Docket No. 50-266

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

Dear Mr. Burstein:

Enclosed is a signed original Order Modifying Confirmatory Order of November 30, 1979 dated January 3, 1980, issued by the Commission for Point Beach Nuclear Plant Unit No. 1. The Order amends Facility Operating License No. DPR-24 by incorporating and confirming those commitments made by Wisconsin Electric Power Company in its letter of December 31, 1979. The commitment is to operate unit 1 at a reactor coolant system pressure of 2000 psia.

We have concluded that this additional limit is necessary for continued 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, Chief
Operating Reactors Branch No. 1
Division of Operating Reactors

Enclosure:
Order Modifying Confirmatory
Order of November 30, 1979

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January 3, 1980

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Executive Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

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Order Modifying Confirmatory
Order of November 30, 1979

Mr. Sol Burstein
Wisconsin Electric Power Company

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230 S. Dearborn Street
Chicago, Illinois 60604

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.

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.

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.

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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 curent 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 (%)	$\leq$ 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.