Wisconsin Electric POWER COMPANY
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October 9, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Denton:

DOCKET NOS. 50-266 AND 50-301
RESPONSE TO IE INFORMATION NOTICE 79-22
ENVIRONMENTAL INTERACTIONS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Your letter dated September 17, 1979, requested all operating light water reactor licensees to address the concerns identified in IE Information Notice 79-22, regarding the potential for certain non-safety grade systems to malfunction due to high energy line breaks and to have an impact on the protective functions performed by safety grade equipment.

This notification was based on a continuing review of the environmental qualifications of equipment by Westinghouse Electric Corporation, the NSSS supplier for the Point Beach Nuclear Plant. The Westinghouse review considered the seven control systems needed to fulfill safety related functional requirements, and seven postulated accidents which encompass all assumed high energy line break environments. Of these forty-nine possible combinations of control system and accident environments, Westinghouse identified fifteen potential interactions. These fifteen interactions are bounded by four specific interaction scenarios involving the non-safety grade systems identified in the Information Notice. These scenarios are:

1. Interaction of a feedwater line break outside containment on the steam generator power operated relief valve (PORV) control system.

2. Interaction of a small feedline rupture on the main feedwater control system.

3. Interaction of a feedline rupture inside containment on the pressurizer PORV control system.

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4. Interaction of an intermediate steamline rupture inside containment on the automatic rod control system.

We have conducted a preliminary engineering evaluation of these specific non-safety grade system interactions and have identified no potential adverse impact that constitutes a significant safety hazard or previously unreviewed safety question. A short discussion of the applicability of each of these specific scenarios is provided below.

The first scenario postulates a feedline rupture outside containment which occurs between the containment penetration and the feedline check valve. At the Point Beach Nuclear Plant the feedline check valves are located as close as possible to the steam generators and are inside the containment. Therefore, if a feedline break occurs outside containment, the spillage involved would only be the feedwater pump discharge. The steam generator secondary water inventory could not be released through this break. Furthermore, the locations in the facade of the feedwater lines at elevations 61' and 48' and the steam generator PORV controls at elevation 88' are such that the possibility of an adverse environment at the PORV controls due to a feedline rupture is extremely remote.

The second scenario postulates a small break in the feedwater line between the steam generator nozzle and the feedline check valve. The length of pipe involved in this scenario at Point Beach is approximately six feet and is located immediately adjacent to the steam generator. This pipe is 16-inch, Schedule 100, Type A106 carbon steel and was subjected to extensive examination during the recent investigation required by IE Bulletin 79-13. Again, a mechanism for the environment produced by a small break (less than 0.2 square feet assumed by Westinghouse for this scenario) impacting the feedwater control circuits of the intact steam generator is not probable. The only elements of the feedwater control system inside containment are the steam flow detectors and the steam generator level detectors. These detectors for the intact steam generator are located on the opposite side of the containment from this postulated small leak and would not be affected. In addition, even if a main feedwater control valve were assumed to close in a steam generator with a decreasing water level due to an unidentified control system interaction, this abnormal operating characteristic would be apparent to the operator. The operator would immediately initiate corrective action to restore main or auxiliary feedwater flow and, if not successful, manually trip the reactor. Further, of course, low low level in either steam generator would automatically initiate electric driven auxiliary feed supply and reactor shutdown.

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The third postulated control system interaction concerns the effects of a feedline rupture inside containment and its resultant environmental impact on the control system for the pressurizer PORVs. In order for this scenario to occur at the Point Beach Nuclear Plant, multiple failures in the PORV control system would be necessary. Recent plant modifications discussed in our letter dated April 27, 1979, require that two out of two separate high pressurizer pressure channels are necessary to open a pressurizer PORV. The signals to open the PORV come from pressurizer pressure transmitters which have been qualified to operate properly for 30 minutes in an accident environment. Even if PORV opening were to occur, the positions of the PORVs are indicated in the control room along with an RTD actuated alarm which indicates when either PORV has lifted. These alarms and indications would alert the operator to this extremely unlikely possibility. The operator would then shut the appropriate blocking valve and terminate the PORV discharge.

The fourth situation specifically referenced in IE Information Notice 79-22 involves an intermediate steam line break inside containment and its interaction with the automatic rod control system. The scenario assumes the adverse environment of this break causes the nuclear instrumentation detectors to initiate a signal which causes the rod control system to withdraw the rods prior to reactor shutdown on overpower ΔT. Again, in the short time available before automatic reactor trip, it is not probable that a steam line break, located between elevations 102' and 84' in the containment, could significantly impact the environment of the nuclear instrumentation channels which are located at approximately elevation 25'. This is obviously an incredible, inconsistent, non-mechanistic assumption. Nevertheless, Westinghouse has performed a bounding analysis of the intermediate steam line rupture with control rod withdrawal on a generic basis which indicates that no fuel damage occurs even given this remote and improbable interaction.

It should be emphasized that the probability of these postulated interactions is exceedingly remote. Implicit in the four bounding scenarios are worst case assumptions concerning the break size and location and the type and extent of consequential failure in the control systems potentially induced by the resultant adverse environment. These worst case assumptions are in addition to the already conservative set of assumptions discussed in the analyses of design basis events as reported in the Final Facility Description and Safety Analysis Report (FFDSAR). These scenarios represent a significantly less probable subset of the design basis events which are dependent on the occurrence of additional worst case events, each having an independent uncertainty of occurring. While no quantitative analysis has been conducted concerning the improbability of these postulated interaction scenarios, the extremely unlikelihood of these events

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provides sufficient basis alone to reaffirm continued safe operation of the Point Beach Nuclear Plant.

In summary, we have considered those interactions identified in your IE Information Notice No. 79-22 and have determined, based on our investigations to date, that there is no significant safety hazard or unreviewed safety question involved. We would, thus, conclude that no licensing action is required and believe that the information provided herein is sufficient for the NRC Staff to reach a similar decision. We shall, of course, continue to investigate these and similar interactions as they come to our attention and will notify the NRC, as necessary, of any safety concerns in accordance with our Technical Specifications and Commission Regulations.

Very truly yours,

Sol Burstein

Executive Vice President

Subscribed and sworn to before me
this 9th day of October, 1979.

Dorothy J. Kirschmann
Notary Public, State of Wisconsin

My Commission expires July 6, 1980.

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