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MEMORANDUM FOR: Albert Schwencer, Chief, Operating Reactors Branch #1, Division of Operating Reactors

FROM: Paul S. Check, Chief, Reactor Safety Branch, Division of Operating Reactors

SUBJECT: APPROVAL OF CHANGE TO TECHNICAL SPECIFICATIONS TO OPERATE POINT BEACH UNITS 1 AND 2 AT EITHER 2250 OR 2000 PSIA

On November 2, 1979, Wisconsin Electric Power Company requested a change in Technical Specifications to operate Point Beach Units 1 and 2 at either 2250 or 2000 psia. We have reviewed their proposed changes and find them acceptable. Our evaluation of these proposed changes is enclosed.

PSCheck

Paul S. Check, Chief
Reactor Safety Branch
Division of Operating Reactors

Enclosure:
As stated

cc: D. Eisenhut
G. Lainas
F. Coffman
V. Panciera
S. Weiss
L. Olshan
K. Kniel
R. Lobel
K. Parczewski

Contact: R. Lobel, RS/DOR Rdg.
X27463

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8002140204
XA

A4

GPP

ENCLOSURE

1.0 Introduction

Wisconsin Electric Power Company (the licensee) has requested changes to the Technical Specifications of Pt. Beach Units 1 and 2 to allow operation at either 2000 or 2250 psia (Reference 1). These changes include (1) defining one 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 Pt. 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 is presently requesting 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.

The request to modify 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 analyses are still satisfied.

2.0 Evaluation

2.1 Overtemperature ΔT Trip and Low Pressurizer Pressure Trip

For condition 1 and condition 2 events 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. The licensee is proposing that the low pressurizer pressure trip setpoint be reduced for 2000 psia operation from 1865 psia to 1790 psia to allow more operating margin between the lower proposed operating pressure (2000psia) and the low pressurizer pressure trip. The licensee has proposed two equations for the overtemperature-ΔT trip, one equation to apply to operation at 2000 psia, the other to apply to 2250 psia. The equation for 2250 psia is the equation which is currently in the Technical Specifications and, therefore, requires no further justification at this time. (The licensee states that the lowering of the low pressurizer pressure setpoint will have no effect on the validity of this equation since it was derived with an even lower value of the low pressurizer pressure trip.)

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 higher trip value, the value was kept the same.

TABLE 1

$\frac{\Delta T}{\Delta T_o}$ VALUES FOR OTΔT TRIP

	WCAP 8151	Present Tech Spec	Present Proposal	
T_{AVG} (F)	2000 psi	2250 psi	2000 psi	2250 psi
550	1.47	1.48	1.465	1.48
560	1.33	1.33	1.315	1.33
570	1.16	1.18	1.165	1.18
578	1.029	1.06	1.045	1.06
580	.9978	1.03	1.015	1.03
590	.839	0.88	.87	.88

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.

2.2 Transient and Accident Analyses Affected by Lower Operating Pressure

The licensee has also reviewed the Condition 2 events 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 psia which is above the low pressure trip being proposed for 2000 psia operation. The licensee states that the analysis 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 with the corresponding low pressurizer pressure trip of 1790 psia, only 0.8 full power seconds would result in the case of the worst small break.

The licensee also stated in conversations with the staff that future safety analyses of pt. Beach Units 1 and 2 will be done at both operating pressures of 2000 psia and 2250 psia. We consider this to be important in continuing to justify the operation at both pressures.

The Large Break Loss of Coolant Accident (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 phenomonologically 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 Pt. Beach Units 1 & 2 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 or the low pressurizer pressure reactor trip (it is a function of the high pressurizer pressure reactor trip). It is therefore unaffected by this proposed change in Technical Specifications.

3.0 Conclusions

The licensee has proposed changes to the Technical Specifications to allow operation at both 2000 and 2250 psia.

The staff has reviewed the proposal and the proposed changes to set points and finds these acceptable based on two points. The first is that the licensee, using the Standard Westinghouse reload methods (Reference 6), has verified that Pt. Beach Units 1 and 2 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.

In evaluating this change, as discussed in Section 2.2 of this Safety Evaluation Report, the licensee stated that future safety analyses will be done at both 2000 psia and 2250 psia. We consider this important and will assume that this will be done in addition to the steps spelled out in the standard Westinghouse reload methods (Reference 6).

With the above condition, we find this change to the Technical Specifications to be acceptable.

References

1. Letter from S. Burnstein, 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, dated 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 Burnstein, 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 Wisconsin Michigan Power Company," September 30, 1974, transmitted by letter; Karl R. Goller to Mr. Sol Burnstein, 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. Burnstein, Wisconsin Electric Power Company to H. Denton, USNRC, December 19, 1979.