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

REGION III

Report No. 50-266/79-06; 50-301/79-07

Docket No. 50-266; 50-301 License No. DPR-24; DPR-27

Licensee: Wisconsin Electric Power Company
231 West Michigan
Milwaukee, WI 53203

Facility Name: Point Beach Nuclear Plant - Units 1 and 2

Inspection At: Point Beach Nuclear Plant, Two Creeks, Wisconsin

Inspection Conducted: May 3 and 4, 1979

Inspectors: D. C. Boyd 5-24-79

L. A. Reyes 5/24/79

Approved By: R. L. Spessard, Chief 5/29/79
Reactor Projects Section 1

Inspection Summary

Inspection on May 3 and 4, 1979 (Report No. 50-266/79-06; 50-301/79-07)
Areas Inspected: This special announced inspection was conducted to observe specific plant conditions; to verify licensee actions in response to IE Bulletins No. 79-06, No. 79-06A, and No. 79-06A, Revision 1; to verify the operator awareness regarding the details of the Three Mile Island event, and the operator awareness of any procedural changes resulting from the above identified IE Bulletins. This inspection involved 27 hours of onsite inspection by two NRC inspectors.
Results: No items of noncompliance were identified.

7907240152 Q

DETAILS

1. Persons Contacted

*J. Greenwood, Assistant to Manager
F. Rhodes, Operations Superintendent
*R. Link, Assistant to Operations Superintendent
*Ms. F. Zeman, Office Supervisor
R. Mulheron, Shift Supervisor
R. Nelson, Shift Supervisor
V. Klosterman, Shift Supervisor
*G. Helgeson, Training Supervisor
L. Kamyzek, Shift Supervisor

In addition to the above, the inspectors interviewed a number of operating personnel during this inspection.

*Denotes those present at the management interview.

2. Observation of Specific Plant Conditions

The inspectors reviewed valve/breaker/switch alignment procedures for selected engineered safety features systems against current piping and instrument diagrams (P&ID) and single line diagrams to verify the adequacy of these alignment procedures. Following this review, the inspectors, accompanied by a licensed operator, physically verified the valve/breaker/switch line-up for the following safety related systems: Auxiliary feedwater; safety injection; containment spray; residual heat removal; and emergency electrical diesel generators. Specific attention was given to those valves or breakers which, in accordance with the facility technical specifications or approved procedure, were required to be locked in a specified position.

No items of noncompliance were identified.

The inspectors also reviewed the most current surveillance test data sheets for the above mentioned systems to determine that the test data does verify that the system is operable and is aligned to perform it's intended function.

Drawings, checklists, and procedures reviewed in the above activities are as follows:

Drawings :

a. Safety Injection Flow 110E017 - 1, 2, 3

b. Auxiliary Coolant System (RHR) 110E029, 035

c. Containment Spray 110E017, 035

d. Electrical, Diesel Generator - Bechtel Drawing E-1 Single Line Diagram Station Electrical

e. Auxiliary Feedwater M-217

Procedures:

a. Diesel Generator Tests TS 2 Revision 7

b. Containment Spray Test IT-05, CL-7A

c. Auxiliary Feed Pump Tests IT-08, CL-13-E

d. RHR Pump and Valve Test IT-03, CL 7A

e. Safety Injection Test HI-Head IT-01, Rev 3, CL-7A; LO-Head IT-04, Revision 3

Other Checklists:

a. Service Water PC-10, Part 1, Revision 2

b. SI and RHR Periodic Checks, Part II, PC-9, Revision 1

c. Steam and electric feed pump, Part I, PC-8, IT-08, Revision 2

d. Master valve lineup check list, CL-7A

An inspection was made to determine if the station's Administrative Control Procedures adequately assure proper return to service after maintenance activities or extended outages. In the case of maintenance activities it was found that the two controlling procedures are the routine and special maintenance procedures 5.1.1, 2, 3, 4 and the equipment isolation procedure 4.13. Both of these procedures plus a "standing order" reference the need for testing of redundant safety related equipment, prior to taking equipment out of service, and reference the need for post maintenance testing to verify operability upon return to service. A master system alignment checklist, (CL-7A), for all safety related systems is performed prior to each start up following a cold shutdown status. This checklist is initialed by the operator as each step is reviewed and signed by the shift-supervisor. This copy is returned as the master system status record.

At this station periodic valve line up verifications are also made on safety related systems each 30 days. Also, it was determined that double valve lineup verification is employed during the equipment isolation of safety related systems (PBNP 4.13).

The results of this portion of the inspection are as follows:

The valve/breaker/switch alignment procedures and checklists are adequate to assure that the systems are properly aligned to perform their design function. The physical check out of several safety related systems verified that the systems were properly aligned. NRC review of the periodic surveillance test procedures and data verified that these tests are adequate to assure system operability.

Several minor procedural changes are in progress to clarify portions of the procedures. These will be reviewed in a separate inspection.

In review of the auxiliary feed water supply system, two items of interest were identified. First, there is no direct auxiliary feed water flow indication in the control room. Control room instrumentation does include pump discharge pressure for each pump and the flow control valve position indication (per cent open). Inspector interview with control operators indicate that the operators can determine when flow exists by observing the discharge pressure and the flow control valve position. They can also establish communication with an operator in the auxiliary feed pump area where direct flow indication does exist. The flow indication in this area is used during the periodic testing to pre-set the control valves for a flow rate of 200GPM per steam generator. The operators also indicated that they could feed a dry steam generator by closing down on the flow control valves and observing the auxiliary feed system pressure changes as they slowly opened the control valves. The second area of interest is in regard to the sharing of the motor driven auxiliary feed pumps by both units. The A motor driven auxiliary feed pump starts automatically on a low steam generator signal and supplies the "A" steam generator in both units and the B motor driven auxiliary feed pump similarly would supply the B steam generator in both units, thus, the automatic initiation of the motor driven auxiliary feed pump results in auxiliary feed flow to all steam generators on both units. This area is covered procedurally by instructing the operators to close the auxiliary feed valves on the non-affected unit. Controls for these valves are located in the control room. It should be noted that each unit has its own steam driven auxiliary feed pump, with separate supply lines, which feed only that units steam generators. This system is automatically initiated and delivers a pre-set flow to each steam generator.

No items of noncompliance were identified.

3. Pressurizer Pressure-Level Circuitry

Presently, per agreement with NRR, the pressurizer circuitry is as follows: All three pressurizer pressure trips are active, two of the three pressurizer level bistables are tripped and one is active. This interim configuration will exist until a technical specification change is approved which will permit a logic modification to place the pressurizer pressure trips in a two out of three logic, and will call for all level bistables to be tripped.

No items of noncompliance were identified.

4. Onsite Review of Operator Training

An inspection was conducted to determine the amount of training the plant operators had received regarding the Three Mile Island incident and to determine that the operators understand how to prevent such an event from occurring in this facility.

A review of the training records shows that all operations personnel have either attended the training sessions given by the NRC or are scheduled to do so. A video tape of this training session was made by the licensee and this video tape will be viewed by those who were not present at the original session.

Several minor procedural changes are being considered as a result of Bulletins No. 79-06, 06A, and 06A Revision 1. These changes are currently in a review process by plant and corporate personnel. Specific operator training is planned prior to the implementation of these changes.

The inspectors interviewed at least two licensed operators on each shift (7-3, 3-11, 11-7) to determine the degree of training or instruction they had received regarding the Three Mile Island event. It was obvious to the inspectors, that the plant operators were thoroughly aware of the details of the Three Mile Island event, and were also aware of plant design and procedural differences between the Three Mile Island Plant and their own plant. All operations personnel questioned displayed a full knowledge and understanding of the pressure/temperature saturation curve for the reactor coolant system and it's proper application. The operations personnel also demonstrated a full understanding of the Safety Injection System (SI) and it's proper use. All operations personnnel contacted indicated that there would be no immediate reason or desire to shut off the safety injection pumps since the pump discharge head at their plant is approximately 1600 psig and therefore could not overpressurize a hot primary system. It was determined that station operators are instructed not to intercept or override automatic action of engineered safety featurs (ESF) unless continued operation of ESF will result in unsafe plant conditions. Rather, the operators have been trained to monitor and perform surveillance of these automatic actions. The existing emergency operation procedures which have been in use since plant operation began, specifically address this issue.

The operations personnel contacted were also aware that a safety injection initiation at their plant automatically results in a containment isolation which terminates all liquid and gaseous discharges and does not automatically restart them when either the SI or containment isolation signal are reset.

The plant licensee personnel indicated that the prompt notification (one hour) posed no problem, but that the details of which phone would be designated as a continuous communication channel, and other information relating to the use of that phone were being deliberated at the corporate level. This will be followed up in a later inspection.

No items of noncompliance were identified.

5. Onsite Assessment of Operating Procedures

An inspection was conducted to determine the licensee's status in terms of procedure review and revision, as necessary, to respond to the guidance of Bulletins 79-06, 06A and 06A Revision 1. It was determined that this licensee does not depend on manual initiation or partial initiation of the Safety Injection

System to assist in the control of the pressurizer level during routine operation event induced pressurizer level transients, thus no procedural changes are necessary.

Additional consideration is required, according to the licensee, to better define when the reactor coolant pumps should be left operating, currently their procedures call for stopping these pumps on major primary system loss of coolant accidents or major loss of feedwater accidents. These procedures are consistent with Westinghouse recommendations and the licensee believes they are also consistent with the intent of the IE Bulletins 79-06, 79-06A, 79-06A Revision 1.

The inspectors interviewed nine licensed operators (two or more from each shift on duty) to determine their understanding of the criteria for the operation of reactor coolant pumps. All were knowledgeable in the pressure/temperature saturation curve considerations for general primary system operations and correctly answered questions related to their present operating procedures as stated above. The recommendations of Bulletin 79-06A, item 7c, have been reviewed by a special licensee task force and it has been determined that no major changes to present procedures are required. More discussion in this area is required since the licensee and Westinghouse do not agree with the Bulletin recommendation that at least one reactor coolant pump should remain in operation in every case where a low pressurizer pressure trip occurs. In addition, the licensee feels that under some conditions the continued operation of a reactor coolant pump could result in massive pump seal failure and could create a greater problem than would have resulted from shutting off the pump. This matter is to be resolved between the licensee and NRR.

Other areas discussed during the interview with the operators included their instructions for feeding a dry steam generator. It was determined that a specific procedure for feeding a dry steam generator does not exist, however, feeding a low level steam generator is addressed. All operators were aware of the thermal shock considerations in overfeeding either a low level or dry steam generator and all correctly answered questions in this area. Plant training does address the rate of auxiliary feed flow upon refilling a low level steam generator and directs the operator to throttle the flow as they approach the level of the main feed water sparger. This precaution considers both thermal shock and water hammer considerations.

The licensee pointed out that the size and location of their steam generators is one of the major design differences between the Westinghouse and the Babcock & Wilcox plants such as Three Mile Island. As an example, Westinghouse analysis indicates that it would take in excess of 30 minutes for their steam generator to boil dry following a trip from full power with no feed flow. In contrast, the Three Mile Island design results in a dry steam generator in approximately one minute.

The operators and staff at this plant have reviewed the Three Mile Island event and have specifically noted the major design differences between the Westinghouse design and the Babcock & Wilcox design of the Three Mile Island plant. The major design differences are:

- The size and location of the pressurizer.
- The specific design of the reactor vessel and vessel internals and the relative location of this vessel in relation to the pressurizer and steam generators.
- The size and location of the steam generators.
- Circuitry differences regarding turbine trip/reactor trip and containment isolation upon receipt of a safety injection initiation.

The operating staff at this plant demonstrated full knowledge and understanding of the above differences and also demonstrated full knowledge of the proper operation of their plant to avoid the conditions that developed at the Three Mile Island plant.

The inspector reviewed the licensee's tagging practices on control panels to ascertain the potential for obscuring status indications, such as, valve or switch positions, meters, indicators and alarms. The inspectors determined that this potential does exist, but the licensee has attempted to minimize the potential by rolling up the tags (normal 3" x 5" size) to fit between lights or switches.

No items of noncompliance were identified.

6. Exit Interview

An exit interview was held on May 4, 1979. Those present are identified by asterix in Details, Section 1. Items discussed included: Operator training on the IE Bulletins 79-06, 06A, and 06A Revision 1; Other training regarding the Three Mile

Island event; Operator response to questions on their understanding of how to prevent voids from forming in the primary coolant system; comments on system valve/breaker/switch line-up review; comments on procedure and checklist review; and comments regarding the two items of interest regarding the auxiliary feed water system.