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Consumers Power Company

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May 9, 1980

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DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT - RESPONSE TO IE BULLETIN 80-04 - ANALYSIS OF A PWR MAIN STEAM LINE BREAK WITH CONTINUED FEEDWATER ADDITION

Consumers Power Company's response to IE Bulletin 80-04 (Analysis of a PWR Main Steam Line Break With Continued Feedwater Addition) dated February 8, 1980 for the Palisades Plant is as follows:

Item 1

Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. In your review, consider your ability to detect and isolate the damaged steam generator from these sources and the ability of the pumps to remain operable after extended operation at runout flow.

Response to Item 1

Consumers Power Company has reviewed the containment pressure response analysis to determine if this analysis included the impact of runout flow from the auxiliary feedwater system as well as the impact of other energy sources (eg, continuation of feedwater or condensate flow). As discussed in the bulletin, a design oversight regarding the delivery of condensate flow to a ruptured steam generator was previously identified. This deficiency is presently being addressed as described in Licensee Event Report 79-041.

The current review identified that the initial analysis did not include the impact of runout flow from the auxiliary feedwater system (AFWS). Auxiliary

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feedwater flow was probably not considered because the AFWS was originally intended to be a manually actuated system. This assumed that for the large steam line break, which might be affected adversely by auxiliary feedwater flow, the operator would recognize the intact steam generator and provide feedwater to that steam generator only. Recognition of the intact steam generator would rely on the large indicated pressure differential between the two steam generators.

A plant modification is presently being performed to provide automatic starting of the Palisades AFWS. An analysis has been conducted to evaluate the effect of auxiliary feedwater flow on the containment pressure response in a Main Steam Line Break (MSLB). The assumptions made in this analysis are as follows:

a. Double-ended guillotine rupture of a 36" main steam line at the steam generator nozzle.

b. Initial reactor power of 2650 MWt (licensed power is 2530 MWt). The full power case is more limiting than the zero power case because of the single failure assumption (Assumption C). For the zero power case, the worst single failure is loss of one of the three containment spray (CTS) pumps (loss of off-site power is not considered credible in the zero power case). With two CTS pumps operating, the cooldown of the containment will proceed more rapidly; thus, the full power case bounds the zero power case.

c. Loss of off-site power and failure of a diesel generator to start results in only one containment spray pump and three air coolers being available to cool the containment. The steam generator blowdown analysis assumed the availability of off-site power to run the primary coolant pumps and, thereby, maximize the rate and magnitude of the initial energy release to the containment.

d. Main feedwater flow rampdown from full flow at time of trip to zero flow at 60 seconds post-reactor trip.

e. Full containment spray flow from one pump at 60 seconds after an MSLB, and main steam line isolation 2 seconds after an MSLB. Spray flow time and main steam line isolation valve closure time are important with respect to the magnitude of the initial pressure peak. However, these have little or no effect on the long-term cooldown of the containment.

f. Runout flow from one auxiliary feedwater pump (assumed to be 750 gpm) is initiated at 120 seconds after the MSLB. The automatic AFWS actuation system incorporates a 2-minute timer delay and an automatic flow controller which will be administratively set to less than 250 gpm per steam generator. Failure of the automatic controller to provide maximum flow demand is assumed even though this represents a second single failure in addition to the diesel generator failure. The automatic AFWS logic blocks the start of the steam-driven auxiliary feedwater pump, unless the motor-driven pump does not start and deliver flow. Therefore, only one pump is assumed to be operating. (Ref: Consumers Power Company letter dated 4-23-80.)

For conservatism, the 750 gpm of cold feedwater was assumed to flash to steam with an enthalpy of 1183.1 Btu/lbm (saturated enthalpy at 80 psia). This assumption ignores any physical limitations on the amount of energy available within the primary coolant system to boil the cold feedwater.

The runout flow from one auxiliary feedwater pump will be less than 750 gpm (runout flow has been calculated conservatively to be less than 650 gpm). This calculation was based on a steam generator pressure of 0 psig and with known pump and injection valve characteristics. (Refer to FSAR Figure 9-11 for the auxiliary feedwater pump-head curve.) Acceptable pump performance at flow rates up to 690 gpm and with a net positive suction head (NPSH) of less than 10 feet (available NPSH exceeds 20 feet during actual operation) has been verified by vendor test.

The results of this analysis, in terms of containment pressure vs time, are shown in Figure 1. A single CTS pump (and three air coolers) are found to be more than adequate in removing the energy being deposited in containment as a result of auxiliary feedwater addition. At 30 minutes, the containment pressure is continuing to slowly decrease. It is assumed that by 30 minutes, the operator would recognize (by observing the steam generator pressure difference) that he was feeding the broken steam generator, and terminate feedwater to it.

Item 2

Review your analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position. If your previous analysis did not consider all potential water sources (such as those listed in 1 above) and if the reactivity increase is greater than previous analysis indicated the report of this review should include:

a. The boundary conditions for the analysis, e.g., the end of life shutdown margin, the moderator temperature coefficient, power level and the net effect of the associated steam generator water inventory on the reactor system cooling, etc.,

b. The most restrictive single active failure in the safety injection system and the effect of that failure on delaying the delivery of high concentration boric acid solution to the reactor coolant system,

c. The effect of extended water supply to the affected steam generator on the core criticality and return to power,

d. The hot channel factors corresponding to the most reactive rod in the fully withdrawn position at the end of life, and the Minimum Departure from Nucleate Boiling Ratio (MDNBR) values for the analyzed transient.

Response to Item 2

Consumers Power Company has reviewed the analysis of the reactivity increase which would result from an MSLB inside or outside containment. It has been determined that, although the impact of runout flow from the auxiliary feedwater system was not considered in the analysis, the analysis adequately bounds this case. The applicable MSLB analyses are located in the following references:

Full Power, Off-Site Power, Inside Containment	Ref 1, Section 3.8.1
Full Power, No Off-Site Power, Inside Containment	FSAR, Amend 17, Section 5.0
Full Power, No Off-Site Power, Outside Containment	FSAR, Amend 15, Section 14.3
Zero Power, Off-Site Power, Inside Containment	Ref 1, Section 3.8.2

Reference 1: XN-NF-77-18, "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt."

In each full power case, the main feedwater flow to each steam generator was assumed to be reduced from full flow to 5% of full flow (the mass equivalent of 560 gpm of cold auxiliary feedwater to each steam generator) over the 60 seconds immediately following a reactor trip. This assumption is conservative when considering runout flow from one auxiliary feedwater pump for the following reasons:

a. The plant is presently being modified as a result of a discovered deficiency (see Licensee Event Report 79-041) to close the main feedwater regulating and bypass valves on a low steam generator pressure signal (approximately 500 psia). This will result in a complete termination of main feedwater much sooner than assumed in the analysis. The analysis also assumed more than 48,000 lbm of main feedwater to be delivered to the broken steam generator during the first 60 seconds.

b. In each case, the peak core heat flux and the MDNBR occurs at or before the time of steam generator dryout (where dryout is defined to occur when break flow = assumed feedwater flow), and before the initiation of auxiliary feedwater at 120 seconds. Steam generator dryout causes an abrupt drop in steam flow, core power and an increase in MDNBR. A slightly greater feedwater flow rate and steam flow rate after steam generator dryout (ie, 750 gpm instead of 560 gpm) would have no significant impact on criticality margins or core power levels.

For the zero power case, 415 gpm of cold auxiliary feedwater were assumed to be delivered to each steam generator for the duration of the analyzed transient.

This assumption is bounding even when considering runout flow from one auxiliary feedwater pump for the following reason. The safety analysis shows that the peak core heat flux occurs long before the time of steam generator dryout. A slightly higher feedwater flow rate (750 gpm vs 415 gpm) would not significantly affect the magnitude or the rate of primary system cooldown (or the resulting criticality margin) prior to dryout. The cooldown rate is chiefly governed by the break area, the primary-to-secondary temperature differential, and the primary coolant flow rate. This small amount of additional auxiliary feedwater prior to dryout would have a negligible effect on secondary temperature and an insignificant effect on the core return to power.

Item 3

If the potential for containment overpressure exists or the reactor-return-to-power response worsens, provide a proposed corrective action and a schedule for completion of the corrective action. If the unit is operating, provide a description of any interim action that will be taken until the proposed corrective action is completed.

Response to Item 3

As a result of this review, no additional safety problems associated with the main steam line break accident have been identified; therefore, no further corrective actions are proposed.

Steven R Frost (Signed)

Steven R Frost
Palisades Licensing Engineer

CC Director, Office of Nuclear Reactor Regulation
Director, Office of Inspection and Enforcement
Resident Inspector-Palisades

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FIGURE 1