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ATLANTA, GEORGIA

VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

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September 14, 1979

Mr. James P. O'Reilly, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

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Subject: IE Bulletin 79-21
Surry Power Station
Units 1 and 2

Dear Mr. O'Reilly:

We have reviewed the subject bulletin "Temperature Effects on Level Measurements", and are forwarding the attached response for Surry Power Station Units 1 and 2.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

Attachment

cc: Director
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

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Attachment

IE Bulletin 79-21
Surry Power Station
Units 1 and 2

NRC Question 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 to Question 1

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

A. Steam Generator Narrow Range Water Level

Open Reference Leg, D/P System

- 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

Steam Generator Wide Range Water Level

Open Reference Leg, D/P System

- Post accident monitoring function

B. Pressurizer Water Level

Sealed Reference Leg, D/P System

- Reactor Trip on high water level
- Post accident monitoring function

NRC Question 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 measurement. The results of this evaluation should be presented in a tabular form similar to Tables 1 and 2 of Enclosure 1.

Response to Question 2

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 magnitude of this bias is addressed in the response to Question 3.

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 typical dry containment plants, recent containment analyses performed using Westinghouse models indicate that such boiling would not occur. (In order to assure that this is the case for any Surry plant, specific plant analyses will be required.)

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. Prediction of the effects of rapid depressurization requires complex calculations. Determination of this bias for Surry will be performed by Westinghouse. No completion date is available at this time.

NRC Question 3

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. 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 to Question 3

Safety Function Setpoints

A. Steam Generator Narrow Range Water Level Trip Setpoints

The only high-energy line rupture within containment for which the steam generator water level provides the primary trip function is a feedline rupture. For such a case the low-low water level trip must be actuated when the pressure difference between the narrow range level taps corresponds to a zero-level value. Thus the trip setpoints must be at or above the value that would be indicated at zero true level. Because large steam generator pressure changes are not expected before the trip, only the reference leg heatup effects need be considered, and not the effects of system pressure changes.

A determination of setpoints for the steam generator low-low level trips is as follows:

	Surry Unit 1	Surry Unit 2
Bottom of span (%)	0	0
Normal channel accuracy (%)	± 5	± 5
Post accident transmitter error (%)	0	±10
Reference leg effects (post accident heatup) (%)	+10–0	+10–0
Minimum allowable setpoint.	15%	25%

Note that Surry 2, which has qualified Rosemont 1153 Δ/P transmitters, is subject to ± 10% post accident transmitter error. Surry Unit 1 is scheduled to have these transmitters installed during the upcoming steam generator replacement project.

The value of +10%, –0% used for reference leg effects in this example was obtained from Westinghouse, assuming that the reference leg temperature does not exceed 280°F prior to reaching the High containment pressure setpoint. This is consistent with the Surry FSAR.

The above setpoint revisions have been made on those trip setpoints that provide primary protection for accidents that result in an adverse environment inside containment.

Westinghouse is evaluating two alternate long term solutions which will allow steam generator water level trip setpoints. The two systems under consideration are as follows:

- Mechanical compensation of sealing reference legs
- Temperature compensation of transmitter output.

The time schedule for design and test of the compensated sealed reference leg system is approximately 20 weeks. The schedule for the electrical compensation of the transmitter output is approximately 45 weeks.

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.

NRC Question 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 erroneus 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 to Question 4

As listed in Response 1, the steam generator narrow range water level and pressurizer water level instruments are a part of the Post Accident Monitoring System. Due to reference leg heatup and process variable changes, the indicated parameters may provide erroneous information to the operator following a high energy line rupture. Graphs such as those presented in Figures 1 and 3 (steam generator) and Figures 2 and 4 (pressurizer) could be made available to the operator to aid in interpreting post accident water level indications. Alternatively, the limits of allowable indicated water level could be provided based on conservative upper bound error calculations. These possible solutions are under review to determine the most effective measure for operator interpretation of the level bias. Surry is in the process of reviewing applicable plant emergency procedures and abnormal procedures to determine if changes are required. Any required procedural revisions will be accomplished by November 15.

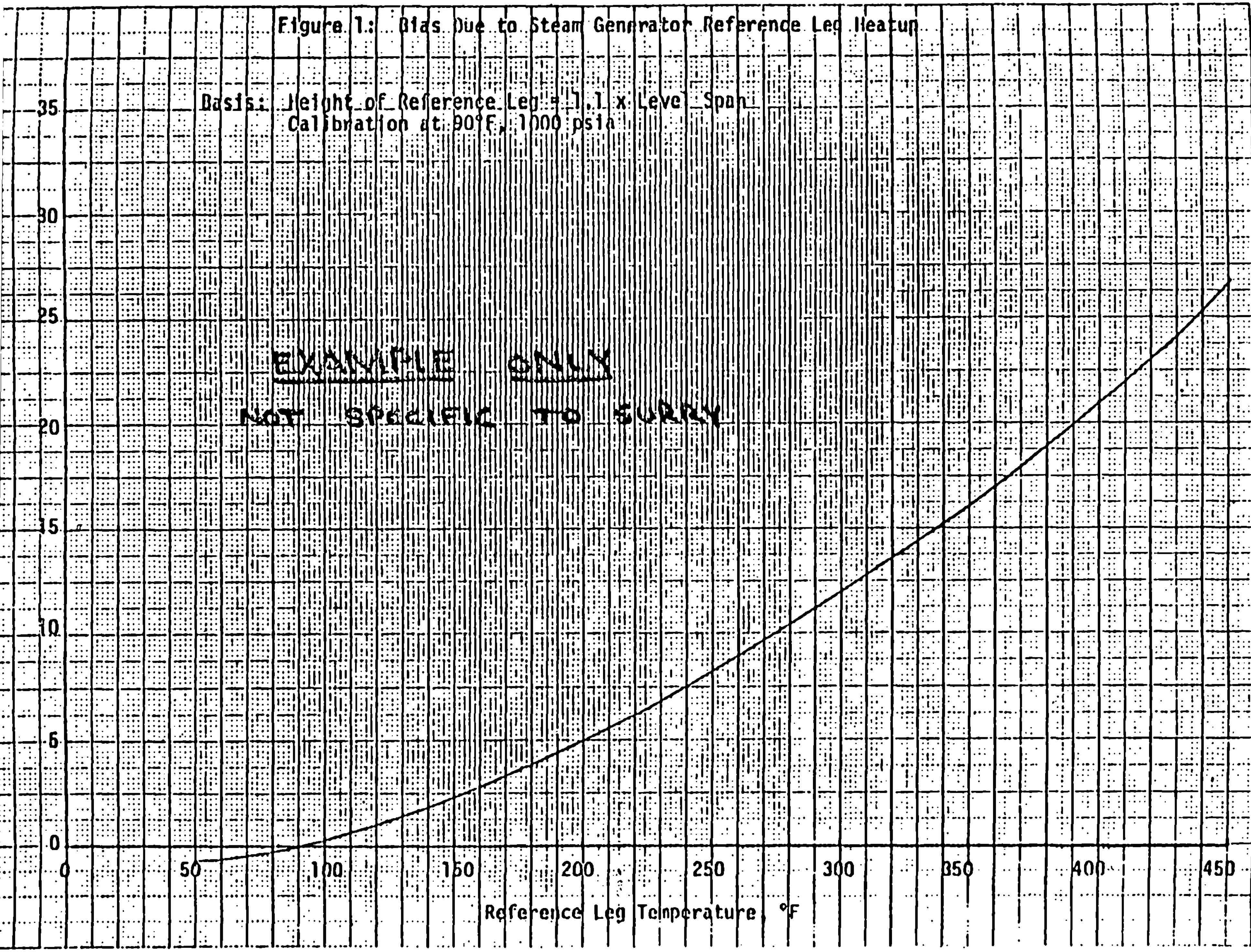

Figure 1: Bias Due to Steam Generator Reference Leg Heatup
Basis: Height of Reference Leg = 1.1 x Level Span
Calibration at 90°F, 1000 psia
EXAMPLE ONLY
NOT SPECIFIC TO SURRY
35
30
25
20
15
10
5
0
0
50
100
150
200
250
300
350
400
450
Reference Leg Temperature, °F

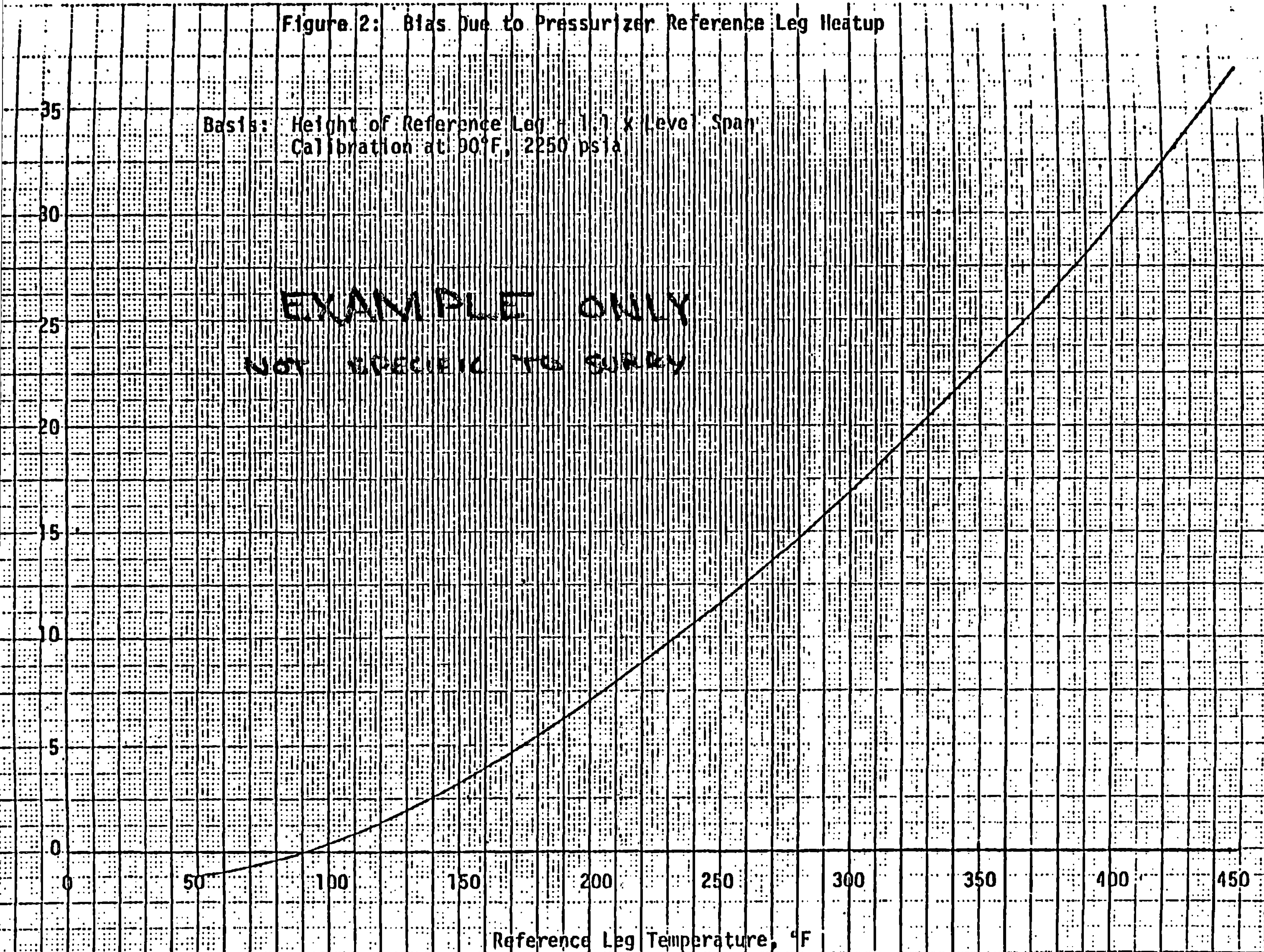

Figure 2: Bias Due to Pressurizer Reference Leg Heatup

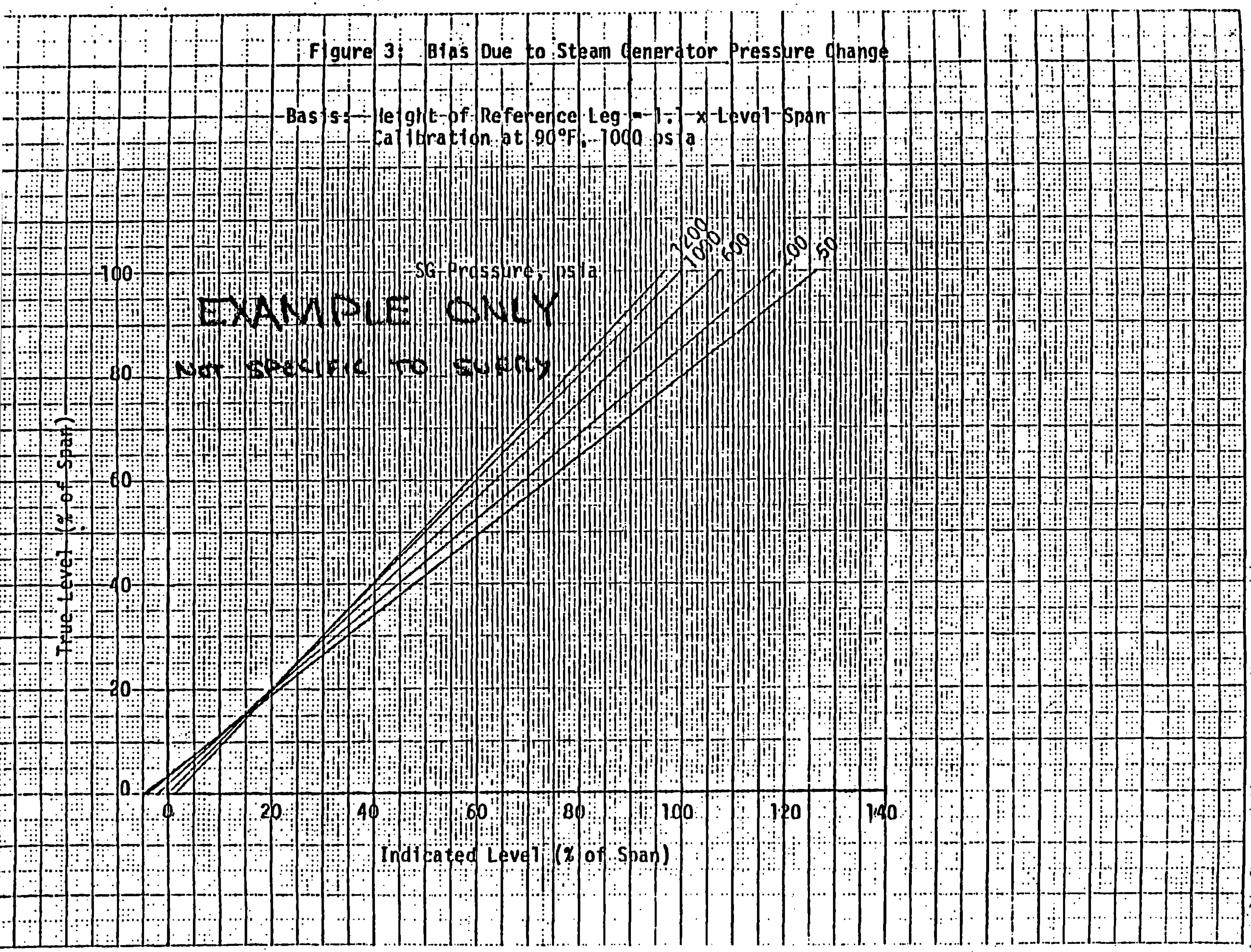
Figure 3: Bias Due to Steam Generator Pressure Change
Basis: Height of Reference Leg = 1.1 x Level Span
Calibration at 90°F, 1000 psia
SG Pressure, psia
1200
1000
600
200
50
EXAMPLE ONLY
NOT SPECIFIC TO SURRY
True Level (% of Span)
0
20
40
60
80
100
Indicated Level (% of Span)
0
20
40
60
80
100
120
140

Figure 4: Bias Due to Pressurizer Pressure Change