U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-266/92014(DRP);
50-301/92014(DRP)

Docket Nos. 50-266; 50-301

Licenses No. DPR-24; DPR-27

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

Facility Name: Point Beach Units 1 and 2

Inspection At: Two Rivers, Wisconsin

Dates: May 27 through June 14, 1992

Inspectors: K. R. Jury
J. Gadzala

Approved By: I. N. Jackiw, Chief
Reactor Projects Section 3A

Date 6-30-92

Inspection Summary

Inspection from May 27 through June 14, 1992, (Reports No. 50-266/92014(DRP); No. 50-301/92014(DRP)

Areas Inspected: Special safety inspection by resident inspectors of the circumstances surrounding the excessive cooldown rate of Unit 1 on May 27, 1992, during a routine refueling outage.

Results: Two apparent violations of NRC requirements were identified. During performance of steam generator crevice flushing, Unit 1 reactor coolant system was cooled from about 305° F (152° C) to 170° F (77° C) in a one hour period. This exceeded the technical specification limit of 100° F/hr (56° C/hr). In addition, while performing steam generator crevice flushing on Unit 2 during the previous refueling outage, the operating residual heat removal pump was secured coincident with neither reactor coolant pump being in operation. This was contrary to the technical specification requirement that at least one decay heat removal method be in operation during the conditions in existence at the time.

DETAILS

1. Persons Contacted (71707)

*G. J. Maxfield, Plant Manager
*C. C. Reisenbuechler, Manager - Operations & Technical Support
*N. L. Hoefert, Manager - Operations
W. J. Herrman, Manager - Technical Services
J. F. Becka, Manager - Regulatory & Staff Services
*F. A. Flentje, Administrative Specialist

Other company employees were also contacted including members of technical and engineering staffs, and reactor and auxiliary operators.

*Denotes the personnel attending the management exit interview for summation of preliminary findings.

2. Excessive Reactor Coolant System Cooldown Rate (93702)

a. Overview

On May 27, the plant was performing "Steam Generator Crevice Cleaning," Procedure RP-6B, Revision 0, dated April 27, 1992. At 1:05 a.m., the reactor coolant system (RCS) had been heated using the reactor coolant pumps to a temperature of approximately 310° F (154° C). The reactor coolant pumps were then secured and the steam generator was depressurized to initiate boiling for the first cycle of crevice flushing. One train of the residual heat removal (RHR) system remained in operation, as required by technical specifications (TS), to provide a decay heat removal method.

The procedure directed the operators to minimize the RCS cooldown rate by bypassing the RHR heat exchangers using valve FCV-626 with heat exchanger flow control valves HCV-624 and HCV-625 closed. These valves, by design, allowed a flow of several hundred gallons per minute through the heat exchanger when fully closed. The resultant cooling of the RCS by the RHR system, coupled with the heat removed by the steam generator, and the low decay heat rate, caused RCS temperature to drop to 163° F (73° C) by 2:10 a.m. The largest decrease over a one hour period was 14[illegible]° F (78° C). This rate exceeded the maximum cooldown limit of 100° F (56° C) in any one hour as stated in technical specification 15.3.1.B.1 (266/92014-01). This is an apparent violation.

At 2:11 a.m., the first cycle of crevice flushing was completed and a reactor coolant pump was started to reheat the RCS. The excessive cooldown was not discovered until the following shift. The shift supervisor performing the second cycle of crevice flushing noted difficulty maintaining cooldown rates within the required limits. While reviewing the log to determine how the previous shift had managed the first crevice flushing cycle, he discovered that the technical specification limits had been exceeded. Once the excessive cooldown rate was recognized, the procedure was aborted and licensee management was informed.

b. Procedure Review

Procedure RP-6B, "Steam Generator Crevice Cleaning" contained a note that the maximum administrative heatup or cooldown rate was 50° F/hr (28° C/hr). This note appeared at a number of locations in the procedure where steps direct the operator to either heat up or cool down the RCS. However, this note did not appear before the procedure step that directed securing the reactor coolant pumps to start the flushing cycle. Operating data obtained from previous crevice flushing evolutions indicated that a cooldown rate of less than 50° F/hr (28° C/hr), during the steam generator boiling cycle under the conditions specified in procedure RP-6B, could not be achieved without additional operator intervention. A replication of this procedure on the plant simulator, using an assumed RHR heat exchanger flow and core decay heat rate, supported this conclusion.

As directed by the procedure, the operators displayed RHR heat exchanger inlet and outlet temperatures, and both hot and cold leg wide range loop temperatures on the plant computer. However, no guidance was given as to which of these was the most accurate indication. Certain of the wide range loop temperature detectors were situated in sections of the piping outside the direct RHR flow path and with the reactor coolant pumps secured, were essentially in stagnant legs. During this evolution the most representative indicator of core temperature, of those required to be trended by the procedure, was RHR heat exchanger inlet temperature. As noted below, the operator relied primarily upon the loop temperatures to monitor the cooldown. A more accurate cooldown rate could have been manually calculated from the incore thermocouple readings; however, the procedure did not require monitoring this indication, nor did it require the manual calculation to be performed.

c. Operator Actions

Based on interviews with the operators, in addition to the trends specified by the procedure, the 20, 40, and 60-minute moving averages of cooldown rate from the hot and cold leg loop temperatures were displayed on a process computer screen. Also, the 20-minute average (SCOLDR1) and the incore temperature thermocouple average (TCAVG) were selected for display on the control board although neither was required to be monitored according to the procedure.

The operators stated that the 20-minute average was used as the principle indicator for cooldown rate and that a large (greater than 400° F/hr (222° C/hr)) rate was initially observed on this indicator. This indicator displayed the calculated cooldown rate based on an algorithm that selected either the past 20 minutes, 5 minutes, or 5 seconds of temperature data, depending on the instantaneous change in the rate of cooldown. Larger instantaneous rate changes caused the shorter data scan interval to be displayed. An initially large cooldown rate was caused by the temperature drop at the start of the flushing cycle when reactor coolant pumps were secured. Cold water from the RHR heat exchanger outlet flowed into the B loop cold leg and past the temperature detectors feeding the cooldown rate calculator. This cool water created an artificially large cooldown rate indication.

At the time, the operators discussed the reasons for the large indicated cooldown rate and since it was expected, the matter was not pursued further. As the cooldown rate stabilized about halfway through the one-hour cycle, the progressively cooler RHR water entering the RCS cold leg mixed with the warmer water drawn from the stagnant leg. This apparently created an artificial lessening in the indicated rate of cooldown. Process computer data indicated that the temperature drop sensed by the incore thermocouples was larger than the temperature drop input to the cooldown rate calculator from cold leg temperature element TE-451C.

The operator was confronted with varied cooldown rate indications and did not adequately resolve the differences in, or accuracy of, their readings. According to the operator, attention was paid almost exclusively to the 20-minute average cooldown rate instead of monitoring the RHR heat exchanger inlet temperature during this phase. By not verifying the displayed cooldown rate against other available indicators, the operator was apparently mislead by the indicated cooldown rate of about 85° F/hr (47° C/hr) near the end of the cooldown while actual cooldown over an hour was 141° F (78° C).

During the initial portion of this procedure, there was a discussion between the control operator and the duty shift supervisor about obtaining a waiver for the 50° F/hr (28° C/hr) administrative heatup rate limit. Because of miscommunication, the operator thought that the waiver had been obtained and assumed that it applied to both heatup and cooldown rates. As a result, he only considered the technical specification limit of 100° F/hr (56° C/hr), which effectively negated the procedure notes limiting heatup and cooldown to 50° F/hr (28° C/hr).

d. Safety Significance

An analysis was performed by the reactor vessel's vendor to evaluate the effects of the cooldown transient. This analysis was performed in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Appendix E and included a safety factor of 1.4 on pressure and thermal stresses and 1.0 on residual stresses. The results, which were reviewed by the NRC (NRR), showed that the lowest ratio of the crack initiation toughness to the total stress intensity factor was 1.18. Since the ratio was greater than 1.00, the licensee concluded that the structural integrity of the vessel was assured and that acceptable margins of safety would be maintained during subsequent operations.

e. <u>Previous Crevice Flushing</u>

This was the second time in recent years that the plant had performed this evolution under these conditions. Before this, recent crevice flushes were performed at reduced temperatures with a vacuum applied to the steam generators.

The previous crevice flush on Unit 1 was performed during the spring 1991 outage at RCS temperatures below 200° F (93° C), with a vacuum applied to the steam generators to induce boiling. During that evolution, a violation of technical specifications was cited for exceeding 200° F (93° C) without containment integrity being established. Two temperature alarms on the plant process computer, warning of the temperature limit approach, were not properly acknowledged. In Wisconsin Electric's response to the notice of violation, a number of corrective actions were specified Among these were a requirement for the duty shift supervisor to determine if control room personnel staffing is adequate during the conduct of the crevice flushing procedure; and the installation of a plant modification to allow specific process computer alarms, based on the evolution, to be selected for control board annunciation.

A step to require the duty shift supervisor to determine that control room staffing is adequate to support performance of the crevice flush procedure was incorporated into the vacuum mode version of the procedure (RP-6A). However, the high temperature flushing procedure (RP-6B), did not contain this requirement. During an interview, the shift supervisor stated that he considered staffing levels and felt they were adequate. While performing the first crevice flushing cycle, the control operator was simultaneously involved in back leakage testing of a charging system check valve. This unrelated evolution required the operator's attention in performing control board manipulations of the chemical and volume control system. Consequently, his attention was not fully available for monitoring performance of the crevice flushing evolution.

The modification to allow specific alarms to be selected for control board annunciation was scheduled for completion by the end of 1992 and had not yet been installed. Although the process computer has the capability to provide alarms of cooldown rate parameters at user-selected setpoints, use of this capability was not specified by the procedure nor utilized by the operator.

f. Conclusions

A principle contributor to the cause of this event was the failure of the operator to adequately monitor and understand the rate of cooldown occurring during the steam generator crevice flushing cycle. Several methods were available to operators to monitor RCS temperatures and cooldown rates. Among these were plant process computer system cooldown rate values, RCS hot and cold leg temperatures, and RHR heat exchanger inlet temperature. Despite this, operators were not aware that the 100° F/hr (56° C/hr) technical specification limit had been exceeded. Several other factors also contributed to this event. The procedure did not reflect the corrective action from a previous event which would have required the duty shift supervisor to determine that control room staffing was adequate for the conduct of this evolution; the duty shift supervisor did not augment shift operator staffing to accommodate the crevice flushing operation. Excessive cooling was not anticipated during the boiling cycle when reactor coolant pumps were secured and consequently, inadequate measures were specified in the procedure to minimize the resultant cooldown rate. As a result, the operator became the final defense against the technical specification limit being exceeded. Additionally, several of the temperature instruments required to be trended according to the procedure were located outside the direct coolant flow path when reactor coolant pumps were secured, therefore, their indications were not representative of actual reactor vessel conditions.

3. 1991 Unit 2 Crevice Flushing (92701)

During the autumn 1991 outage on Unit 2, high temperature crevice flushing was performed using procedure WMTP 11.19, "Steam Generator Crevice Cleaning", Revision 7, dated October 22, 1991. This procedure was similar to the one used during the current Unit 1 crevice flushing evolution and was used as a basis for writing the current crevice flushing procedure. Instead of the step in the current procedure which directs minimizing RCS cooldown rate by shutting the flow control valves, the former procedure simply stated to adjust RHR cooling to try to maintain RHR outlet temperature between 290° F (143° C) and 300° F (149° C). Operators on different shifts used three different methods to achieve this: component cooling water flow to the RHR heat exchanger was throttled to reduce cooling; the manual RHR heat exchanger isolation valves were shut or throttled; or the operating RHR pump was secured.

Securing the operating RHR pump was contrary to technical specification 15.3.1.A.3 which stated in part that when the RCS was between 140° F (60° C) and 350° F (177° C) at least one decay heat removal method, consisting of either an RHR loop or a reactor coolant pump, shall be in operation except when required to be secured for testing (301/92014-02). This is an apparent violation.

The operators that performed the crevice flushing procedure during the autumn 1991 Unit 2 outage apparently interpreted this statement to allow them to secure RHR during conduct of this procedure. For the most recent crevice flushing evolution, however, the procedure writer interpreted this specification to require not only the RHR pump running, but also flow through the heat exchanger. The onsite safety review staff did not address interpretation of this specification as it applied to the crevice flushing procedure.

4. <u>Exit Interview (71707)</u>

A verbal summary of preliminary findings was provided to the Wisconsin Electric representatives denoted in Section 1 on June 15, at the conclusion of the inspection. No written inspection material was provided to company personnel during the inspection.

The likely informational content of the inspection report with regard to documents or processes reviewed during the inspection was also discussed. Wisconsin Electric management did not identify any documents or processes as proprietary.