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Wisconsin Electric POWER COMPANY
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September 17, 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. 79-21
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

Attached are the responses to IE Bulletin No. 79-21, "Temperature Effects on Level Measurements". The attached information responds, item by item, to the four action areas listed in the bulletin.

Occurrence of a high energy line break inside containment, followed immediately by the uniform heatup of the reference leg of any level measuring system to temperatures near the maximum containment temperature for such accidents, is a very remote possibility. Furthermore, we do not believe that the possible effect of containment temperature on indicated water levels inside containment presents a "potential substantial safety hazard". We have, however, performed our reviews and analyses assuming the worst potential temperature effects.

Very truly yours,

C. W. Fay

C. W. Fay, Director
Nuclear Power Department

Attachments

Copy to: Director
Office of Inspection and Enforcement
Division of Reactor Operations Inspection

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ATTACHMENT

RESPONSE TO IE BULLETIN 79-21

Item 1

Review the liquid level measuring systems within containment to determine if the signals are used to initiate safety actions or are used to provide post-accident monitoring information. Provide a description of systems that are so employed; a description of the type of reference leg shall be included, i.e., open column or sealed reference leg.

RESPONSE

After a review of all liquid level measuring systems within containment, the following systems, described below, were determined to initiate safety actions or provide information required for post-accident monitoring:

1. Steam Generator Narrow-Range Level

   Each steam generator (S/G) has three narrow-range level measuring channels which measure downcomer liquid level above the top elevation of the S/G U-tubes. Each system consists of one Foxboro Model No. 613 HM "force-balance" differential-pressure transmitter. The transmitter is connected to the steam generator downcomer on the variable leg side and is connected to an open-column reference leg on the other side. The outside of the reference leg pipe is exposed to containment atmosphere and has a condensing pot on the top which is connected to the steam generator steam space. Signals from these transmitters are used for remote level indication, level control, alarms, and safety trips.

2. Steam Generator Wide-Range Level

   Each steam generator has one wide-range level measuring system. It is identical to the S/G narrow-range system except that the larger

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level span covers downcomer liquid level from just above the S/G tube sheet to above the moisture separators. Signals are used for remote level indication only.

3. Pressurizer Level

The pressurizer has three hot-calibrated and one cold-calibrated liquid level measuring systems. Each system consists of one Foxboro Model No. 613 HM "force-balance" differential-pressure transmitter. The transmitter is connected to the pressurizer near the bottom on the variable-leg side. It is connected to a sealed reference leg on the other side. The outside of the reference leg pipe is exposed to the containment atmosphere and has a bellows-type seal near the top. A water seal is maintained on the top of the bellows by a condensing pot connected to the pressurizer steam space. The signals from the hot-calibrated transmitters are used for remote level indication, level alarms, level control, and safety trips. The cold-calibrated transmitter signals are used for remote level indication only.

4. Containment Sump B Level

Containment Sump B has two magnetically-actuated float switch assemblies which energize lights in the control room at containment sump water levels of 3 and 7 feet, and 5 and 9 feet, respectively. The indication is used to confirm sufficient water inventory in the sump prior to shifting the Emergency Core Cooling Systems to the recirculation mode following safety injection. This type of measuring system has no reference leg.

Item 2

On those systems described in Item 1 above, evaluate the effect of post-accident ambient temperatures on the indicated water level to determine any change in indicated level relative to actual water level. This evaluation must include other sources of error including the effects of varying fluid pressure and flashing of reference leg to steam on the water level measurements. The results of this evaluation should be presented in a tabular form similar to Tables 1 and 2 of Enclosure 1.

RESPONSE

The effect of post-accident ambient temperatures on the indicated water level for the systems described previously is presented in Tables 1 through 4. The containment sump B level system is not affected significantly by ambient temperatures so no table was provided for it.

The possibility of reference leg flashing on indicated pressurizer and steam generator level, during high energy line break accidents, was analyzed. Reference leg flashing on the pressurizer is not significant since a sealed reference leg is used. During a major steam line or feedwater line break accident, reference leg flashing, on the affected steam generator only, may occur late in the blowdown phase. The affected steam generator would boil dry in the normal course of the accident, so level indication late in the transient is not required. Steam generator pressure during a loss of coolant accident is expected to remain high enough to prevent reference leg flashing during that accident. In conclusion, reference leg flashing is not a concern on any high energy line break.

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Table I

Correction Required to Indicated Steam Generator
Narrow-Range Level for Reference Leg Heatup Effects
Due to Post-Accident Containment Temperature

Reference Leg Temperature (°F)	Correction Required to Low-Low Level Trip Setpoint (Before Reactor Trip) (% of span)		Correction Required to Minimum Allowed Post-Accident Indicated Level* (% of span)			Correction Required to Maximum Allowed Post-Accident Indicated Level** (% of span)		
	S/G Pressure (psig) 800	500	800	500	200	800	500	200
68	-1	-2	-1	-1	0	-1	3	8
100	0	-1						
150	2	0						
200	4	3						
250	7	6						
300	10	9	10	10	11	10	14	19
350	14	13						
400	18	17						

<u>Basis</u>: Level Calibration Steam Generator Pressure = 800 psig (saturated water)
Reference Leg Calibration Temperature = 100°F

* 20% of level span

** 70% of level span

Table 2

Correction Required to Indicated Steam Generator Wide-Range Level for Reference Leg Heatup Effects Due to Post-Accident Containment Temperature

Reference Leg Temperature ($^\circ$F)	Correction Required to Minimum Allowed Post-Accident Indicated Level* (% of span)			Correction Required to Maximum Allowed Post-Accident Indicated Level** (% of span)		
S/G Pressure (psig)	800	500	200	800	500	200
68	-18	-15	-10	-21	-17	-12
300	-10	- 7	- 3	-12	- 9	- 4

Basis: Level Calibration Steam Generator Pressure = 0 psig (subcooled water at 68°F)
Reference Leg Calibration Temperature = 68°F

* 80% of level span
** 90% of level span

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Table 3

Correction Required to Indicated Pressurizer Hot-Calibrated
Level for Reference Leg Heatup Due to Post-Accident Containment Temperature

Pressurizer Pressure (psia)	Correction Required to Minimum Allowed Post-Accident Indicated Level* (% of span)			Correction Required to Maximum Allowed Post-Accident Indicated Level** (% of span)	
	Reference Leg Temperature (°F)	68	300	[illegible]	300
2250		-1	15	-1	15
2000		-2	13	2	18
1750		-3	13	6	23
1500		-4	13	10	26
1000		-6	11	17	32(1)
500		-5	12	26	38(1)

Basis: Level Calibration Pressurizer Pressure = 2250 psia (saturated water)
Reference Leg Calibration Temperature = 100°F

* 20% of level span

** 70% of level span

(1) Indicated level cannot exceed 100% of level span

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Table 4

Correction Required to Indicated Pressurizer Cold-Calibrated
Level for Reference Leg Heatup Due to Post-Accident Containment Temperature

Pressurizer Pressure (psia)	Correction Required to Minimum Allowed Post-Accident Indicated Level* (% of span)		Correction Required to Maximum Allowed Post-Accident Indicated Level** (% of span)	
Reference Leg Temperature (°F)	68	300	68	300
2250	0	7	-26	-18
2000	-2	7	-24	-16
1750	-2	6	-22	-15
1500	-3	6	-21	-13
1000	-3	5	-17	- 9
500	-2	6	-13	- 5

Basis: Level Calibration Pressurizer Pressure = 14.7 psia (subcooled water at 68°F)
Reference Leg Calibration Temperature = 68°F

* 20% of level span

** 70% of level span

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Item 3

Review all safety and control setpoints derived from level signals to verify that the setpoints wil' 'nitiate the action required by the plant safety analyses throughout th range of ambient temperatures encountered by the instrumentation, including accident temperatures. Provide a listing of these setpoints.

If the above reviews and evaluations require a revision of setpoints to ensure safe operation, provide a description of the corrective action and the date the action was completed. If any corrective action is temporary, submit a description of the proposed final corrective action and a timetable for implementation.

RESPONSE

A review of all safety and control setpoints derived from the level measurement systems described earlier showed that the only setpoint which could be adjusted for conservatism was the steam generator "lo-lo level reactor trip" and "auxiliary feedwater system startup" setpoint. This setpoint provides primary protection for loss of feedwater accidents (including pipe breaks) and is used as a backup signal for auxiliary feedwater system startup on other high energy line breaks inside containment. No other safety or control setpoints initiate any action required by the safety analyses for high energy line break accidents.

Heatup of the reference leg for steam generator level indication following a high energy pipe break in containment requires some elaboration. If a loss of feedwater due to a small pipe break inside containment were to occur, the plant operators would be alerted by increased containment humidity, increased sump level alarms, and increased containment pressure. The main feedwater pumps would continue supplying water to the steam generators, albeit the indicated level in one steam generator could be slightly in error. A "high feedwater flow" alarm could be annunciated, depending upon the size of the leak. Safety injection and reactor trip is initiated at a containment pressure of 6 psig or less and would terminate power operation if operating personnel had not done so. The saturation temperature for a homogeneous air-steam

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mixture at this pressure is about 170°F, well below the 300°F temperature assumed in our analysis.

The physical configuration of the steam generators at Point Beach Nuclear Plant tends to reduce the impact of a major feedwater line break. The feedwater line penetrates the steam generator above the biological shield, eliminating any "compartmental" or "chimney" effect of hot steam rising up to affect the reference legs. In addition, the level taps are on the opposite side of the steam generator from the feedwater line preventing direct impingement of steam on the reference legs. Therefore, the rapid water level decrease resulting from this accident would trip the reactor on lo-lo level before the steam could significantly bias the indicated level.

The effects of a steam line break upon the reference legs have also been evaluated. The results of an assumed small leak are similar to that of a feedwater line leak. The primary protection for a large steam line break is a high containment pressure signal backed up by a low steam line pressure signal.

The steam generator lo-lo level reactor trip and auxiliary feedwater system startup setpoint may be raised to account for possible reference leg heatup effects following a feedwater line break inside containment. The maximum possible containment bulk temperature which could occur, prior to receiving a safety injection signal from high containment pressure, is 170°F. A temperature of 300°F corresponds to the temperature of superheated steam in containment resulting from the release of steam generator secondary-side saturated steam at the maximum expected enthalpy. This situation is not plausible because the containment and equipment in it serve as a heat sink to reduce the superheat. The new setpoint for lo-lo steam generator level is conservatively calculated as shown below:

BASIS	Narrow-Range Level Span
Safety Limit Used in Accident Analyses	0%
Process Measurement Accuracy	2%
Instrument and Bistable Error	1%
Reference Leg Heatup Maximum Error (for a reference leg temperature of 300°F)	10%
New Setpoint	13%

The new setpoint will increase the plant's vulnerability to inadvertent reactor trips and auxiliary feedwater system startups. Although we believe there is minimal positive benefit and there are negative aspects resulting from the setpoint change, the setpoints will be changed. The setpoint adjustment to a new value greater than or equal to 13% of narrow-range level span will be accomplished on Unit 2 by September 30, 1979, and on Unit 1 prior to startup following the refueling outage scheduled to begin on October 5, 1979.

The safety and control setpoints derived from steam generator narrow-range level and pressurizer level are shown in Tables 5 and 6, respectively. The other level measurement systems described earlier do not have safety and control functions other than post-accident monitoring.

TABLE 5

STEAM GENERATOR NARROW-RANGE LEVEL CONTROL AND SAFETY SETPOINTS

Function	Level Span
Low-Low Level Reactor Trip and Auxiliary Feed System Startup Setpoint	10%
Low Level Reactor Trip Setpoint (in coincidence with feedwater flow/steam flow mismatch)	20%
Level Program (low limit for 0% power)	39%
Level Deviation Alarm Setpoint	±3% from level program
Level Program (high limit for ≥20% power)	52%
High-Level Override for Feedwater Valve Closure, Turbine Trip, and Main Feed Pump Trip	70%

Pressurizer Hot-Calibrated Level
Control and Safety Setpoints

Function	Level Span
Low-Low Level Heater Cutout, Letdown Isolation Setpoint	12%
Level Program (low limit for $T_{avg} \leq 547°F$)	20%
Level Deviation Alarm Setpoint	$\pm$ 5% from level program
Level Controller Gain	1% charging pump speed/1% level deviation from level program
Level Program (high limit for $T_{avg} \geq 570°F$)	45.8%
High-Level Deviation Heaters-On Setpoint	5% above level program
High-Level Alarm Setpoint	70%
High-Level Reactor Trip Setpoint	90%

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Item 4

Review and revise, as necessary, emergency procedures to include specific information obtained from the review and evaluation of Items 1, 2 and 3 to ensure that the operators are instructed on the potential for and magnitude of erroneous level signals. All tables, curves, or correction factors that would be applied to post-accident monitors should be readily available to the operator. If revisions to procedures are required, provide a completion date for the revisions and a completion date for operator training on the revisions.

RESPONSE

The operators at Point Beach Nuclear Plant have been instructed on the general effect of post-accident ambient temperatures on level instrument readings following a high-energy line break inside containment; however, slight modifications to emergency procedures and/or standing orders would provide more specific information to the operator. These revisions will be made on a schedule compatible with the schedule of procedure modifications required in response to IE Bulletin No. 79-06C, "Nuclear Incident at Three Mile Island - Supplement". The revisions will be completed by December 31, 1979. Implementation of the revised procedures and completion of operator training will require an additional one to two months. If NRC review of the procedure modifications is required, the implementation and operator training will not begin until the NRC review is completed.

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