubes R30-C41 and R26-C53. Unfortunately the top of the tubesheet could not be identified on tube R19-C37.

However, assuming that the upper most defect detected in the tube is the defect which was identified by field ECT, there is a good correlation between the laboratory and field ECT. More importantly, the defect which was detected was small enough so as not to jeopardize tube integrity. Primary-to-secondary and secondary-to-primary hydrostatic tests conducted on March 6 revea one tube (R23-C44) which exhibited a slight leak at a rate of 3 drips per minut and one wet plug in a previously plugged tube (R23-C50) both in S.G. B. No tube ruptures occurred. The defect found by ECT just above the tubesheet in tube R19-C37 in S.G. B withstood the simulated accident pressure differentials. This provides additional support to our previously stated conclusion that multifrequency ECT can detect defects at or above the top surface of the tubesheet which would jeopardize tube integrity during normal operating or postulated accident conditions.

The staff wants to emphasize that as inspection techniques with increased capabilities, such as multifrequency ECT, are developed, that many small volume defects which previously went undetected will now be found. These defects must be evaluated in the context of the magnitude of defects which jeopardize tube integrity during normal or postulated accident conditions. As inspection techniques become more capable, correspondingly more discriminate criteria must be established. Many plants which have not been inspected with multifrequency ECT are going to show new defects when multifrequency inspections are performed. These results must be dealt with rationally and requirements for tube inspection, plugging, and removal must be carefully applied.

METALLOGRAPHIC EXAMINATIONS

Members of the NRC staff and their consultant from Brookhaven National Laboratory met with representatives from WEPCO and their Westinghouse consultants in Pittsburgh on March 28, 1980 to review results of the metallographic examinations. Review of the photomicrographs described earlier revealed no general IGA similar to that occurring within the tubesheet crevice above the top of the tubesheet in tubes R26-C53 or R30-C41. Shallow grain boundary separation on the order of two grains or less existed on all photomicrographs of these tubes. Shallow grain boundary dissolution of this nature can result from several mechanisms including previous operating environments or tube pickling during manufacturing. This grain boundary separation is much less severe than that occurring within the tubesheet. The staff has concluded that the shallow grain boundary dissolution at and above the top of the tubesheet is not significant in terms of tube integrity. Metallographic examination of tube R19-C37 revealed stress corrosion cracking and shallow IGA of the tube near the top of the tubesheet. Re-evaluation of past ECT tapes showed that this defect existed as far back as 1976 but was overlooked using single frequency ECT. The nature of the crack is similar to that of stress corrosion cracks which occurred during previous operating periods. The staff believes that this is an old defect which has not significantly changed since 1976.

CONCLUSIONS

Based on the information presented above the staff has reached the following conclusions:

1) The inspection and tube plugging performed has been adequate to ensure the great majority of defective tubes have been removed from service.

2) Multiple frequency eddy current testing used to perform the inspection is capable of detecting defects near the tubesheet and tube support plate interfaces which would jeopardize integrity of the tube during normal operation or postulated accident conditions.

3) Hydrostatic tests simulating postulated accident conditions performed prior to returning to operation will identify any significant defects overlooked during ECT examination.

4) Intergranular attack at and above the top of the tubesheet as observed in the removed tube samples is extremely shallow and poses no threat to tube integrity at or above the top of the tubesheet.

5) Based on the number of new defects, the rate of deep crevice corrosion appears to have decreased.

6) A maximum 90 effective full power day operating period, prior to the next ECT inspection as proposed by the licensee, will provide adequate assurance that a large number of tubes will not simultaneously reach a point of incipient failure.

7) Remedial actions proposed by the licensee will continue to mitigate the effects of postulated accidents and retard the rate of corrosion.

The staff has determined that the following conditions should be required for continued operation:

1) Within 90 effecting full power days from the date of this order, a 2,000 psid primary-to-secondary hydrostatic test and 800 psid secondary-to-primary hydrostatic test shall be performed. Also during this plant outage, an eddy current examination shall be performed on tubes in each steam generator. The program shall require such examinations of about 1000 tubes in the central region of the hot leg, three (3) percent of all hot leg tubes outside this central region and 3% of the cold leg tubes. The Central region shall encompass all areas where deep crevice corrosion has previously been observed.

2) Primary coolant activity for Point Beach Nuclear Plant Unit 1 will be limited in accordance with the provisions of Sections 3.4.8 and 4.4.8 of the Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 2, July 1979, rather than Technical Specification 15.3.1.C appended to License DPR-24.

3) Close surveillance of primary to secondary leakage will be continued and the reactor will be shut down for tube plugging on detection and confirmation of any of the following conditions:

   a) Sudden primary to secondary leakage of 150 gpd (0.1 gpm) in either steam generator;

   b) Any primary to secondary leakage in excess of 250 gpd (0.17 gpm) in either steam generator; or

   c) An upward trend in primary to secondary leakage in excess of 15 gpd (0.01 gpm) per day, when measured primary to secondary leakage is above 150 gpd.

4. The reactor will be shut down, any leaking steam generator tubes plugged, and an eddy current examination performed if any of the following conditions are present:

   a) Confirmation of primary to secondary leakage in either steam generator in excess of 500 gpd (0.35 gpm); or,

   b) Any two identified leaking tubes in any 20 calendar day period.

   This eddy current program will be as described in item 1.

5. The NRC Staff will be provided with a summary of the results of the eddy current examination performed under items 1 and 4 above. This summary will include a photograph of the tubesheet of each steam generator which will verify the location of tubes which have been plugged.

6. The licensee will not resume operation after the eddy current examinations required to be performed in accordance with condition 1 or 4 until the Director Office of Nuclear Reactor Regulation has determined in writing that the results of such tests are acceptable.

These conditions are similar to those in the November 30, 1979 Order except that the approved operating period has been lengthened from 60 to 90 effective full power days, and no shutdown to perform hydrostatic tests are being required prior to the end of the 90 day period. These conditions differ from the licensees proposal in that the primary to secondary leak rate limits and requirements for ECT examination are more conservative.

On the basis of our review and evaluation, we conclude that continued safe operation of Point Beach Unit 1 may be permitted within the stated terms of the Confirmatory Order.

APPENDIX I ·
TABLE I

POINT BEACH #1 'A' S/G

Tube # R	C	% 1980	M.F. Dec. 1979	M.F. Oct. 1979			
12	19	80% 19-21" ATE	SAME R251 No change	SAME R651 N.C.			
7	22	29%/96% 12"ATE/17"ATE	SAME R251 N.C.	NDD/SAME 12"ATE/17"ATE R551			
18	22	66% 12-17" ATE	SAME R251 N.C.	NDD R551			
1	23	41% 20" ATE	NDD R251	R551			
7	24	83% 17"-20" ATE	MAYBE(?) NDD R251	NDD R551			
8	24	79% 17"-21" ATE	MAYBE(?) NDD R251	NDD R551			
25	45	69% 12"-20" ATE	Squirrels R351	NDD R851			
20	48	85% 21" ATE	SAME R251 N.C.	SAME R851			
9	49	90% 21" ATE	NDD R251				
17	50	85% 19" ATE	NDD R251				
19	50	97% 11" ATE	NDD R251				
20	50	97% 11" ATE	NDD R251				
12	59	87% 21" ATE	MAYBE(?) NDD R151	NDD R951			
12	61	83% 17" ATE	NDD R151				
14	63	83% 19" ATE	MAYBE(?) Squirrels R151				

POINT BEACH #1 'A' S/G

Tube # R	C	% 1980	M.F. Dec. 1979	M.F. Oct. 1979			
15	66	60% 18" ATE					
8	27	Squirrels 15-20" ATE	SAME R251 N.C.				
15	28	Squirrels 21" ATE	No Squirrels R251				
28	34	Squirrels 18-21" ATE	SAME R251 N.C.				
28	35	Squirrels 17" ATE	SAME R251 N.C.				
20	41	91% 19" ATE	NDD R351				
25	43	73% 17" ATE	SAME R351	Very S.V. N.D.D. R751			
11	46	Squirrels 12"-21" ATE	SAME R351				
29	52	Squirrels 14" ATE	SAME R151 N.C.				

APPENDIX I

TABLE II

B S/G INLET POINT BEACH #1

Tube # R	C	% 1980	M.F. Dec. 1979	M.F. Oct. 1979	S.F. Aug. 1979		
18	26	75% 18" ATE	SAME R151 No change	Changed R651	NDD R551		
13	26	73% 21" ATE	SAME R151 N.C.	SAME R651 N.C.	NDD R551		
13	33	71% 20" ATE	SAME R151 N.C.	Changed R651	NDD R552		
5	24	91% 11" ATE	SAME R151 N.C.	SAME R651 N.C.	NDD R552		
20	35	68% 21" ATE	SAME R151 N.C.	Changed R351	NDD R552		
8	37	89% 5" ATE	NDD R151				
19	37	58% 1/2" ATS	SAME 53% R151 N.C.	SAME R351 N.C.	NDD R552		
10	41	70% 21" ATE	SAME R251 N.C.	NDD R751	R651		
30	41	47% 21" ATE	SAME R251 N.C.	Some Change R751	NDD R651/R151		
30	42	48% 21" ATE	SAME R251 N.C.	Changed R751	NDD R151		
22	46	76% 15" ATE	SAME R251 N.C.	NDD R351			
24	48	84% 12" ATE	Changed R251	NDD R351	R652		
30	48	85% 21" ATE	SAME R251 N.C.	SAME R951 N.C	NDD R652		
25	49	84% 5" ATE	Changed R251	NDD R351	R652		
20	51	99%(?) 16" ATE	SAME R251 N.C.	NDD R351	R652		
23	54"	86% Full length	Squirrels some are new R251	SAME AS DEC. R351			

B S/G INLET POINT BEACH #1

Tube # R	C	% 1980	M.F. Dec. 1979	M.F. Oct. 1979	S.F. Aug. 1979		
23	57	56% 17" ATE	NDD R251				
21	58	83% 21" ATE	SAME R251				
14	59	75% 21" ATE	NDD R251				
21	63	62% 21" ATE	SAME R351	NDD R1051			
2	67	66% 21" ATE	NDD R351	R1051			
2	72	92% Top of Roil	SAME R351 N.C.	NDD R1051			
26	53	86% (New) 18" ATE					
30	43	Squirrels 21" ATE	SAME R251	SAME R751			
26	53	Squirrels Full T.S.	NDD R251				
25	55	Squirrels Full T.S.	NDD R251				
22	63	Squirrels 21" ATE	SAME R251	SAME R1051			
22	64	Squirrels 20" ATE	SAME R351	No Squirrels R1051			
25	55	74% (New) 15" ATE					

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
August 8, 1980

Docket No. 50-266

ATTACHMENT 6

Mr. Sol Burstein
Executive Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Burstein:

The NRC staff has reviewed the results of the Point Beach Nuclear Plant, Unit No. 1, steam generator tubes inspection which was submitted by letter dated August 5, 1980 in accordance with the April 4, 1980 Modification of November 30, 1979 Order. Based on the results of that review and for the reasons stated in the attached Safety Evaluation Report, we find that the results are acceptable and, thus, it is not necessary to place any further restriction on the resumption of operation of the Point Beach Nuclear Plant, Unit No. 1.

We would like to emphasize that all the conditions of the November 30, 1979 Order and the January 3, 1980 Modifications to the November 30, 1979 Order remain in effect in accordance with their terms.

It is our understanding that Unit 1 will be removed from service November 1980 for a refueling outage, and that during that outage, hydrostatic tests and eddy current examinations of 100 percent of all unplugged steam generator tubes, will be performed. Pursuant to 10 CFR 50.54(f) we request that you provide the NRC, within 15 days of your receipt of this letter, your plans and schedule for these steam generator tube inspections. Following the staff's review of your proposed plans and schedule, consideration will be given to the necessity of issuing an Order confirming your program or modifying your program to establish additional requirements.

If you have any questions on this subject, please contact us.

Sincerely,

Edson G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

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

SAFETY EVALUATION REPORT
RELATED TO POINT BEACH UNIT 1
STEAM GENERATOR TUBE DEGRADATION
DOCKET NO. 50-266

Introduction

In accordance with the Order dated April 4, 1980, Point Beach Unit 1 was shut down on July 25, 1980 for steam generator hydrostatic testing and eddy current inspection after having completed ninety (90) effective full power days (EFPD's) of operation since the restart following the March 1980 steam generator inspection. The evaluation herein provides an update of the SER's issued in support of the Confirmatory and Supplementary Orders, respectively, to reflect the recent operating experience at Unit 1 and the results of the August 1980 steam generator inspection. The background information and results of previous steam generator inspections as discussed in the November 30, 1979 and April 4, 1980 SER's are incorporated into this evaluation by reference.

Background and Discussion

Inservice inspections of the Point Beach Unit 1 steam generators performed during the August and October 1979 outages indicated extensive general intergranular attack (IGA) and stress corrosion cracking on the external surfaces of the steam generator tubes within the thickness of the tubesheet (generally referred to as "deep crevice corrosion"). In view of these findings and of the apparent high rate at which this corrosion phenomenon was developing, the licensee agreed to certain conditions to assure safe operation of Unit 1 for a period of sixty (60) effective full power days. This commitment was formalized by a Confirmatory Order dated November 30, 1979, amending the Operating License to include, in part, the following conditions:

1. a) Hydrostatic testing to be performed within 30 EFPD's.

   b) Hydrostatic testing and eddy current inspection within 60 EFPD's. Submittal of the proposed eddy current inspection program for NRC staff review. Eddy current inspection results also to be submitted, with no resumption of power until the Director, Office of Nuclear Reactor Regulation determines in writing that the results are acceptable.

2. More restrictive limits on primary to secondary steam generator leakage.

3. More restrictive limits on primary coolant activity.

4. Unit 1 not to be operated with more than 18% of tubes plugged in either of the steam generators.

While not covered under terms of the Confirmatory Order, the licensee implemented additional measures in an attempt to retard further tube degradation. These measures included: 1) a crevice flushing program to remove harmful chemicals from the tubesheet crevices, 2) reduced operating temperature and pressure, 3) continued close surveillance of feedwater chemistry and condenser tube leakage, and 4) sludge lancing to be performed within 12 months from the return to power.

In accordance with the Confirmatory Order, Unit 1 shut down on February 29, 1980 after having completed sixty (60) EFPD's of operation. The March 1980 eddy current results indicated a marked reduction in the number of tubes with indicated defects compared to the August and October 1979 inspections. By Order dated April 4, 1980, Unit 1 was required to be shut down for steam generator hydrostatic and eddy current inspections after ninety (90) EFPD's. With the exceptions that the operating period had been changed from 60 to 90 EFPD's, and that no shutdown to perform hydrostatic tests was required before the end of this period, the conditions of the Confirmatory Order remained in force under the April 4, 1980 Order.

July-August Steam Generator Inspection Results

Subsequent to the plant shutdown on July 25, 1980, both steam generators were subjected to hydrostatic tests and eddy current examination in accordance with the April 4, 1980 Order. A tubesheet inspection during the secondary to primary hydrostatic leak test revealed two "dripping" tube plugs and two "wet" tube plugs in the hot leg side of steam generator A, and one wet tube plug and one dripping tube (at rate of one drip per two minutes) in steam generator B. At the time of shutdown on July 25, the Unit 1 steam generators had been leaking (primary to secondary) at a very low level, approximately 20 gpd.

The dripping tube identified in steam generator B was inspected up through the U-bend using the multifrequency eddy current test (ECT) technique, but only a 46% through wall indication, located three inches above the tube end (within the tubesheet thickness), was identified. A possible explanation suggested by the licensee is that the source of the leak may be a small volume defect located in the transition region of the expanded tube zone (near the bottom of the tubesheet) which would be particularly difficult to discriminate, even with multifrequency ECT. This tube has subsequently been plugged.

The multifrequency ECT inspection program for both steam generators consisted of an examination of 100% of the tubes to the first support plate on the hot leg side, and 3% of the tubes inspected over their entire length (i.e. hot and cold leg). The results of these inspections are summarized as follows:

ECT Inspection Summary

S.G.		% Tubes Inspected	Eddy Current Indications	Elevation
S.G.A	Hot Leg	100%	1 tube - undefinable signal 3 tubes - <20% 3 tubes - 20 to 39% 7 tubes - 40 to 59% 5 tubes - 60 to 79% 9 tubes - 80 to 99%	Within thickness of tubesheet
			1 tube - 34%	Top of tubesheet
			1 tube - 34%	½" above tubesheet
	Cold Leg	3%	1 tube - 29%	½" above tubesheet
			5 tubes - <21%	1 to 2" above tubesheet
S.G.B	Hot Leg	100%	7 tubes - 40 to 59% 8 tubes - 60 to 79% 6 tubes - 80 to 99%	Within thickness of tubesheet
			1 tube - leaker	Unknown
	Cold Leg	3%	None	-

As seen in the summary Table, a total of 28 and 22 tubes in the hot leg of steam generators A and B, respectively, were identified to contain tubesheet crevice indications; i.e., indications located within the thickness of the tubesheet. The elevations of these indications range from three (3) inches above the tube ends to in excess of one inch below the top of the tubesheet. Two (2) additional tubes on the hot leg side of steam generator A were found to contain 34% small volume indications at the top of the tubesheet and one-half inch above the top of the tubesheet, respectively. Six (6) tubes on the cold leg side of steam generator A were indicated to contain minor through wall penetrations (<30%) located ½ to 2 inches above the top of the tubesheet elevation. No cold leg tubesheet crevice indications were identified which is consistent with previous experience.

Eddy current tapes from previous inspections dating back to October 1979 are being reviewed by the licensee for each of the tubes found during this inspection (July-August 1980) to contain eddy current indications. For some tubes, the licensee determined that small volume indications were probably present (but were not identified by the data evaluators) in one or more previous inspections by reviewing the previous tapes in close detail over the specific area of interest. These include ten (10) of the total of 50 tubes identified during this inspection as containing tubesheet crevice indications, and two (2) tubes in the hot leg of steam generator A found to contain indications at one-half inch above the top of the tubesheet. It is the licensee's evaluation that the eddy current data evaluators were unsuccessful in discriminating these small volume defects because of the low signal-to-noise ratio of the eddy current signal during previous inspections. However, the licensee's review of the previous eddy current tapes has established that the majority of the eddy current indications were not previously detectable.

The previous inspection in March 1980 included a 100% sample of tubes in the central bundle region ("Kidney zone") of each steam generator (approximately 1000 tubes), and a 3% random sample inspection outside this zone. The central region where 100% inspection was performed was defined to encompass the region of previously observed activity. However, the results of the latest inspection revealed 24 tubesheet crevice indications located up to several tubes beyond the boundary of this previously defined zone that were not inspected in March 1980. As indicated to the staff during discussions held on August 6, 1980, the licensee does not consider these results to be unexpected since the concentration of chemicals in the tubesheet crevices will occur regardless of whether there is a sludge pile at the surface. The licensee believes that, while the sludge pile may contribute chemicals for concentration in the crevice, there is no reason to believe that the crevice corrosion will be limited to the kidney zone, since chemicals from the bulk water will also concentrate in the tubesheet crevices.

All 50 tubes with indications in the tubesheet crevice, including the leaking (dripping) tube, have been mechanically plugged. In addition, three tubes were inadvertently plugged. The two dripping plugs in steam generator A were weld repaired and the steam generator was subsequently and successfully hydrostatically leak checked. The two tubes in steam generator A containing 34% indications outside the tubesheet crevice region were left unplugged since these indications are less than the 40% Technical Specification plugging limit and these indications appear to have remained unchanged since at least October 1979. The licensee has committed to re-examining these tubes during the next eddy current inspection.

To date, approximately 12.2% of the total number of steam generator tubes at Unit 1 have been plugged, which is well within the 18% tube plugging assumed in the LOCA-ECCS analysis for this unit.

Plans For Continued Operation

Based upon the results of this inspection, the licensee has concluded that the condition of the Point Beach Unit 1 steam generators has not changed significantly since the previous inspection in March 1980. The licensee plans to return Unit 1 to service for an additional 90 effective full power days until its scheduled refueling outage in early November 1980.

Evaluation

The July 1980 inspection of 100% of unplugged tubes at the completion of 90 effective full power days (EFPD) has satisfied the requirements of NRC's Confirmatory Order, dated April 4, 1980. The 2000 psi primary-to-secondary hydrostatic test and 800 psi secondary-to-primary test required by the April 4 Order confirmed that no tubes had reached a state of degradation that would cause a sudden primary-to-secondary leakage during the 90 EFPD operation.

The current multifrequency ECT results, compared with similar ECT results performed in March 1980 and December 1979, do not indicate an appreciable increase in tube degradation within the tubesheet crevice. Of the 14 tubes in steam generator A containing ECT tubesheet crevice indications and which were previously examined in March 1980 and December 1979, nine had tubesheet crevice defects which did not show an increase in defect size. Regarding the five (5) tubes that now show a significant ECT indication, but did not show indications previously, we believe that intergranular corrosion attack existed which could not be identified in previous inspections. This had been demonstrated in the laboratory analysis of tubes pulled in March 1980 and November 1979.

With regard to the 18 tubes in steam generator A with new ECT indications within the tubesheet crevice, it should be noted that the current inspection is the first time that 100% of the tubes in steam generator A have been examined for tubesheet crevice defects since October 1979. Thus, there is no basis to indicate that identification of these new tubes reflects a rapid deterioration of the Point Beach Unit 1 steam generators.

The above analysis applies also to the ECT indications found with the tube-sheet crevices of steam generator B.

The current ECT results for both steam generators show that intergranular corrosion attack has not progressed above the tubesheet. The two tubes in steam generator A with small volume defects at the top of the tubesheet or just above were present during the October 1979 inspection and have not shown an increase in defect size. Steam generator B had no tubes with defects of this kind.

The random ECT inspections of tubes on the cold leg side confirm that tubesheet crevice corrosion is confined to the hot leg side of each steam generator.

As was the case during the previous inspection in March 1980, the latest ECT results continue to show a marked reduction in the number of tubes with indicated tubesheet crevice defects relative to the August and October 1979 inspections during which approximately 230 tubesheet crevice indications were identified. In addition, ten (10) of the 50 tubes in both steam generators identified to contain tubesheet crevice indications during the latest inspection have been shown to have been present since at least October 1979 based upon a re-examination of the eddy current tapes from the previous inspections. Similarly, 20 of the 41 tubes identified in March 1980 to contain tubesheet crevice indications were also shown to have been present during the October 1979 inspection. The latest inspection findings continue to suggest that some of the remedial actions taken by the licensee following the October 1979 inspection, particularly the lower temperature operation, may be succeeding in retarding the rate of tubesheet crevice corrosion. In this regard, it should be noted that the deep crevice indications first identified during this inspection, but which were apparently present during the October 1979 inspection, have essentially remained stable since that time without developing into leaks.

Analysis of the six (6) tube specimens removed from the Unit 1 steam generator during the October 1979 and March 1980 outages has demonstrated that the presence of integranular attack within the tubesheet crevice cannot be reliably detected with single or multifrequency ECT until cracks are developed along the grain boundaries. Partially through wall cracks of significant size are generally detectable with ECT, even in the tubesheet region. However, very small volume defects, which in turn result in very small ECT signal-to-noise ratios in the tubesheet region, may be easily overlooked by the data evaluators. As noted earlier, several of the eddy current indications observed in the current inspection and the March 1980 inspection were apparently present since October 1970, but were not identified at that time. We believe the licensee's inability to identify the source of the leaking tube in steam generator B to be a further example of the difficulties in discriminating very small volume defects in the tubesheet region. However, we believe tubes with small volume defects (small signal to noise ratio) can generally maintain their integrity during the full range of normal operating and accident conditions.

The safety significance of intergranular attack and stress corrosion cracking within the tubesheet crevices was evaluated in our November 30, 1979 SER. Based upon our review of the latest inspection results, the November 30, 1979 evaluation remains valid and is incorporated into this SER by reference.

Conclusions

We conclude that the Point Beach Unit 1 steam generator may operate under the conditions of the November 30, 1979 Order and the January 3, 1980 Order without impairment to the health and safety of the public for the following reasons:

1. The 100% inspection and hydrostatic tests have identified all tubes with significant defects to ensure an adequate margin of safety for the proposed period of operation.

2. The operating conditions (i.e., reduced pressure and temperature) during the past 150 EFPD has been successful in retarding the rate of tube degradation.

3. The cumulative number of tubes plugged (12.2%) is well below the 18% assumed for the LOCA-ECCS analysis.

Dated: August 8, 1980

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

February 13, 1981

ATTACHMENT 7

Docket Nos. 50-266
and 50-301

Mr. Sol Burstein
Executive Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Burstein:

This refers to your letters of December 18 and December 23, 1980 and to discussions held with members of your staff on December 24, 1980 regarding the recent Point Beach Unit 1 steam generator tube inspection. As discussed with you on December 24, we completed our review of your inspection results on that date and found them acceptable for continued operation.

Also, based on recent inspection results and operating experience as it stands today, we agree that both your inspection schedule and plans for the next inspection are reasonable subject to the following conditions:

A. That you conduct a 2000 PSI primary-to-secondary and an 800 PSI secondary-to-primary hydrostatic test as has been done in the past.
B. That you include support plate flow slots in your next inspection as discussed with you on December 24th.

With respect to your operating conditions identified in your December 18 letter, item 4 (page 3) is not correct. Our Confirmatory Order for Modification of license dated November 30, 1979, portions of which are still in effect, requires, among other things, that you will not resume operation until we determine in writing that eddy current examinations are acceptable in the event of:

A. primary to secondary leakage in any steam generator in excess of 500 gpd; or
B. any two identified leaking tubes in any 20 calendar day period.

Please refer to paragraphs 4, 5 and 6 of this Order for the specific requirements.

In response to your comments regarding your sleeving of tubes you are reminded that your Technical Specifications require that any tubes degraded greater than 40% nominal wall thickness are required to be taken out of service by plugging and that to sleeve such tubes, in lieu of plugging would require an amendment to your license.

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We request that you submit the results of your next inspection to us for our review prior to plant start-up as you have done in the past.

A response to this letter is requested within 30 days of receipt.

Sincerely,

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

cc: See next page

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
WISCONSIN ELECTRIC POWER COMPANY	)	Docket Nos. 50-266
	)	50-301
(Point Beach Nuclear Plant,	)	(OL Amendment)
Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Motion For Authorization For Interim Operation of Unit 1 With Steam Generator Tubes Sleeved Rather Than Plugged," accompanying Attachments 1-7, and "Licensee's Proposed Form of Memorandum and Order On Licensee's Motion For Authorization For Interim Operation of Unit 1 With Steam Generator Tubes Sleeved Rather Than Plugged", dated September 28, 1981, were served, by deposit in the U.S. Mail, first class, postage prepaid to all those on the attached Service List, except to those individuals indicated by an asterisk on the Service List, which were hand delivered, on this 28th day of September, 1981.

Delissa A. Ridgway

Dated: September 28, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
WISCONSIN ELECTRIC POWER COMPANY	)	Docket Nos. 50-266
	)	50-301
(Point Beach Nuclear Plant, Units 1 and 2)	)	(OL Amendment)

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