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April 25, 1980

Mr. James G. Keppler, Regional 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
IE BULLETIN NO. 80-04
ANALYSIS OF PWR MAIN STEAM LINE BREAK
WITH CONTINUED FEEDWATER ADDITION
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

IE Bulletin No. 80-04 expressed concern with the effects of additional feedwater flow from the auxiliary feed pumps following a steam line break accident. The concerns were related to the possibility of containment overpressurization and whether the additional feedwater flow would adversely affect the return-to-power transient characteristics. IE Bulletin No. 80-04 was initiated as a result of experiences at three other PWR NSSS plants, not of Westinghouse design. Thus, the conditions which initiated the Bulletin are not strictly representative of those which might occur at Point Beach following a main steam line break. The following is provided in response to the subject Bulletin as it applies to Point Beach, Units 1 and 2.

CONTAINMENT PRESSURE BUILDUP

We have reviewed the Point Beach containment capability in respect to the steam line break analyses as presented in Section 14.2.5 of the Point Beach Final Facility Description and Safety Analysis Report (FFDSAR). This section of the FFDSAR discusses the analyses performed and the safety systems which provide protection against various steam line break accidents.

The Point Beach analyses for the steam line break did not include a detailed blowdown and containment response analysis as such. However, the containment pressure response as a function of released energy for the Loss-of-Coolant Accident (LOCA) conditions is presented in Section 14.3.4, Containment Integrity Evaluation, in the FFDSAR.

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For the Point Beach steam line break, an instantaneous release of 165,000 lbs. and 140 x 10^6 BTU was conservatively assumed for the respective mass and energy release to the containment. Without taking credit for containment cooling and spray systems, this results in a containment pressure of 52 psig as compared with the containment design pressure of 60 psig.

Section 14.2.5 of the Point Beach FFDSAR lists the systems which provide protection against a steam pipe rupture and describes the redundant valve arrangement which prevents blowdown of more than one steam generator to the containment. Following a main steam line break, the protective systems are designed to initiate safety injection, trip the main feed pumps, isolate the main feed lines and close the main steam isolation valves. Operational procedures call for isolation of auxiliary feed flow to the affected steam generator. In addition, the containment cooling and spray systems are available for removal of energy discharged to the containment when and if containment pressure exceeds a specified value.

Section 14.3.4 of the FFDSAR presents a discussion of the energy releases assumed and related containment pressure responses to LOCA events. The energy released to the containment during the LOCA blowdown transients is about 170 x 10^6 BTU blowdown enthalpy. The resultant peak containment pressure is approximately 52 or 53 psig, about the same as expected for a steam line break where 140 x 10^6 BTU was assumed to be instantaneously released. The LOCA analyses also assumed that one spray pump and two fan coolers would start at about 60 seconds, after the initial pressure peak was reached (except for the 0.5 ft^2 break size).

Since the total energy released and the rate of energy release is less in the steam line break than during the analyzed LOCA accident, it is apparent that the steam line break accident is less severe than the LOCA. The difference in assumed total energy released is approximately 30 x 10^6 BTU, which could accommodate continued auxiliary feedwater flow. The auxiliary feed lines at Point Beach contain flow restrictions which limit auxiliary feed flow to about 400 gpm. Conservatively assuming that the 400 gpm would reach saturation conditions at 60 psig in passing through the affected steam generator, a continued energy release rate of about 3.9 x 10^6 BTU/min. blowdown enthalpy would result. Thus, the margin in energy release of 30 x 10^6 BTU could accommodate more than 7-1/2 minutes of continued auxiliary feed pump operation before the LOCA assumptions were exceeded, without taking credit for containment spray and cooling systems. The continued energy addition at the rate of 3.9 x 10^6 BTU/min. can easily be accommodated by the containment spray and containment cooling systems which would have ample time to begin operation.

Based on the results of the LOCA analyses and the foregoing considerations, it is apparent that there would not be a problem with containment overpressurization at Point Beach as a result of a main steam line pipe break inside containment with continued auxiliary feedwater addition.

CORE REACTIVITY AND THERMAL-HYDRAULIC TRANSIENT

The assumptions for main and auxiliary feedwater flow, as they apply to the Point Beach steam line break transients, were reviewed by Westinghouse Electric Corporation and the Licensee. Several of the relevant assumptions used in the core transient analyses are presented in Section 14.2.5 of the Point Beach FFDSAR and further explained below.

1. The reactor is assumed initially to be at hot shutdown conditions, at the minimum allowable shutdown margin.

2. For the Condition IV breaks, i.e., double-ended rupture of a main steam pipe, full main feedwater is assumed from the beginning of the transient at a very conservative cold temperature.

3. All auxiliary feedwater pumps are initially assumed to be operating, in addition to the main feedwater. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure.

4. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, approximately 10 seconds after the break occurs, while auxiliary feedwater is assumed to continue at its initial flow rate.

5. Main feedwater flow is completely terminated following feedwater isolation.

Based on the manner in which the analysis is performed for Westinghouse plants, the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary-secondary heat transfer, which is the forcing function for both the reactivity and thermal-hydraulic transients in the core. The effect of auxiliary feedwater runout (or failure of runout protection where applicable) is minimal. Greater feedwater flows during the large steam line breaks serve to reduce secondary pressures, accelerating the automatic safeguards action, i.e., steam line isolation, feedwater isolation and safety injection. The assumptions described above are, therefore, appropriate and conservative for the short-term aspect of the steam line break transient.

The auxiliary feedwater flow becomes a dominant factor in determining the duration and magnitude of the steam flow during later stages in the transient. However, the limiting portion of the transient occurs during the first minute, both due to higher steam flows inherently present early in the transient and due to the introduction of boron to the core via the safety injection system.

Westinghouse has evaluated the effect of runout auxiliary feedwater flows in the core transient for steam line break and, based on this evaluation, has determined that the assumptions presently made are appropriate. The concerns outlined in the introduction to IE Bulletin 80-04 relative to, 1) limiting core conditions occurring during portions of the transient where auxiliary feedwater flow is a relevant contributor to plant cooldown; and 2) incomplete isolation of main feedwater flow, are not representative of the Westinghouse NSSS designs and associated balance of plant requirements.

CONCLUSION

Based on the foregoing, there is no containment over-pressurization or worsening of the return-to-power transient following the hypothetical steam line break. As stated above, procedures already in effect call for isolation of and securing all feedwater flow to the affected steam generator. Containment peak pressures resulting from a steam line break are less severe than those for the assumed LOCA and the pressure peak would occur later. The margin which exists plus the containment spray and cooling system energy removal capability can readily accommodate the incremental energy addition resulting from continued auxiliary feedwater flow if it were to occur. The core reactivity transient analyses have taken continued auxiliary feedwater flow into account and it does not affect the core transient during a steam line break.

Please advise us if you have further questions regarding these analyses.

Very truly yours,

C. W. Fay

C. W. Fay, Director
Nuclear Power Department

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