U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-266/79-01

Docket No. 50-266 License No. DPR-24

Licensee: Wisconsin Electric Power Company
231 West Michigan
Milwaukee, WI 53201

Facility Name: Point Beach Nuclear Plant, Unit 1

Inspection At: Point Beach Site, Two Creeks, WI

Inspection Conducted: January 22-23, 1979

Inspectors: E T Chow
E. T. Chow 2/13/79

E T Chow for
J. E. Kohler 2/14/79

E T Chow for
J. F. Streeter (January 23, 1979)

Approved By: Jerry D. Smith for
J. F. Streeter, Chief 2/14/79
Nuclear Support Section 1

Inspection Summary

Inspection on January 22-23, 1979 (Report No. 50-266/79-01)
Areas Inspected: Routine, announced inspection of control rod worth measurement; core thermal power evaluation; power coefficient; shutdown margin determination; reactivity anomaly; core power distribution limits; safety valve surveillance. The inspection involved 26 inspector-hours onsite by three NRC inspectors.
Results: No items of noncompliance or deviations were identified.

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DETAILS

1. Persons Contacted

*J. Greenwood, Assistant to Manager
*J. J. Zach, Reactor Engineer
*J. Bauer, Technical Assistant
*F. Zeman, Office Supervisor

*Denotes those present at the exit interview.

2. Control Rod Worth Measurement

The inspectors reviewed the measurements performed to determine control rod worth.

The inspectors noted that the worth of Control Bank A was determined from the worths of Control Bank C and Control Bank D. The initial measurement of the worth of Control Bank A by the reactivity computer was incorrect because the boron dilution rate was 97 ppm/hr and appeared to be unconservatively fast. The inspectors further noted that the test procedure did not specify any boron dilution rate as a guideline. In addition the inspectors stated that there was no documentation of the predicted worth of Control Bank A used in the integral worth calculation.

The licensee acknowledged the need for specifying a boron dilution rate to obtain meaningful results and stated that a dilution rate of $\leq$60 ppm/hr would be incorporated in the test procedure. The licensee further stated that the predicted worth of Control Bank A used in the integral worth calculation was not documented in WCAP 9368 (The Nuclear Design and Core Management of the Point Beach Unit 1 Nuclear Reactor Cycle 7) and was provided by Westinghouse based on the specific control rod configuration used in the calculation.

No items of noncompliance or deviations were identified.

3. Core Thermal Power Evaluation

The inspectors reviewed the information performed on January 16, 1979, relating to core thermal power evaluation.

The inspectors noted that the excore detectors were calibrated and the input to the offsite timesharing computer program were taken from the actual plant conditions. The inspectors verified

that the onsite computer program was working properly, and the core thermal power calculated by the onsite computer and the power calculated by the offsite time-sharing program agreed with the acceptance criterion of ±5%.
No items of noncompliance or deviations were identified.

4. Power Coefficient

The inspectors reviewed the measurements performed on November 4 and 5, 1978, for determining power coefficient. The power during the measurements varied from 95% to 85%, and was larger than the 65% specified in the test procedure.

An acceptance criterion of ±10% was established for the difference between the measured power coefficient and the predicted value provided by Westinghouse. The average measured power coefficient was -12.7 pcm/% change of power and the predicted power coefficient was -11.5 pcm/% change of power (WCAP 9368 predicted value at 900 ppm and 90% power at beginning of life unrodded conditions). The licensee's measurements showed that the acceptance criterion was met.

No items of noncompliance or deviations were identified.

5. Shutdown Margin Determination

The inspectors reviewed the information relating to an analytical determination of shutdown margin beginning of life (BOL) and end of life (EOL).

Since the difference between the measured and the predicted worth of the control rod banks satisfied the review and the acceptance criteria established by NRR.[1/2/] The measurements confirmed that adequate reactor shutdown capability existed at BOL and EOL.

No items of noncompliance or deviations were identified.

6. Reactivity Anomaly

The reactivity anomaly was the difference between the measured critical boron concentration and the predicted value at any burnup during the cycle. The acceptance criterion stated that reactivity anomaly should be less than 1000 pcm which was 1% of reactivity.

The inspectors compared the measured boron rundown curve with the Westinghouse predicted curve. The measured boron concentration

1/ A ltr dtd 10/12/78, from A. Schwencer, NRR to S. Burstein, WEPCO.
2/ A ltr dtd 11/30/78, from T. R. Wilson, WEPCO to A. Schwencer, NRR.

appeared to be higher than the predicted value. A maximum boron concentration value of 25 ppm was noted between the two curves. Since the boron worth was approximately 10 pcm/ppm, the acceptance criterion of 1% of reactivity was equivalent to 100 ppm. The inspectors determined that the 25 ppm satisfied the acceptance criterion of 100 ppm.

No items of noncompliance or deviations were identified.

7. Core Power Distribution Limits

The inspectors reviewed the surveillance data taken at hot zero power on October 15, 1978, to determine core power distribution limits. The review indicated that all prerequisites were met, the onsite computer was using input values from the actual plant conditions, all thermal margins satisfied Technical Specifications requirements, and the calculated values by the computer were within the acceptable criteria established by the licensee.

In addition, the inspectors reviewed the results of full core maps taken at full power on October 20, 1978. The review indicated that the distribution of core power was being maintained within Technical Specifications limitations.

No items of noncompliance or deviations were identified.

8. Setpoint Testing of Safety Valves

The inspector reviewed the September 5, 1978, results of procedure PT.R-1, "Main Steam and Pressurizer Safety Valves Setpoint Check," and noted that the as-left setpoints were as specified in FSAR Sections 4 and 10, which are referenced by TS Table 15.4.1-2, Items 11 and 12. From the test results and discussions with licensee personnel, the inspector determined that normal surveillance testing of pressurizer and main steam safety valves is conducted as follows:

Main Steam Safety Valves

All valves (8) are removed each refueling outage from the main steam header, disassembled, and inspected for defective or worn parts. The nozzle and disc seats are then lapped and any defective or worn parts are replaced before the valves are reassembled and leak tested. Testing is then conducted to establish the correct setpoints before the valves are reinstalled on the main steam header.

The licensee's rationale for using this approach is tha (1) historical evidence indicates that this type (Crosby spring-loaded) valve has never experienced a generic setpoint drift problem and even if a drift occurred it would be in the conservative (downward) direction caused by a reduction in spring tension, and (2) the purpose of surveillance testing of these valves is to provide assurance that the valves will lift at the required setpoint during future operations and is not to confirm that setpoints have not drifted during past operations.

The licensee believes that experience indicates that setpoint testing of these valves without first cleaning them would result in foreign material (corrosion products) being trapped in the seats. This foreign material would not allow the valves to seat properly and seat cutting due to leaking steam would result. The licensee outinely goes through the lapping process on each valve to avoid this problem which could result in a forced plant outage due to excessive MSSV leakage.

Pressurizer Safety Valves

These valves (2) are the same type (Crosby spring-loaded) valve as the MSSVs. The licensee usually determines the as-found setpoint condition and does not routinely disassemble the valves. This is because the reactor coolant is a clean system compared to the main steam system and, therefore, foreign material is not expected to be trapped in the valve seats. The valves are only disassembled and corrective maintenance performed without first verifying the as-found setpoint when boron crystal buildup indicates seat leakage and the need for lapping.

The inspector discussed the following items with the licensee:

a. Procedure PT.R-1 does not accurately reflect the sequence and scope of safety valve testing being done in that (1) the pressurizer relief valves are not leak tested unless boron crystal buildup is noted, and (2) the main steam safety valves are not leak tested until after the seats are lapped.

b. Procedure PT.R-1 does not require recording as-found conditions. This appears to be contrary to the licensee's commitment to ANSI 18.7-1976, Section 5.3.10, last sentence.

The licensee stated that he would review PT.R-1 and assure that the procedure reflects actual setpoint testing practice. The licensee and inspector both stated they would review the licensee's commitment

to ANSI 18.7-1976, Section 5.3.10, last sentence, further applies to determine if that document applied to as-found settings of safety valves. Pending completion of these efforts, this matter is an unresolved item (266/77-01-01).

9. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. An unresolved item disclosed during the inspection is discussed in Paragraph 8.

10. Exit Interview

The inspector met with licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on January 23, 1979. The inspectors summarized the purpose, scope, and findings of the inspection.