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Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

August 27, 1979

Mr. James G. Keppler, Director
Office of Inspection and Enforcement,
Region III
U. S. NUCLEAR REGULATORY COMMISSION
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

DOCKET NOS. 50-266 AND 50-301
RESPONSE TO IE BULLETIN NO. 79-06C
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Attached are responses to IE Bulletin No. 79-06C, "Nuclear Incident at Three Mile Island - Supplement". The attached information responds, item by item, to the five short-term and one long-term action areas listed in the bulletin.

As noted in the response to Item 4 in the attachment, the review of emergency procedures currently being conducted by the Westinghouse Owner's Group is expected to be completed by mid-October 1979. Following the receipt of the procedural guidelines resulting from that review, we will prepare a summary of the review results and draft modifications to the Point Beach emergency procedures which are indicated by the review results. This summary of the review results along with our intended revisions to the Point Beach procedures will be completed by December 31, 1979.

The schedule for implementing the revisions to the procedures and completing operator training will depend upon the extent of involvement desired by your staff regarding these procedural revisions. If your review of our modified procedures is required, we will not implement the revised procedures or revise operator training programs responding to the modified procedures until your review is complete.

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We believe that this schedule provides for the proper application of administrative procedures and established safety reviews to ensure the continued safe operation of the Point Beach Nuclear Plant.

Very truly yours,

C. W. Fay, Director
Nuclear Power Department

Attachments

Copy to: Director
Office of Inspection and Enforcement

Director
Office of Nuclear Reactor Regulation

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ATTACHMENT

RESPONSE TO IE BULLETIN 79-06C

Short-Term Actions

1. In the interim, until the design change required by the long-term action of this Bulletin has been incorporated, institute the following actions at your facilities:

   A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

RESPONSE:

A special order was written and implemented which directed the operator upon reactor trip and initiation of high pressure safety injection caused by low reactor coolant pressure to immediately trip both reactor coolant pumps in the affected unit.

   B. Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurrence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

RESPONSE:

We disagree in principle with the need for two licensed operators in the control room at all times for a single unit or three licensed operators in a dual unit control room to deal with the possible added requirement of tripping the reactor coolant pumps following safety injection caused by low reactor coolant system pressure. Our existing procedures in effect prior to the Three Mile Island accident required the tripping of the reactor coolant pumps under similar conditions and our special order merely reiterated a previously required action. Our present administrative practice is to provide a third man in the control room with supervisory responsibility but his presence was not mandatory. Only because it was expedient to do so, the special order made the presence of the third operator in the control room a mandatory requirement instead of an administrative practice. This action is intended only as an interim measure until this issue is resolved.

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2. Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200 degrees F is identified.

RESPONSE:

A series of Loss of Coolant Accident (LOCA) analyses for a range of break sizes and a range of time lapses between initiation of break and pump trip applicable to the 2, 3 and 4 loop plants has been performed by the Westinghouse Owner's Group. A report summarizing the results of the analysis of delayed Reactor Coolant Pump trip during small loss of coolant accidents for Westinghouse designed nuclear steam supply systems will be submitted to Mr. D. F. Ross by Mr. Cordell Reed on August 31, 1979. In the report, maximum peak clad temperatures for each break size considered and pump shutoff times have been provided. The report concludes that if the reactor coolant pumps are tripped prior to the reactor coolant system pressure reaching 1250 psia, the resulting peak clad temperatures are less than or equal to those reported in the FFDSAR. In addition, it is shown that there is a finite range of break sizes and RCP trip times, in all cases 10 minutes or later, which will result in peak clad temperatures in excess of 2200°F as calculated with conservative Appendix K models. The operator in any event would have at least 10 minutes to trip the Reactor Coolant Pumps following a small break LOCA, especially in light of the conservatisms in the calculations. This is appropriate for manual rather than automatic action based on the guidelines for termination of Reactor Coolant Pump operation presented in WCAP-9600.

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3. Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

RESPONSE:

The Westinghouse Owner's Group has developed guidelines which were submitted to the NRC in Section 6 and Appendix A of WCAP-9600. The analyses provided as the response to Item 2 are consistent with the guidelines in WCAP-9600. No changes to these guidelines are needed for both LOCA and non-LOCA transients.

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4. Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidelines developed under Item 3 above.

RESPONSE:

The Owner's Group effort to revise emergency procedures covers many issues, including operation of the reactor coolant pumps. The action taken in response to Item 1 is sufficient as an interim measure and no immediate need exists for changing our emergency procedures to include the tripping of the reactor coolant pumps. Wisconsin Electric Power Company is a member of the Westinghouse Owner's Group which expects to complete its review of the LOCA, steamline break and steam generator tube rupture emergency procedures by mid-October, 1979. Following the receipt of the procedural guidelines resulting from that review, we will prepare a summary of the review results and draft modifications to the Point Beach emergency procedures which are indicated by the review results. This summary of the review results along with our intended revisions to the Point Beach procedures will be completed by December 31, 1979. An additional one to two months will be required for implementing the revisions to the procedures and for completion of operator training. If NRC review of our modified procedures is required, we will not implement the revised procedures or revised operator training programs responding to the modified procedures until the NRC review is completed.

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5. Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI 2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

RESPONSE:

Analyses related to inadequate core cooling and the definition of conditions under which a restart of the reactor coolant pumps should be attempted will be performed. Resolution of the requirements for the analyses and an acceptable schedule for providing the analyses and procedure guidelines resulting from the analyses will be arrived at between the Westinghouse Owner's Group and the NRC Staff.

Long-Term Action

1. Proposed and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

RESPONSE:

As discussed in our response to short-term Item 2, we do not believe that automatic tripping of the reactor coolant pumps is a required function based on the analyses that have been performed and the guidelines that have been developed for manual reactor coolant pump tripping. We propose that this item be discussed with the NRC Staff following their review of the Owner's Group submittal.