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Director of Nuclear Reactor Regulation
Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC DOCKETS 50-321, 50-366
OPERATING LICENSES DPR-57, NPF-5
EDWIN I. HATCH NUCLEAR PLANT UNITS 1, 2
REACTOR WATER LEVEL INSTRUMENTATION IN BWRS
(GENERIC LETTER 84-23)

Gentlemen:

By letters dated November 20, 1984 and May 6, 1985, Georgia Power Company (GPC) responded to Generic Letter 84-23 (dated October 26, 1984). Additional information regarding GPC's response was requested by the NRC staff in a June 5, 1985 telephone conversation. Enclosure 1 provides the requested information.

Please contact this office if you have any further questions.

Very truly yours,

L T Gucwa

L. T. Gucwa

JH/mb

xc: (w/encl.)

Mr. J. T. Beckham, Jr.
Mr. H. C. Nix, Jr.
Dr. J. N. Grace (NRC-Region II)
Senior Resident Inspector

A002
1/1

ENCLOSURE 1

EDWIN I. HATCH NUCLEAR PLANT UNITS 1 AND 2

ADDITIONAL INFORMATION IN RESPONSE TO

GENERIC LETTER 84-23

REACTOR WATER LEVEL INSTRUMENTATION IN BWRS

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ADDITIONAL INFORMATION IN RESPONSE TO GENERIC LETTER 84-23 REACTOR WATER LEVEL INSTRUMENTATION IN BWRs

I. INTRODUCTION

This report provides additional information concerning Georgia Power Company's (GPC's) response to Generic Letter 84-23 as requested by Mr. Wayne Hodges on June 5, 1985.

II. BACKGROUND

Generic Letter 84-23 provided the NRC staff's position regarding reactor water level measurement system improvements necessary to satisfy NUREG-0737 Item II.F.2. GPC responded in letters dated November 20, 1984 and May 6, 1985. To address the concern of level indication errors caused by high drywell temperature, GPC proposed to insulate the unheated portion of the Yarway wide range (-150 to +60") instrument reference legs to delay their heatup and to utilize the Safety Parameter Display System (SPDS) which provides a level reading compensated for instrument line heatup. Regarding the concern of mechanical level instrument reliability, GPC had already installed analog equipment in the Hatch units.

III. ADDITIONAL INFORMATION

A. NRC QUESTION - INSTRUMENT LINE TEMPERATURE MEASUREMENT

GPC was requested to provide justification that the instrument line temperature used in the SPDS density compensation is representative of the entire line, i.e., could a hot spot in an instrument line cause erroneous compensation?

RESPONSE

The Hatch SPDS estimates instrument line temperature based on drywell air temperature. In order to address this question, it is necessary to justify that (1) the average drywell air temperature used in the SPDS density compensation is representative of that near the instrument lines, and (2) the nearby drywell air temperature can be used to accurately predict the instrument line temperature.

The SPDS density compensation uses a single value for drywell air temperature, which is a weighted average of the readings of the resistance temperature detectors (RTDs) nearest the level instrument lines. The locations of those RTDs are as follows:

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HATCH UNIT 1			HATCH UNIT 2		
SENSOR NO.	LOCATION (ELEV/AZ)	GROUP	SENSOR NO.	LOCATION (ELEV/AZ)	GROUP
T47-N001A	200'/190°	C	2T47-N001A	188'/10°	C
T47-N001B	201'/200°	C	2T47-N001K	188'/200°	C
T47-N001J	208'/20°	C	2T47-N002	180'/90°	C
T47-N001K	193'/55°	C	2T47-N010	187'/270°	C
T47-N002	186'/87°	C	2T47-N003	162'/90°	B
T47-N010	193'/35°	C	2T47-N009	162'/270°	B
T47-N003	162'/76°	B			
T47-N009	162'/270°	B			

Figures 1 and 2 show the positions of these RTDs relative to the level instrument lines and other equipment in the drywell. The average drywell temperature T_D used in the SPDS density compensation is calculated as:

$$T_D = 0.70(\text{Group B average}) + 0.30(\text{Group C average})$$

The Unit 1 RTDs will be relocated closer to the reference columns during the fall 1985 refueling outage. The Unit 2 RTDs have already been relocated. A review was performed to assess air temperature variations in the upper cylindrical portion of the drywell, where the level instrument reference columns are located. This review addressed:

1. relative positions of the columns, the ambient temperature sensors, and possible heat sources in the form of insulated steam and water pipes;

2. the air distribution patterns caused by the drywell coolers; and

3. historical ambient temperature readings from the RTDs.

Heat sources affecting this area are the feedwater and main steam lines (See Figures 1 and 2). Since the feedwater lines are equally spaced around the annulus between the drywell shell and the shield wall, their temperature contribution is fairly uniform. The main steam lines are arranged with two lines in each of two opposite quadrants, and none in the remaining quadrants. Since the reference columns are located in the quadrants having no main steam lines, they are not subject to localized heating.

Small steam leaks present the possibility of localized heating of a reference leg, while an RTD located in an area of high cooling air discharge could read low. The effect of these variations on the SPDS compensated water level display is slight. For example, a difference of 20°F between the highest reading RTD and the average of the other RTD readings would result in only a

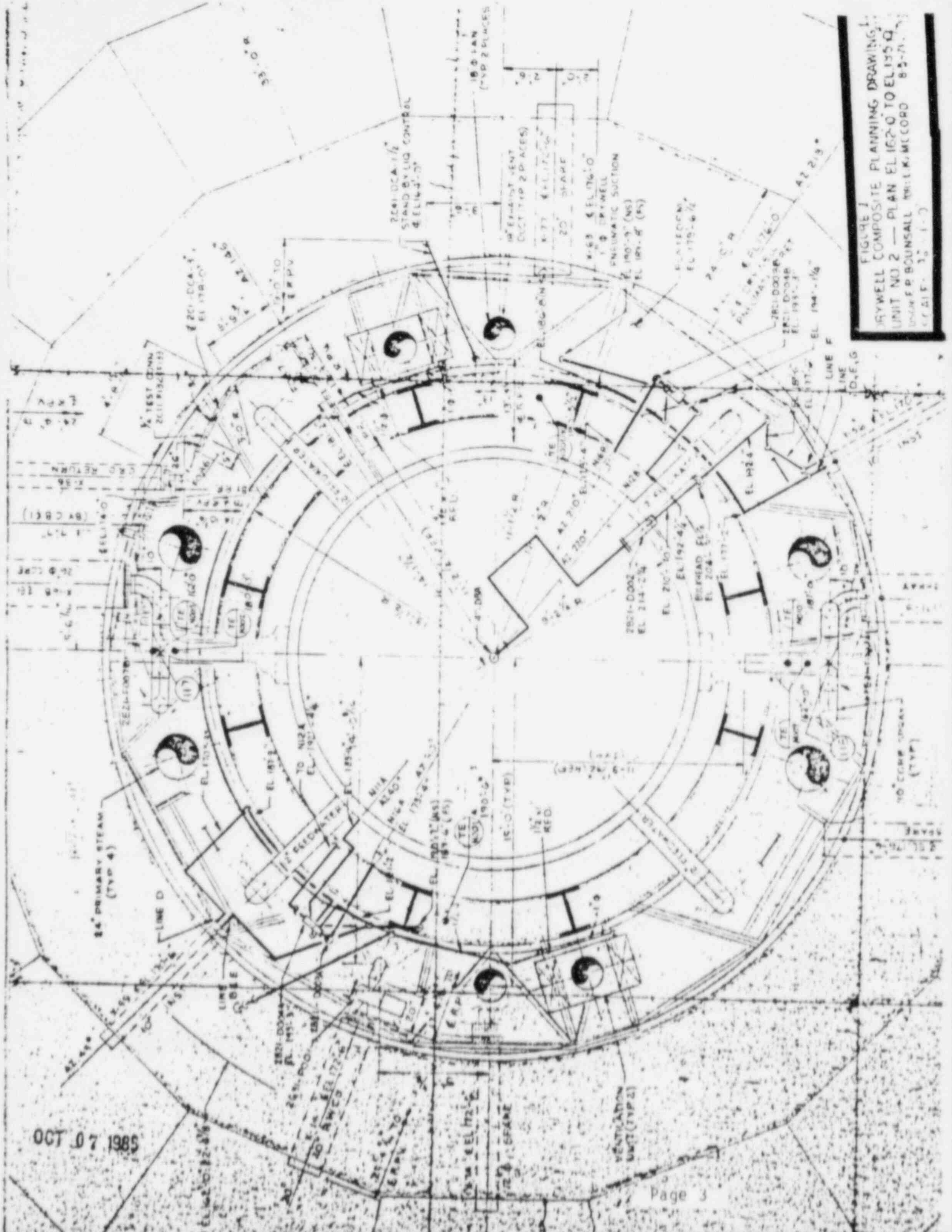

FIGURE 1
DRYWELL COMPOSITE PLANNING DRAWING
UNIT NO 2 — PLAN EL 162'-0" TO EL 175'-0"

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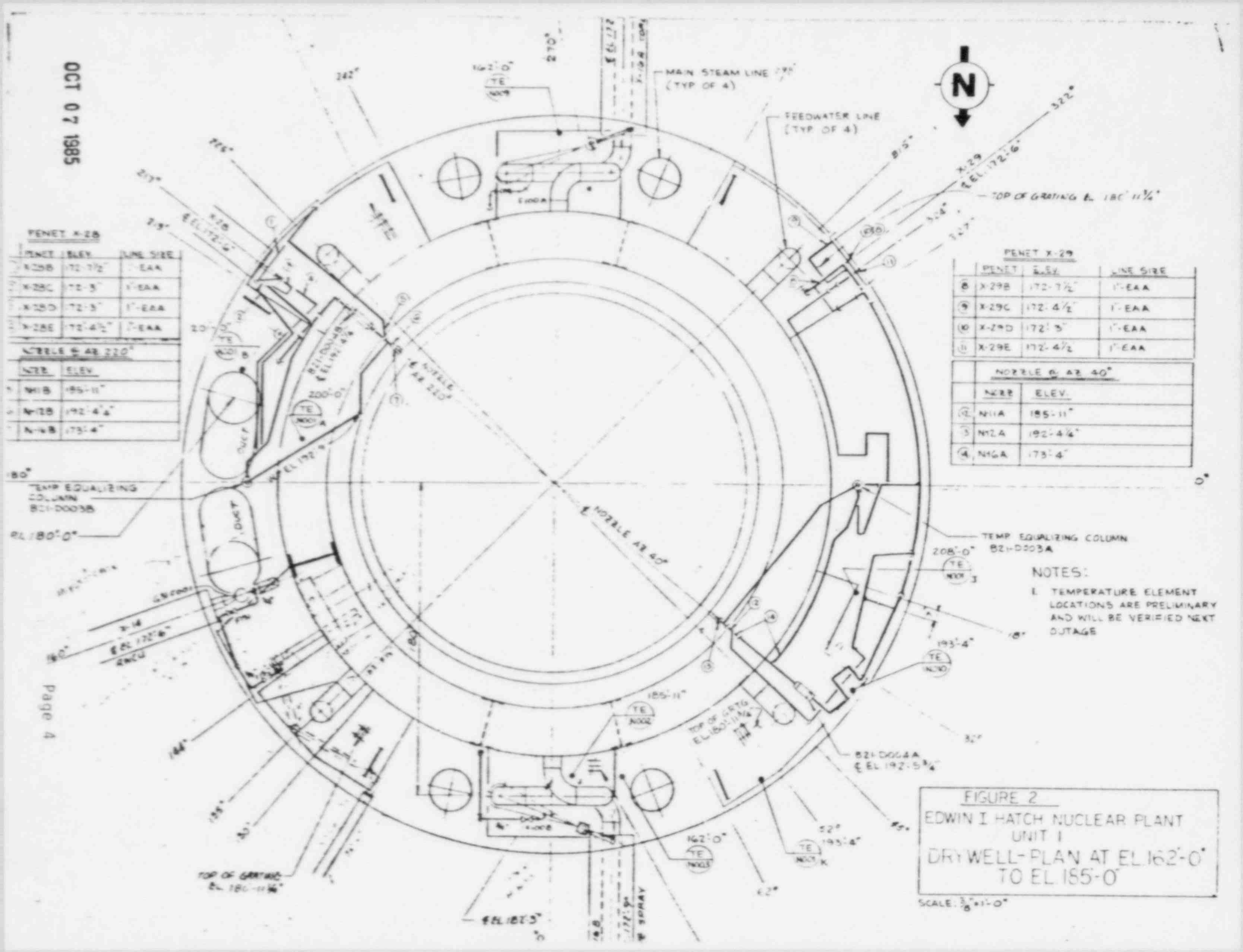

PENET X-28		
PENET	ELEV.	LINE SIZE
X-28B	172'-7½"	1"-EAA
X-28C	172'-3"	1"-EAA
X-28D	172'-3"	1"-EAA
X-28E	172'-4½"	1"-EAA

NOZZLE @ AZ 220°	
NZZ.	ELEV.
N11B	185'-11"
N12B	192'-4¼"
N16B	173'-4"

PENET X-29			
	PENET	ELEV.	LINE SIZE
8	X-29B	172'-7½"	1"-EAA
9	X-29C	172'-4½"	1"-EAA
10	X-29D	172'-3"	1"-EAA
11	X-29E	172'-4½"	1"-EAA

NOZZLE @ AZ 40°		
	NZZ.	ELEV.
12	N11A	185'-11"
13	N12A	192'-4¼"
14	N16A	173'-4"

NOTES:

1. TEMPERATURE ELEMENT LOCATIONS ARE PRELIMINARY AND WILL BE VERIFIED NEXT OUTAGE

FIGURE 2
EDWIN I HATCH NUCLEAR PLANT
UNIT 1
DRYWELL-PLAN AT EL.162'-0" TO EL.185'-0"

SCALE: 3/8" = 1'-0"

two to four inch error during normal operation. Under more critical conditions, such as a loss of drywell coolers or a heatup of the drywell from a LOCA, the error decreases.

The second part of this response depends on the heat transfer characteristics of the Yarway reference column. The SPDS algorithm uses a coefficient defined as:

$$U = \frac{T_{\text{ref leg}} - T_{\text{drywell}}}{T_{\text{reactor}} - T_{\text{drywell}}}$$

This coefficient is assumed to be constant at 0.40 for all water levels.

General Electric has provided data indicating that, for this type of column, the coefficient varies from 0.285 at normal water level to 0.62 at the lower end of its range.

During plant startup testing, the temperatures of the reference columns in both Hatch units were measured directly to confirm the temperature assumed in the original calibration calculation. These tests encompassed the complete range of reactor power levels and recirculation flows. There is some variation in reactor water level on Unit 1 which confirms the dependence of the coefficient on water level. The heat transfer coefficients were found to be:

Hatch Unit 1		Hatch Unit 2	
Ref. Column	U	Ref. Column	U
B21-D003A	0.29	2B21-D003A	0.32
B21-D003B	0.36	2B21-D003B	0.26

The coefficient used by the SPDS is therefore conservative at normal water level. At the lower end of the range the difference between the SPDS coefficient and the GE data could result in a maximum adverse level error of 16 inches. However, the level would need to be at the bottom of the range for 20 minutes or more for the error to reach 16 inches. Furthermore, the error decreases under the more critical condition of high drywell temperature, and the coefficient of 0.62 is believed to be conservative.

It can be concluded from the above that the Hatch SPDS provides a valid density compensated level display based on average drywell temperature in the vicinity of the instrument reference columns.

ADDITIONAL INFORMATION IN RESPONSE TO GENERIC LETTER 84-23 REACTOR WATER LEVEL INSTRUMENTATION IN BWRs

B. NRC QUESTION-DENSITY COMPENSATION LOGIC

GPC was requested to provide a detailed explanation of the logic used by the SPDS to correct reactor water level readings for drywell heatup.

RESPONSE

Several reactor water level signals provide input to the SPDS. The water level inputs are corrected to account for variations in water densities due to changes in reactor pressure and drywell temperature. This information is then displayed, in any of several available formats, on the SPDS computer display.

The following instruments provide the input:

Instrument	Range	Group
B21/2B21-LIS N691A,B,C,D	-150 to +60 in.	B
B21/2B21-LIS N685A,B	-317 to -17 in.	A
B21/2B21-LT N038A,B	-317 to +60 in.	A
B21/2B21-LT N027	-17 to +383 in.	C

Ranges are with respect to instrument zero.

The following is an explanation of the logic and equations currently used to correct these reactor water level inputs for changes in reactor pressure and drywell temperature. We note that this methodology is presently under review and could be revised if improvements are identified.

B21/2B21-LIS N691A,B,C,D,

These instruments provide a reactor water level signal which is partially density compensated because the Yarway reference column inside the drywell is heated by reactor steam condensate. These instruments are calibrated for the following conditions:

Drywell temperature:	135°F
Reactor pressure:	1000 psi which @ saturated conditions corresponds to a temperature of 545°F.

Using the variables defined in Figure 3, the equation for actual reactor water level is determined by equating the pressure differential sensed by the level transmitter to that of the indicated level at calibration conditions:

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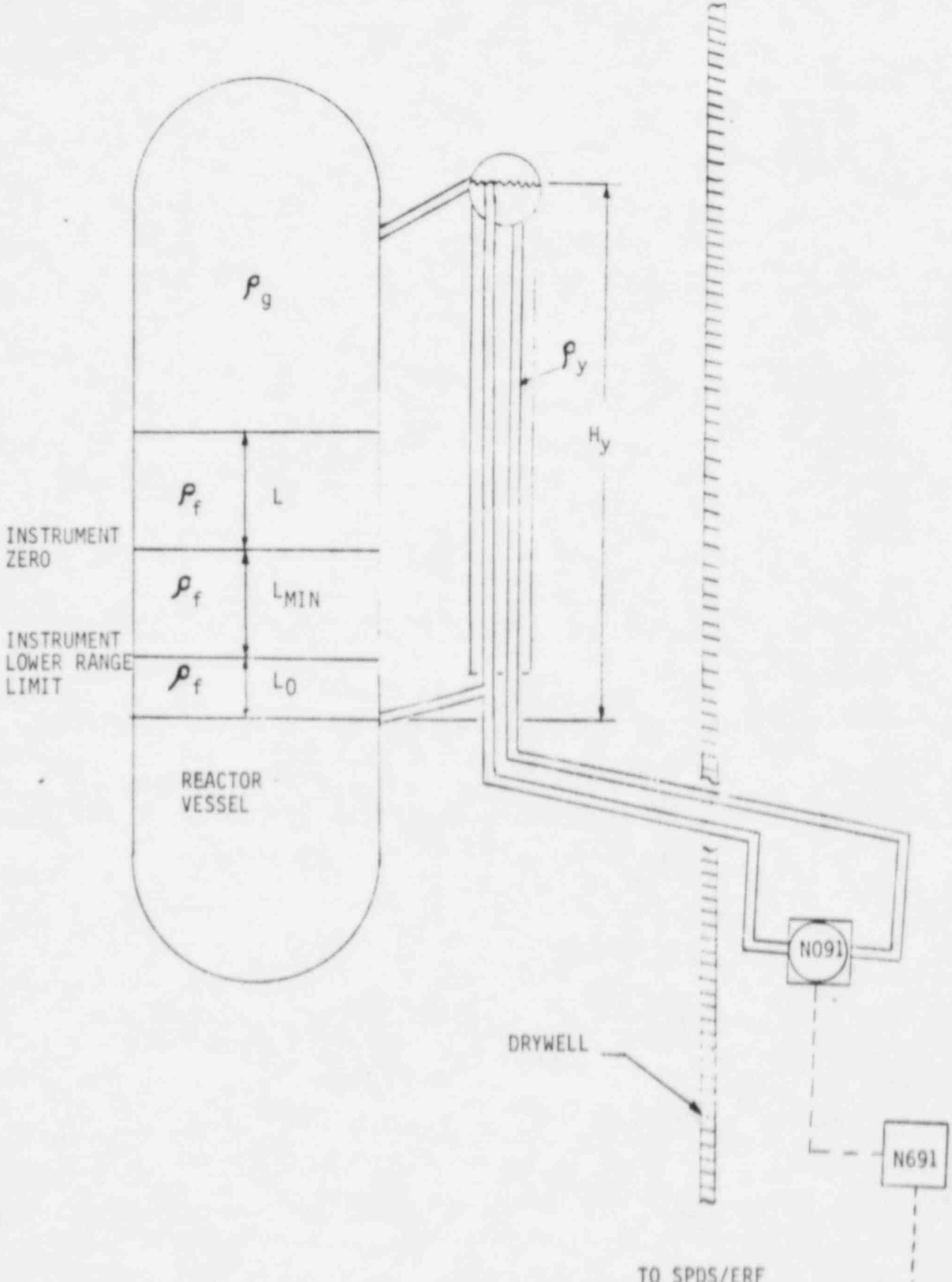
FIGURE 3



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$$(L_A+L_{MIN}+L_O)\rho_f + (H_y-L_A-L_{MIN}-L_O)\rho_g-H_y\rho_y = (L_I+L_{MIN}+L_O)\rho_{fc} + (H_y-L_I-L_{MIN}-L_O)\rho_{gc}-H_y\rho_{yc} \quad \text{(Eq. 1)}$$

where: L_A = Actual Corrected Level; L_I = Indicated (Input) Level — shown as "L" on Figure 3

ρ_{fc}, ρ_{gc}, ρ_{yc} — Densities @ Calibration Conditions

Rearranging Equation 1, the actual corrected water level equation is:

$$L_A = (\rho_{fc}-\rho_{fg})/(\rho_f-\rho_g)[L_I+L_{MIN}+L_O+H_y(\rho_y-\rho_g-\rho_{yc}+\rho_{gc})/(\rho_{fc}-\rho_{gc})]-L_{MIN}-L_O \quad \text{(Eq. 2)}$$

The only variables in Equation 2 that vary with reactor pressure and/or drywell temperature are ρ_g, ρ_f, and ρ_y. (All other variables remain fixed for a particular instrument except L_I, which is the input to the computer). ρ_g and ρ_f are calculated using an algorithm which relates the density as a function of reactor pressure assuming saturated conditions.

ρ_y is a function of reactor temperature and drywell temperature. The temperature of the reference leg in the Yarway column is calculated using the following equation:

$$T_Y = T_D + 0.4\ (T_R-T_D) \quad \text{(Eq. 3)}$$

where:
- T_Y = Temperature in Yarway Column
- T_D = Temperature in Drywell (sensed by RTDs in the vicinity)
- T_R = Temperature of Reactor Water (Calculated by an algorithm which provides the temperature as a function of pressure assuming saturated conditions)

ρ_y is calculated using an algorithm which relates the density as a function of T_y (reference column temperature).

Using the calculated densities and the water level inputs (indicated level), the corrected level is calculated using Equation 2.

B21/2B21-LT N038A,B, B21/2B21-LIS N685A,B and B21/2B21-LT N027

These instruments do not have temperature compensated condensing chambers and are calibrated for the following conditions:

ADDITIONAL INFORMATION IN RESPONSE TO GENERIC LETTER 84-23 REACTOR WATER LEVEL INSTRUMENTATION IN BWRs

B21/2B21-LT N038A,B and B21/2B21-LIS N685A,B

Drywell Temperature:	212°F
Reactor Pressure:	14.7 PSIA which @ saturated conditions corresponds to a temperature of 212°F

B21/2B21-LT N027

Drywell Temperature:	82°F
Reactor Temperature:	82°F

These instruments have different ranges, but the water level inputs are compensated, by use of the same basic equation, for changes in reactor pressure and drywell temperature. Using the variables as defined in Figure 4, the equation for actual reactor water level is determined by equating the pressure differential sensed by the level transmitter to that of the indicated level at calibration conditions.

$$(L_A+L_{MIN}+L_O)\rho_f + X_m\rho_d + (H-L_A-L_{MIN}-L_O)\rho_g-X_r\rho_d - H_r\rho_r = \qquad \text{(Eq. 4)}$$

$$(L_I+L_{MIN}+L_O)\rho_{fc} + X_m\rho_{dc} + (H-L_A-L_{MIN}-L_O)\rho_{GC}-X_r\rho_{dc} - H_r\rho_{fc}$$

where:

L_A = Actual Corrected Level
L_I = Indicated (Input) Level
} Shown as "L" on Figure 4

ρ_{fc} =
ρ_{gc} =
ρ_{dc} =
ρ_{rc} =
} Densities @ Calibration Conditions

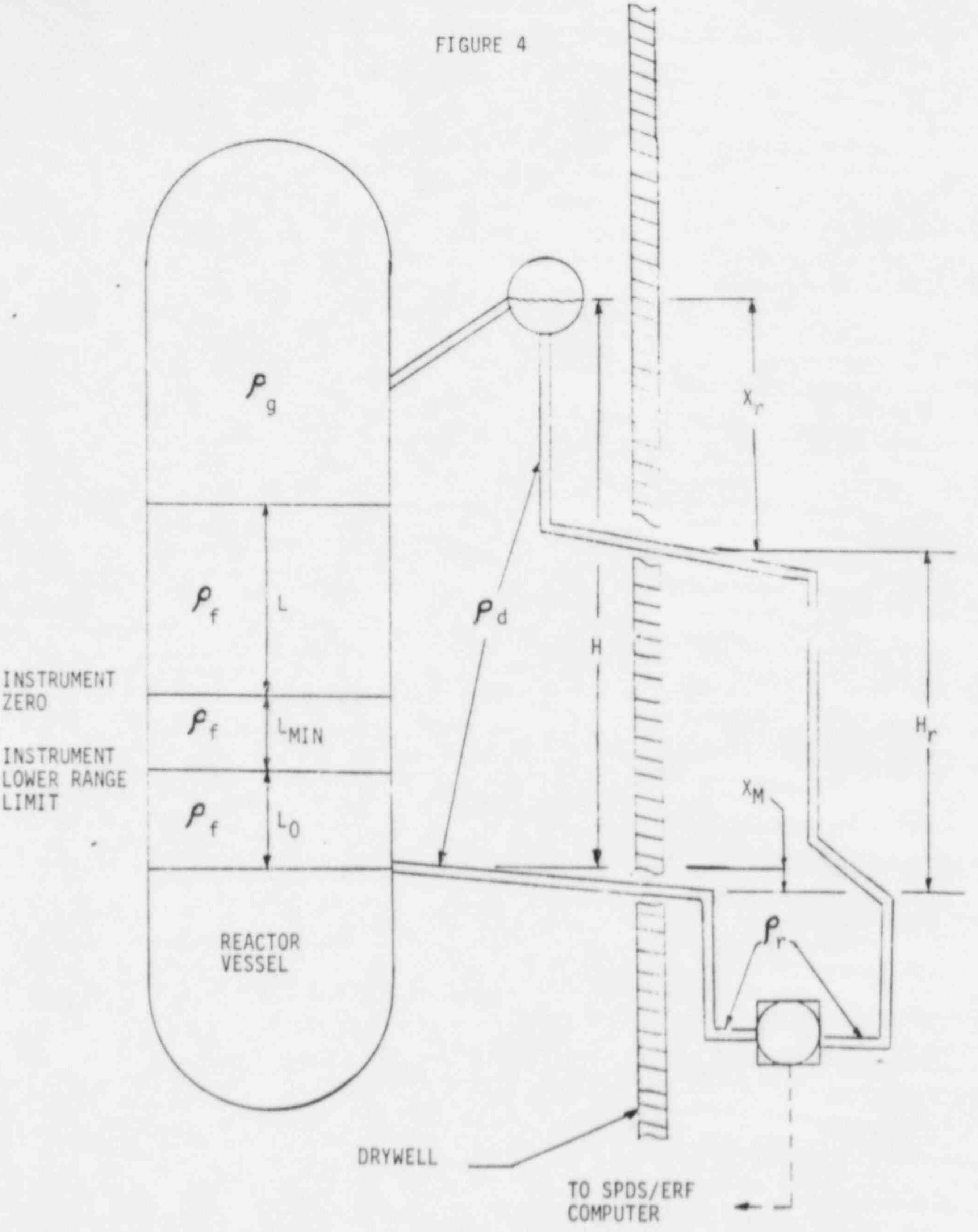
Rearranging Equation 4 and neglecting changes in reactor building fluid density, $(\rho_r-\rho_{rc})$, the actual corrected level is:

$$L_A =(\rho_{fc}-\rho_{gc})/(\rho_f-\rho_g)[L_I+L_{MIN}+L_O+[(X_m-X_r)(\rho_d-\rho_{dc})-H(\rho_g-\rho_{gc})]/(\rho_{fc}-\rho_{gc})]-L_{MIN}-L_O \qquad \text{(Eq. 5)}$$

The only variables in Equation 5 that vary with reactor pressure and/or drywell temperature are ρ_g, ρ_f, and ρ_d. (All other variables remain fixed for a particular instrument except L_I, which is the input to the computer). ρ_g and ρ_f are calculated using the same algorithm used for the B21/2B21-LIS N691 instruments. ρ_d is calculated using an algorithm which relates density as a function of drywell temperature.

Using the calculated densities and the water level inputs (indicated level), the corrected level is calculated using Equation 5.

FIGURE 4



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It should be noted that B21/2B21-LT N038A,B and B21/2B21-LIS N685A,B are not accurate when recirculation pumps are running since the variable leg is pressurized by the jet pump. These instruments contribute to the SPDS level reading only when the recirculation pumps are off.

SPDS DISPLAYS

Reactor water level information is available from the SPDS in several formats. These formats are referred to as the primary display, the core trend display, the reactor water level trend display, and the reactor water level diagnostic display. The SPDS has three color CRTs. One is normally used for the primary display, while the other two can be used for any other display.

The primary display provides a digital level reading and a bar graph representation of water level superimposed on a vessel mimic. The level reading is a weighted average of the corrected level values from available transmitters. The weighting factors assigned to each transmitter in the three level transmitter groups are: Group A=0.25, Group B=1.0, and Group C=0.5. The Group A transmitters are only used when recirculation pumps are not running. Use of the Group C transmitters when recirculation pumps are running is being re-evaluated. The primary display also provides warning messages. "MAY MISS TRIP" appears when indicated level, plus an arbitrary 4 inch margin, is above the Level 1 trip setpoint with corrected level below the variable leg nozzle. "HOT LEG MAY BOIL" appears when the estimated reference leg temperature, plus an arbitrary $10^{o}F$ margin, exceeds the saturation temperature corresponding to the sensed vessel pressure.

The core and reactor water level trend displays provide a time history plot of corrected water level, a digital reading of current corrected level, and a digital reading of the current rate of change of level. The rate of change is determined by a least-squares fit of the water level values from the last six seconds.

The reactor water level diagnostic display provides separate bar graphs of corrected level for the nine level sensors, comparison of corrected and uncorrected level for each sensor, and digital displays of reactor pressure, drywell temperatures, and current corrected level. The time history of reactor water level for the previous hour is also provided.

C. NRC QUESTION-FLASHING PROBABILITY AND CONSEQUENCES

GPC was requested to discuss the probability and consequences of instrument line flashing at Plant Hatch.

RESPONSE

At high drywell temperatures and low reactor pressures, flashing of the reference and variable legs could occur. Flashing could cause loss of a portion of the reference leg liquid resulting in high indicated levels. (Note that the variable legs would be refilled by water from the vessel.) Insulation of the unheated portion of the wide range reference legs as proposed by GPC would delay the onset of flashing, providing the operator with additional time to recognize the potential for flashing. An estimate of the flashing probability at Plant Hatch, which conservatively neglects the effect of the reference leg insulation, can be made by referring to generic studies performed on behalf of the BWR Owners Group.

The frequency of reference leg flashing can be estimated at 1.2×10^{-3} events/reactor year, based on the work presented in Section 4 of SLI-8218, "Inadequate Core Cooling Detection in Boiling Water Reactors." The events which might lead to reference leg flashing were specifically addressed under the heading of Loss of Drywell Cooling in the process of estimating the total contribution of the water level detection system contribution to core melt frequency.

To produce the estimate given above all of the event trees which include a loss of drywell cooling were identified and the frequency of all end states reached by passing through a loss of drywell cooling were evaluated. This resulted in the following flashing frequency estimates for the event initiation:

Initiator		Flashing Frequency Events/Reactor Year
Manual Shutdown Due to Loss of Drywell Cooling	(T_{MT})	4.68×10^{-4}
Turbine Trip	(T_T)	1.18×10^{-5}
Main Steam Isolation Valve	(T_F)	1.47×10^{-5}
Manual Shutdown	(T_M)	8.31×10^{-6}
Small Break LOCA	(S_2)	5.0×10^{-4}
Medium Break LOCA	(S_1)	1.0×10^{-4}
Inadvertent Opening of Relief Valve	(T_1)	2.3×10^{-6}
Loss of Offsite Power	(T_E)	1.3×10^{-4}
TOTAL		1.2×10^{-3} Events/Reactor Yr.

While these results are based on a generic analysis intended to be representative of BWR plants in general, the applicability to Plant Hatch is reasonable for the following reasons.

Hatch 1 and 2 are similar to the plant on which the generic analysis was based. Further, the dominant contributors (T_{MT} and S_2) are largely independent of plant-specific details. The T_{MT} contribution is based entirely on the analysis of LER data to estimate the frequency of drywell over heating as a transient initiation and an estimate of the reliability of the operator to initiate containment spray. Neither are plant-specific.

The S_2 contribution is based on an estimate of the probability of a small break and an estimate of the reliability of the operator to initiate containment spray as in T_{MT}. The small break probability is only weakly dependent on plant design and the operator reliability is not plant-specific.

Second order contributions (approximately 10%) are made by the S_1 and T_E initiators. The S_1 contribution is only weakly related to plant specific details for the same reasons as in S_2 (see above). The T_E initiator tends to be strongly plant-specific since it is based in part on the plant's frequency of loss of offsite power (0.053/reactor year), the probability of recovering offsite power within 1/2 hour (0.33/demand), and the common mode failure probability of the emergency diesels (0.0011/demand). A qualitative review of the Hatch plants indicated that these estimates are reasonable or conservative.

The contribution of the remaining initiators is so small that even large plant-specific differences would not be expected to change the conclusions that 1.2×10^{-3} events/reactor year is a reasonable estimate of reference leg flashing frequency.

Even in the unlikely event that flashing occurred, no adverse consequences would be expected. Operator training and plant procedures alert the operator to the conditions under which flashing could occur. The SPDS would provide a warning message as the conditions for flashing were approached. The sudden and erratic changes in level indication which accompany a flashing condition would be easily recognizable. Emergency procedures explicitly instruct operators to flood the reactor vessel under such conditions, thus assuring adequate core cooling.

IV. CONCLUDING SUMMARY

The preceding discussion demonstrates that the Hatch SPDS is capable of effectively compensating for reactor water level indication errors caused by drywell heatup. The approach to flashing conditions will be delayed by the addition of insulation on wide range reference legs, and operators will be aware of the possibility of flashing because of previous training, procedural guidance, and warnings provided by the SPDS. Should flashing occur, diagnosis would be straightforward, and operators would be directed by procedures to flood the reactor vessel to assure adequate core cooling.

Based on the above, GPC believes that the proposed plan for compliance with Generic Letter 84-23 assures that the Hatch reactor water level measurement system provides the inadequate core cooling instrumentation required by NUREG-0737 Item II.F.2.