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Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555

Attention: Mr. David H. Wagner, Project Manager
Project Directorate III-3

Gentlemen:

DOCKET NO. 50-266 and 50-301
SUPPLEMENT TO BULLETIN 80-04
POINT BEACH NUCLEAR PLANT

In letters dated April 25, 1980; April 14, 1982; and May 4, 1982; Wisconsin Electric responded to those concerns identified by the NRC in IE Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition." Our conclusions from the evaluation of this bulletin stated that there would be no containment over-pressurization or more severe consequences of a return-to-power transient following a hypothetical main steam line break (MSLB) with continued addition of auxiliary feedwater, if it were to occur. Mr. Robert A. Clark's letter dated October 8, 1982 forwarded the NRC Safety Evaluation of our analysis and concluded that the analysis was acceptable.

On February 19, 1988, while conducting an evaluation of the safety related scope of plant valves, we discovered a postulated single failure scenario involving feedwater addition during a MSLB which brings into question the conclusions of our IE Bulletin 80-04 response. As discussed in FSAR Section 14.2.5, feedwater isolation during a postulated MSLB accident is accomplished by a Safety Injection signal which rapidly closes the main feedwater regulating valves, trips power to the main feed pumps and closes the feedwater pump discharge valves. The latter valves are 16 inch motor operated gate valves which require approximately two minutes to cycle shut. The isolation of feedwater is necessary to limit the positive reactivity additions to the reactor core resulting from the rapid cooldown and to limit the mass and energy release into the containment.

During our review of the QA scope of the main feedwater regulating valves, it was recognized that we would rely upon the main feedwater pump discharge valve to isolate the feedwater flow if one

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assumes no loss of off-site AC power during a MSLB accident and a single failure of a main feedwater regulating valve to close. As the steam generator pressure decreases during the transient, a pressure would be reached at which the condensate and/or heater drain tank pumps could begin to inject feedwater into the faulted steam generator. This water injection would continue until the main feedwater pump discharge valve was fully closed (approximately two minutes). Depending upon factors such as the time at which the back pressure in the faulted steam generator equaled the shut off head of the condensate and heater drain tank pumps, the number of pumps running, head loss in the feedwater lines, etc., the amount of water injected into the faulted steam generator may exceed the amount of feedflow assumed in the plant safety analysis.

As you are aware, the two main concerns in a MSLB accident are the core response, which includes a possible return-to-power situation due to excessive cooldown of the reactor coolant system, and the containment pressure response for the postulated MSLB inside containment. The latter response is dependent on the mass and enthalpy of the steam released to the containment. We have conducted a preliminary evaluation of this postulated single failure scenario as discussed below.

Core Response

The core responses for the present cycles of operation (U1C15 and U2C14) were estimated from data in Nuclear Design Reports provided to Wisconsin Electric by Westinghouse. This evaluation concluded that the reactor cores would remain subcritical at average reactor coolant system (RCS) temperatures greater than 250°F. This evaluation was based on the actual end of life (EOL) shutdown margin by all-rods-in less the most reactive rod, which is assumed to be stuck in the fully withdrawn position. These EOL shutdown margins were 3.87% and 3.96% for Units 1 and 2, respectively. The EOL case is the most severe for MSLB because the moderator temperature coefficient is the most negative at that time in the cycle. The FSAR accident analysis assumes that the EOL shutdown margin would be the Technical Specification limit of 2.77%. In that analysis the reactor can become critical at approximately 365°F average RCS temperature. Therefore, the return-to-power situation would not occur during a MSLB in the present cycles of operation because concentrated boric acid, injected by the SI system, would provide adequate shutdown margin before 250°F could be reached.

Thus, DNB and subsequent core damage would not occur in excess of the amount previously analyzed in the FSAR Chapter 14 Rupture of a Steam Pipe analysis.

Containment Response

As mentioned previously, the pressurization of containment during a postulated MSLB inside containment occurs due to the mass and energy release into the containment. The time response of containment pressure depends upon the rate of mass and energy addition and the rate of mass and energy removal from the containment atmosphere. The mass and energy release from the faulted steam generator depends upon the initial steam generator inventory and assumptions for feedwater addition rate and enthalpy during the transient. The initial rate of mass release from the steam generator is governed by steam generator pressure and break size. The FSAR Chapter 14 analysis initializes the MSLB analysis at Hot Shutdown (HSD). The maximum initial steam generator mass and energy exist for the HSD case, because the mass inventory is highest at HSD. Thus, the initial mass release rate should be maximum for the HSD case.

For a MSLB inside containment, with the reactor operating at power, the accident analysis contained in the FSAR Section 14.2.5 states that after the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analyses which assumed a zero power load condition at time zero. In reality, when the RCS average temperature reaches the no-load value of 547°F, the mass release rate will probably be lower than the HSD case due to lower steam generator pressure at this time in the transient. Therefore, the rate of mass release early in the transient is lower for the at-power case than for the HSD case analyzed in the FSAR.

The effect of continued feedwater addition during the initial phase of the transient is more difficult to determine. The blowdown rates early in the accident should be insensitive to feedwater addition rate. An increase in water to the steam generator will increase the total blowdown and provide more cooling to the primary side. Later in the accident, after approximately one minute of elapsed time, the actuation of full containment safeguards and heat removed by containment structures will tend to reduce containment pressure.

Containment pressure depends on the mass and energy addition rate to and removal rate from the containment atmosphere. A detailed reanalysis of this postulated single-failure scenario is necessary to determine the effect of continued feedwater addition on the containment pressure response. In the unlikely event that the blowdown rate could exceed the containment heat removal capability, the containment design pressure could be exceeded. However, it is unlikely the containment integrity would be challenged due to conservative design margins. Even if the containment developed leaks, the consequences of this accident would not be more severe than the MSLB outside containment, as analyzed in the FSAR.

Conclusion

In order to more accurately assess the potential consequence of this postulated scenario, we are contracting with the NSSS vendor, Westinghouse Electric Corporation, Inc., to perform a detailed reanalysis of this MSLB scenario. This analysis will be based on conservative assumptions for the single-failure identified above. The results will be used to evaluate both the core and containment response and to determine the sensitivity of the accident to continued feedwater addition. The results and conclusions of this reanalysis will be provided to you either in an additional supplement to this bulletin or, if the results of the reanalysis show an accident response which exceeds the criteria in the FSAR for MSLB, in a Licensee Event Report submitted pursuant to 10 CFR 50.73 (a)(2)(v) and (vi).

We have discussed aspects of this information with both the NRC Resident Inspector and with Mr. David Wagner of your staff. We will be happy to answer any questions you may have regarding our plans for evaluating this situation. It is anticipated that the vendor's initial analysis will be completed by June 30, 1988, and we can provide a status report within 30 days of receiving those results.

Very truly yours,

CW Fay

C. W. Fay
Vice President
Nuclear Power

Copies to NRC Resident Inspector
NRC Regional Administrator, Region III