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September 18, 1979

United States Nuclear Regulatory Commission
Attention: Boyce H. Grier, Director
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334
License No. DPR-66
Response To IE Bulletin 79-21

Gentlemen:

NRC IE Bulletin 79-21, which addresses the effect of increased containment temperature on the reference leg water columns of safety related liquid level measuring systems, has been reviewed.

The information requested by the subject Bulletin is provided as follows:

1. The following liquid level measuring systems inside containment are used to initiate safety actions.

   A. Steam Generator Narrow Range Water Level

   These are all Barton 386 differential pressure transmitters on open reference legs which provide the following actions:

   - Turbine trip and feedwater isolation on high-high steam generator water level

   - Reactor trip on low steam generator water level in coincidence with steam flow - feed flow mismatch

   - Reactor trip on low-low steam generator water level

   - Auxiliary feedwater pump initiation on low-low steam generator water level

   - Post accident monitoring function

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Steam Generator Wide Range Water Level

The wide range instrumentation utilizes the Barton 386 differential pressure transmitter and is similar to the narrow range installation with the exception that it only provides a post accident monitoring function.

B. Pressurizer Water Level

The pressurizer level instrumentation transmitters are Barton Model 313b bellows differential pressure transmitters on sealed reference legs and provide the following functions:

- Reactor trip on high water level
- Post accident monitoring function.

C. The containment sump level instrumentation consists of two Fischer & Porter multilevel float type switches and serve as a post accident monitoring function. These were designed to operate in a post accident environment and are addressed in the BVPS FSAR, Section 7.3.1.3.1.5.

2. Evaluation of the effect of post-accident conditions on indicated water level.

A. Reference Leg Heatup

High energy line breaks inside containment can result in heatup of level measurement reference legs. Increased reference leg water column temperature will result in a decrease of the water column density with a consequent apparent increase in the indicated steam generator water level (i.e., apparent level exceeding actual level).

The following formula can be used to calculate the magnitude of this bias:

$$E = \frac{H_L}{H} \left(\frac{\rho_L,cal - \rho_L}{\rho_f,cal - \rho_g,cal} \right)$$

Where:

E = level error due to reference leg heatup, as a fraction of level span,

H = level span = vertical distance between narrow range taps on steam generator,

H_L = height of reference leg,

= maximum vertical distance from lower tap to water level in condensing pot on upper tap. This must be determined for the limiting instrument connections,

ρ_L,cal = water density at containment temperature and steam generator or pressurizer pressure for which the level indication system was calibrated. If this information is not available, an upper-bound density (lower-bound temperature) must be assumed.

ρ_L = water density in reference leg at the time of interest

(ρ_f,cal - ρ_g,cal) = difference between saturated water density and dry saturated steam density at the steam generator or pressurizer pressure for which the level indication system was calibrated. An upper-bound pressure must be assumed.

This procedure is based on the assumption that the tubing from the upper and lower taps, below the elevation of the lower tap, have the same temperature at all times.

For an example, Figure 1 shows the level bias as a function of reference leg temperature, assuming H_L/H = 1.1 and assuming that the calibration conditions are the following: Containment temperature = 90°F, steam generator pressure = 1000 psia. Figure 2 shows the level bias for a pressurizer with the assumptions noted on the Figure. Please note that Figures 1 and 2 are only examples; however, they approximate the effects that would be experienced at BVPS.

B. Reference Leg Boiling

In addition to the above reference leg density change under subcooled conditions, boiling could conceivably occur in the reference leg following depressurization of any steam generator with high containment temperature. This combination of conditions could only occur following a steamline or feedline rupture inside containment. If such boiling were to occur, it could cause a major bias in the indicated level for a short time period, in the extreme case indicating 100% level when the vessel is actually empty.

For a typical dry containment plant, recent containment analyses performed using Westinghouse models indicate that such boiling would not occur.

C. Coolant Density Changes

A bias in indicated water level may also be introduced by changes in pressurizer or steam generator pressure, due to changes in the density of the saturated water and steam within those vessels. While prediction of the effects of rapid depressurization requires complex calculations for each specific case, the bias which would exist at low power under quiescent conditions can be calculated directly, using the following formula:

$$E = \frac{H_L}{H}\left(\frac{\rho_L,cal - \rho_L - \rho_g,cal + \rho_g}{\rho_f,cal - \rho_g,cal}\right) + \frac{L}{H}\left(\frac{\rho_f - \rho_g}{\rho_f,cal - \rho_g,cal} - 1\right)$$

Where:

E = level error due to density changes in both the vessel and the reference leg, as a fraction of level span,

L = true water level in the vessel, above the lower level tap,

ρ_f = saturated water density at the pressure of interest,

ρ_g = dry saturated steam density at the pressure of interest,

and other symbols have the same meanings as in Section 2A.

For an example, Figure 3 shows the true water level as a function of steam generator pressure and indicated level, assuming the following calibration conditions: containment temperature = 90°F, steam generator pressure = 1000 psia, and the reference leg is at 90°F. Figure 4 is similarly calculated for a pressurizer, with the assumptions noted on the Figure. Please note that Figures 3 and 4 are only examples; however, they approximate the effects that would be experienced at BVPS.

3. Safety Function Setpoints

A. Steam Generator Narrow Range Water Level Trip Setpoints

The only high-energy line rupture within the containment for which the steam generator water level provides the primary trip function is a feedline rupture. To account for reference leg heatup effects, the trip setpoints that are required in the station Technical Specifications have been raised by 10% of level span as a temporary corrective action. A final corrective action has not yet been decided and will be considered during the upcoming refueling outage. Actions under consideration are:

1) Insulation of reference legs

2) Mechanical compensation of sealed reference legs

3) Temperature compensation of transmitter output.

B. Pressurizer Water Level Trip Setpoint

No credit is taken for this reactor trip function following a high energy line rupture inside containment. Thus the trip setpoint need not be revised to include environmental errors.

4. The required accuracy following Condition IV events for pressurizer and steam generator level instrumentation is addressed in the BVPS FSAR, Table 7.5-2, and are as follows:

A. Pressurizer Level - indicate the level is somewhere between 0 and 100% of span.

B. Steam Generator Level - indication need only convey to the operator that water level in the steam generator is somewhere between the narrow range steam generator level taps.

Present plant procedures consistent with the requirements of IE Bulletins 79-06A and 79-06C provide the operators with instructions necessary to meet the requirements of the FSAR even if the upper-bounded errors due to the high energy break are subtracted.

The emergency procedures will also be revised in the future to reflect on the new reference guidelines developed by Westinghouse and the Utility Owner's Group. An implementation schedule is not available at this time but will be developed after NRC approval of the reference guidelines.

If you have any questions regarding this response, please contact my office.

Very truly yours,

C. N. Dunn
Vice President, Operations

Attachments: Figures 1, 2, 3, 4

cc: United States Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

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Figure 1 - Attachment to
Response to IE Bulletin 79-21
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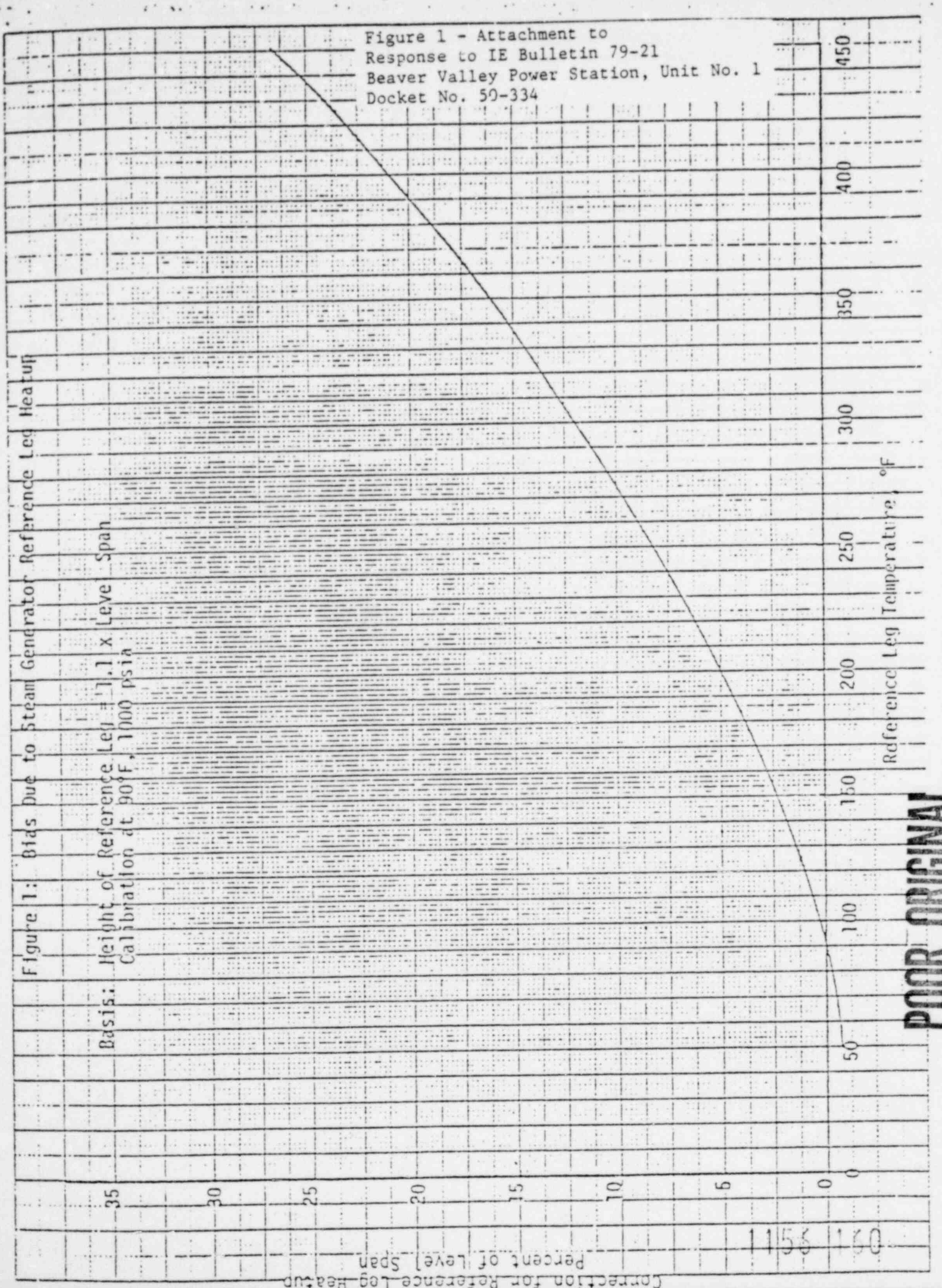

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Figure 2 - Attachment to
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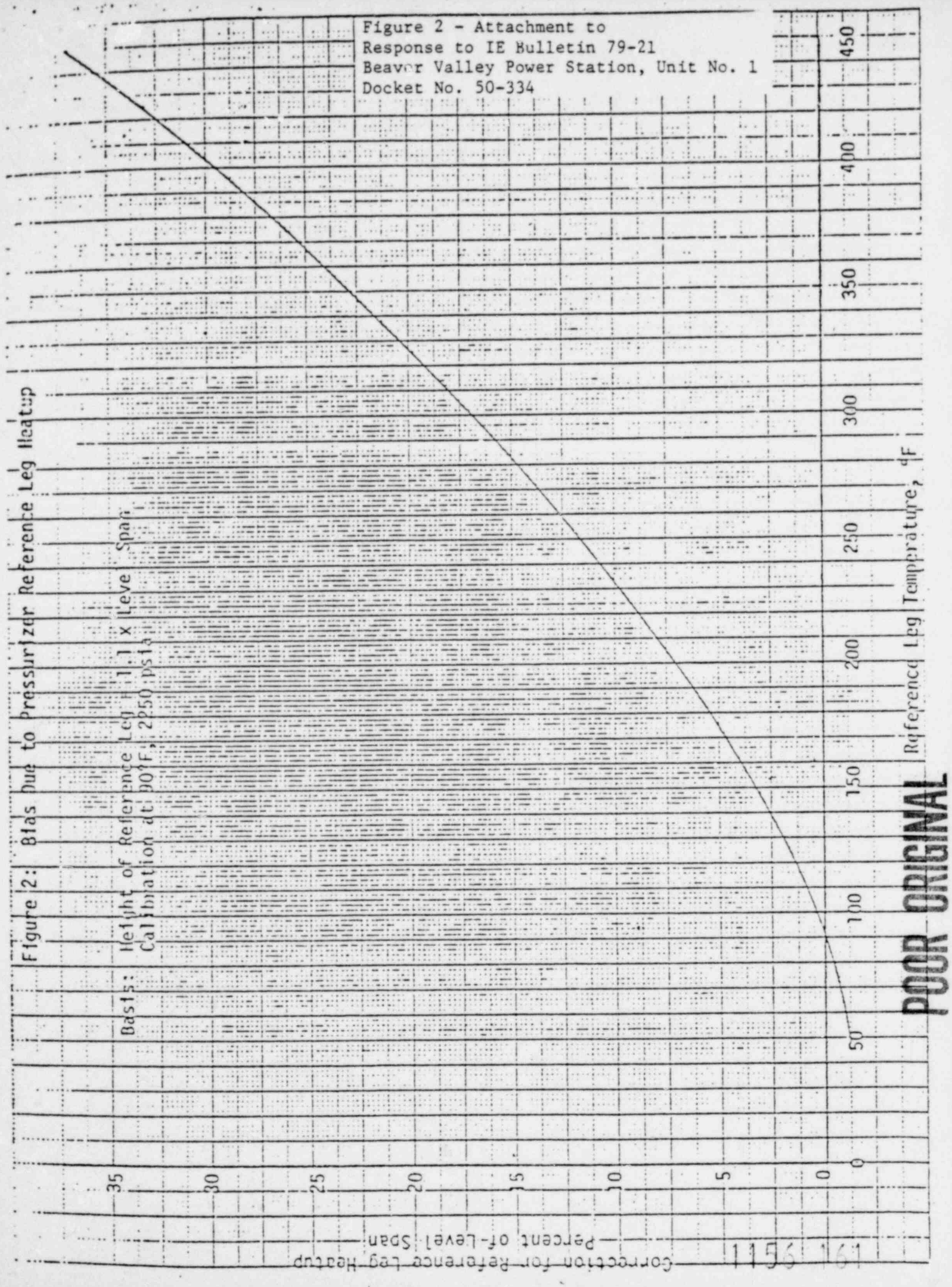

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Figure 3 - Attachment to
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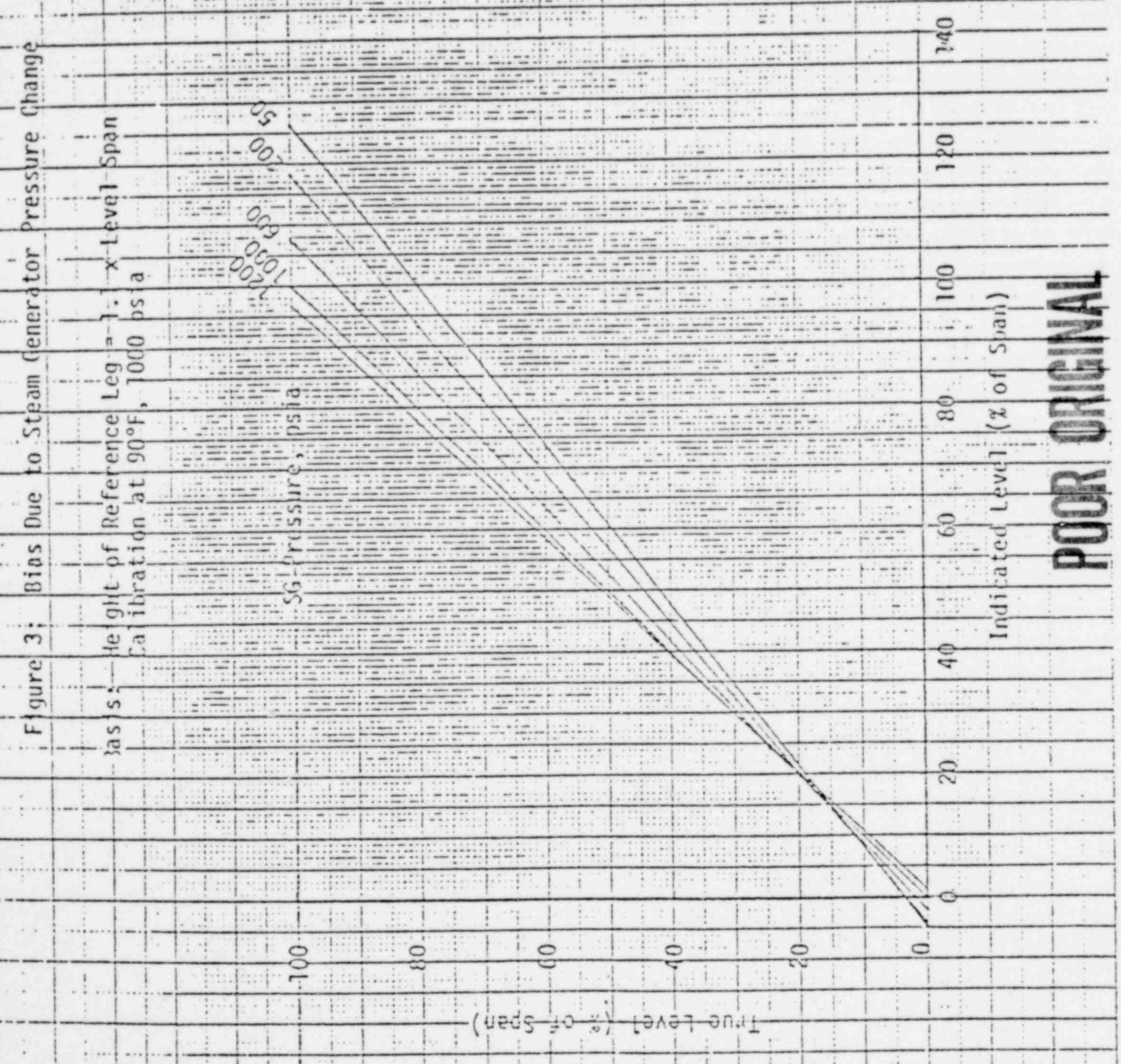

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Figure 4 - Attachment to
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