CENTRAL FILES

POWER AUTHORITY OF THE STATE OF NEW YORK

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November 2, 1979
IPN-79-79

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
I. E. Bulletin 79-21

Dear Mr. Grier:

Attachment A to this letter contains the Authority's response to the subject item.

Very truly yours,

Joseph R. Schmieder
Executive Vice President
and Chief Engineer

cc: Office of Inspection and Enforcement
Director of Reactor Operation Inspection
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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ATTACHMENT A

RESPONSE TO IE BULLETIN NO. 79-21

TEMPERATURE EFFECTS ON LEVEL MEASUREMENTS

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
November 2, 1979

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RESPONSE TO IE BULLETIN NO. 79-21

"TEMPERATURE EFFECTS ON LEVEL MEASUREMENTS"

1. NRC Requirement

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

Review of the above requirement indicated the following systems to be applicable:

Systems	Description
Steam Generator Level	Open Column
Pressurizer Level	Sealed Reference Leg.

2. NRC Requirement

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 messurements. Results of this evaluation should be presented in a tabular form.

Response

As requested, an evaluation has been performed and the results are presented in the following four tables which are included in this attachment.

Table No. 1: Correction to indicated steam generator water level of Reference Leg Heatup effects due to post-accident containment temperature for narrow range level transmitters (before reactor trip).

Table No. 2: Corrections to allowable indicated steam generator water level for Reference Leg Heatup and Pressure changes following a high-energy line break for narrow range level transmitters to assure that true level is between the level taps.

Table No. 3: Correction to indicated pressurizer water level for Reference Leg Heatup effects due to post-accident containment temperature (before reactor trip).

Table No. 4: Corrections to allowable indicated pressurizer water level for Reference Leg Heatup and Pressure changes following a high-energy line break, to assure that true level is between the level taps or above the pressurizer heater located below 17% level.

Typically, boiling occurs in the reference leg after rapid depressurization of any steam generator with high containment temperature coincident with density changes in the reference leg following a steam line or feedwater line rupture inside containment. Although a plant-specific analysis has not been performed, recent containment analyses performed using Westinghouse models, indicate that such boiling in the reference leg would not occur.

3. NRC Requirement

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

Response

In Westinghouse's June 29, 1979 letter (NS-TMA-2104) to Mr. Victor Stello, it was stated that in the case of feedline rupture, an adverse environment could be present and could delay or prevent the primary signal arising from steam generator water level (low-low steam generator level). Westinghouse has recommended a change in the allowable water level trip set point to accommodate a potential temperature-induced bias which could result from containment temperatures up to 280°F

(see attached Table 1). The conservative assumption of 280°F would result in a 10% temperature-induced bias. Since initial startup, Indian Point Unit No. 3 has operated with the low-low steam generator level reactor trip set at 15% of span versus the 0% of span assumed in the accident analysis (the technical specification requirement is 5% of the span). The existing 15% margin is more than adequate to accomodate a potential 10% temperature-induced bias and therefore, no steam generator water level trip setpoint change is required. In addition, the reactor would actually be tripped considerably earlier (prior to the ambient containment temperature reaching 280°F) since the high containment pressure trip is set at 3.5 psig which (based on a review of steamline break analysis) corresponds to an expected containment temperature of less than 200°F and, therefore, a potential temperature-induced bias of 4% of span.

4. NRC Requirement

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

Existing plant procedures require that the steam generator levels be maintained between 35% and 60% of span thus assuring that the steam generator tubes are fully covered and the steam generators are not water solid. This required range is more than adequate to compensate for any potential temperature-induced bias (see Table 2). On this basis, we feel that no further instructions or guidance to the operators is necessary.

With regard to the pressurizer, Westinghouse has calculated that a postulated 280°F containment temperature would result in a potential 15% temperature-induced bias due to heatup of the pressurized sealed type reference leg. We are evaluating whether or not any of our post-accident monitoring procedures are impacted by this effect. Any necessary procedural revisions and retraining of operators will be completed by January 1, 1980.

Table No. 1

Correction to indicated steam generator water level for Reference Leg Heatup effects due to post-accident containment temperature for narrow range level transmitters (before reactor trip).

Max. Contain. Temp. reached before Reactor Trip °F.	Correction to S/G Level Per cent of Span
90°	0
200°	4.0
280°	10.0
320°	13.0
400°	20.0

Basis

Level calibration pressure $\leq$ 1000 psia

Reference leg calibration temperature $\geq$ 90°F

Height of reference leg = level span

True level = indicated level - correction

Table No. 2

Corrections to allowable indicated steam generator water level for Reference Leg Heatup and Pressure changes following a high-energy line break for narrow range transmittal level to assure that the true level is between the narrow range tape.

Contain. Temp. F.	Correct. to Min. Allow. Ind. Level Per cent of Span	Correct. to Max. Allow. Ind. Level Per cent of Span
90°	+1	-4
200°	+6	-4
280°	+11	-4
320°	+14	-4
400°	+21	-4

Basis

Level calibration pressure $\leq$ 1000 psia

Reference leg calibration temperature $\geq$ 90°F

Height of reference leg = level span

Pressure $\geq$ 50 psia

Maximum steam generator pressure $\leq$200 psi + calibration pressure $\leq$ 1200 psia

Boiling in the reference leg is not assumed.

Table No. 3

Correction to indicated pressurizer level for Reference Leg Heatup effects due to post-accident containment temperature (before reactor trip)

Max. Contain. temp. reached before reactor trip °F	Correction to Pressur. Level. per cent of Span
90°	0
200°	7
280°	14.5
320°	19.25
400°	29.5

Basis

Level calibration pressure $\leq$ 1000 psia

Reference leg calibration temperature $\geq$ 90°F

Height of reference leg $\leq$ 1.1 x level span

Allowable level = indicated + correction

Table No. 4

Corrections to allowable indicated pressurizer water level for Reference Leg Heatup and Pressure changes following a high-energy line break, to assure that true level is between the level taps or above the pressurizer heater located below 17% level.

Contain. Temp. °F.	Correct. to Min. Allow. Ind. Level Per cent of Span (above level tap)	Correct. to Min. Allow. Ind. Level Per cent of Span (above heater)	Correct. to Max. Allow. Ind. Level Per cent of Span
90°	+6	4	-9
200°	+13	11	-9
280°	+20.5	18.5	-9
320°	+24.25	22.5	-9
400°	+33.5	33.5	-9

Basis

Level calibration pressure $\leq$ 1000 psia

Reference leg calibration temperature $\geq$ 90°F

Height of reference leg $\leq$ 1.1 x level span

Pressure $\geq$ 50 psia

Pressure $\leq$ 200 psi + calibration pressure $\leq$ 1200 psia

Allowable level = indicated + correction

Boiling in the reference leg is not assumed