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FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220

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VASSALLO,D.B.	Operating Reactors Branch 2

SUBJECT: Forwards summary of analysis performed evaluating high drywell ambient temp effects on reactor vessel water level setpoint,concluding that lo-lo-lo reactor vessel water setpoint rev appropriate,in response to 841002 ltr.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/300 ERIE BOULEVARD WEST, SYRACUSE, N.Y. 13202/TELEPHONE (315) 474-1511

November 26, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Re: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Dear Mr. Vassallo:

Your October 2, 1984 letter provided Amendment No. 64 to the Nine Mile Point Unit 1 Technical Specifications. Your letter also requested additional information relative to low and low-low reactor vessel water level setpoints. An analysis was previously performed evaluating high drywell ambient temperature effects on reactor vessel water level instrumentation. That analysis concluded revision to the low-low-low reactor vessel water level setpoint would be appropriated and resulted in Amendment 64. The analysis also concluded that changes to the low and low low reactor vessel water level setpoints were not required. A summary of that analysis is enclosed for your information.

Sincerely,

NIAGARA MOHAWK POWER CORPORATION

C. V. Mangan
C. V. Mangan
Vice President
Nuclear Engineering and Licensing

RJP/bd
Enclosure

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Drywell Temperature Effects on
Reactor Vessel Water Level Instrumentation

I. Effects on Loss of Coolant Accident Analysis

Analyses were performed to determine the effect of the differences between measured and actual reactor vessel water level on the results of the loss of coolant accident analysis for Nine Mile Point Unit 1. Table 1 summarizes those systems of benefit during a loss of coolant accident whose initiations are prompted by a water level signal.

A. Assumptions

The following conservative assumptions were utilized in the analysis:

1. Systems required which initiate on low water level (i.e., scram) during a loss of coolant accident were assumed to initiate on low water level without a delay in initiation. This assumption is realistic since low water level is attained relatively quickly and sufficient time does not exist for drywell temperature to affect the water level indication instrumentation.

2. Systems required which initiate on double low or triple low water level during loss of coolant accident were assumed to initiate on triple low water level as indicated by the instrumentation.

3. Systems which initiated on either water level signals or other signals were assumed to initiate only on the water level signal.

4. It was assumed that drywell temperature immediately rose to 340 degrees F following the accident. However, due to the thermal time constant of the water level indication instrumentation, heat up of this instrumentation and corresponding differences between measured and actual reactor vessel water level were assumed to follow the relationship shown in Figure 1.

B. Calculational Approach

Table 2 summarizes the loss of coolant accident break spectrum considered in the analysis. This break spectrum bounds breaks at Nine Mile Point Unit 1 since postulated breaks smaller than those analyzed will result in lower peak clad temperatures. The following procedure was utilized to determine the effect of the water level difference on the results of the loss of coolant accident analysis, specifically peak clad temperatures.

1. Time to reach triple low water level was determined for each break analyzed.

2. Based on the times calculated in 1 above, the difference between the indicated and actual vessel water level was calculated from the thermal time constant of the water level instrumentation, Figure 1.

3. Based on the water level differences calculated in 2 above, the additional time required to initiate the systems (i.e., core spray, containment spray, Automatic Depressurization System and isolation) were determined.

4. Based on the additional time requirements calculated in 3 above, revised peak clad temperatures were determined. It was assumed that a delay in the systems (i.e., core spray, containment spray, Automatic Depressurization System and isolation) would result in an increase in the peak clad temperatures by an amount equal to the rate of the cladding temperature rise prior to the delay. Peak clad temperature increases were determined during the delay of the core spray system since core spray is the system that turns around the temperature rise. The change in cladding temperature during the delay in initiation of the core spray system was conservatively determined from the slope of the temperature rise during the delay period. The additional cladding temperature was added to the previously calculated peak clad temperatures to obtain the revised peak clad temperatures due to a delay in the initiation of the water level initiated safety systems.

C. Results

The results of the analyses are shown in Table 2. As shown the effect of the differences between measured and actual reactor vessel water level on the result of the loss of coolant accident analysis result in a less than 20 degrees F increase in previously calculated peak clad temperatures. Peak clad temperature differences within this realm are within the error band of the results of the loss of coolant accident analysis.

These results are conservative due to the assumptions utilized above. These results appear consistent with that which would be expected since during the accident, initiation is relatively fast and water level instrumentation heat up with the associated inaccuracy is slow. This would result in little change in the projected peak clad temperatures as was the result in this analysis.

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[illegible]

[illegible]

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II. Instrumentation Effectiveness

A. Discussion

Figure 2 contains a sketch of the relative positioning of the water level instrumentation and trip setpoints for low, low-low and low-low-low water level.

As water level drops, indication is received up to the point of the lower tap for the water level instrumentation. As vessel water level falls below this point, the water level instrumentation would not indicate the additional drop and may show the water level to be stable at the level of the lower tap.

Since the lower tap is the lowest level that can be positively monitored by the water level instrumentation, any discrepancy in the indicated to actual water level must be added to the lower tap level to obtain the new lowest positive monitorable level. As an example, if the indicated water level was 10 inches higher than actual, when the actual water level reached the lower tap point, the instrument may indicate water level at the lower tap plus 10 inches. If the actual water level continued to drop, the instrument may continue to show the water level at lower tap plus 10 inches.

Suctioning effects may more than compensate for this phenomena, but qualifying these effects are extremely difficult and, therefore, no credit is taken in this evaluation.

B. Applicability to Nine Mile Point Unit 1

As shown in Figure 2, the lower tap for the water level instrumentation at Nine Mile Point Unit 1 is less than 4 inches below the triple low water level trip setpoint. Therefore, if the indicated to actual water level is greater than 4 inches, the triple low water level setpoint would not be initiated if the actual water level were to drop below the lower tap level. The actual water level could be below the lower tap level but the instrumentation may show the water level at the lower tap level plus 4 inches (i.e. greater than the triple low setpoint).

For the spectrum of loss of coolant accident analyses analyzed above, the greatest variation between indicated to actual water level is less than 10 inches. However, other break sizes not analyzed (i.e. small breaks with slow depressurization and drywell heat up) may affect the indicated to actual level by up to 19 inches. Due to this discrepancy, during certain loss of coolant accidents (i.e. very small water or steam break, slow depressurizing loss of coolant accidents) actual water level may drop below the triple low setpoint but triple low may not be indicated on the instrument, the Automatic Depressurization System may not initiate, core spray initiation time would increase and greater peak clad temperatures may result. Therefore, to account for the difference in indicated to actual water level readings, it is prudent to change the triple low water level setpoint.

III. Recommendations

Based on the aforementioned analysis, the following corrective action is recommended:

Recalibrate the triple low setpoint to level 296'-6" (i.e. 20 inches above the current setpoint). Although the analysis indicates a recalibration 10 inches above the current setpoint may be adequate, the larger recalibration is recommended to account for other possible break spectrums (i.e. General Electric calculates a maximum difference of 19 inches; use 20 inches for added conservatism).

[illegible]

TABLE 1

NINE MILE POINT UNIT 1
SYSTEM INITIATION

System		Initiation	Systems Required During LOCA
SCRAM	L	water level or high drywell pressure	Yes
Isolation of Primary Coolant	LL	water level	Yes
Emergency Condenser	LL	water level or high reactor pressure	No**
Core Spray	LL	water level or high drywell pressure	Yes
Containment Spray	LL	water level and high drywell	Yes
ADS	LLL	water level and high drywell	Yes*
HPCI	L	water level	No

L = Low Level
LL = Double Low Level
LLL = Triple Low Level

* For large breaks ADS is not required.

** Feedwater line break, main steam line break and some emergency condenser line breaks take credit for one emergency condenser; however, its incremental worth is small in reducing accident effects.

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[illegible]

[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]

[illegible]

[illegible]

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[illegible]

[illegible]

TABLE 2

NINE MILE POINT UNIT 1

EFFECTS OF REACTOR VESSEL WATER LEVEL INSTRUMENT CHANGES

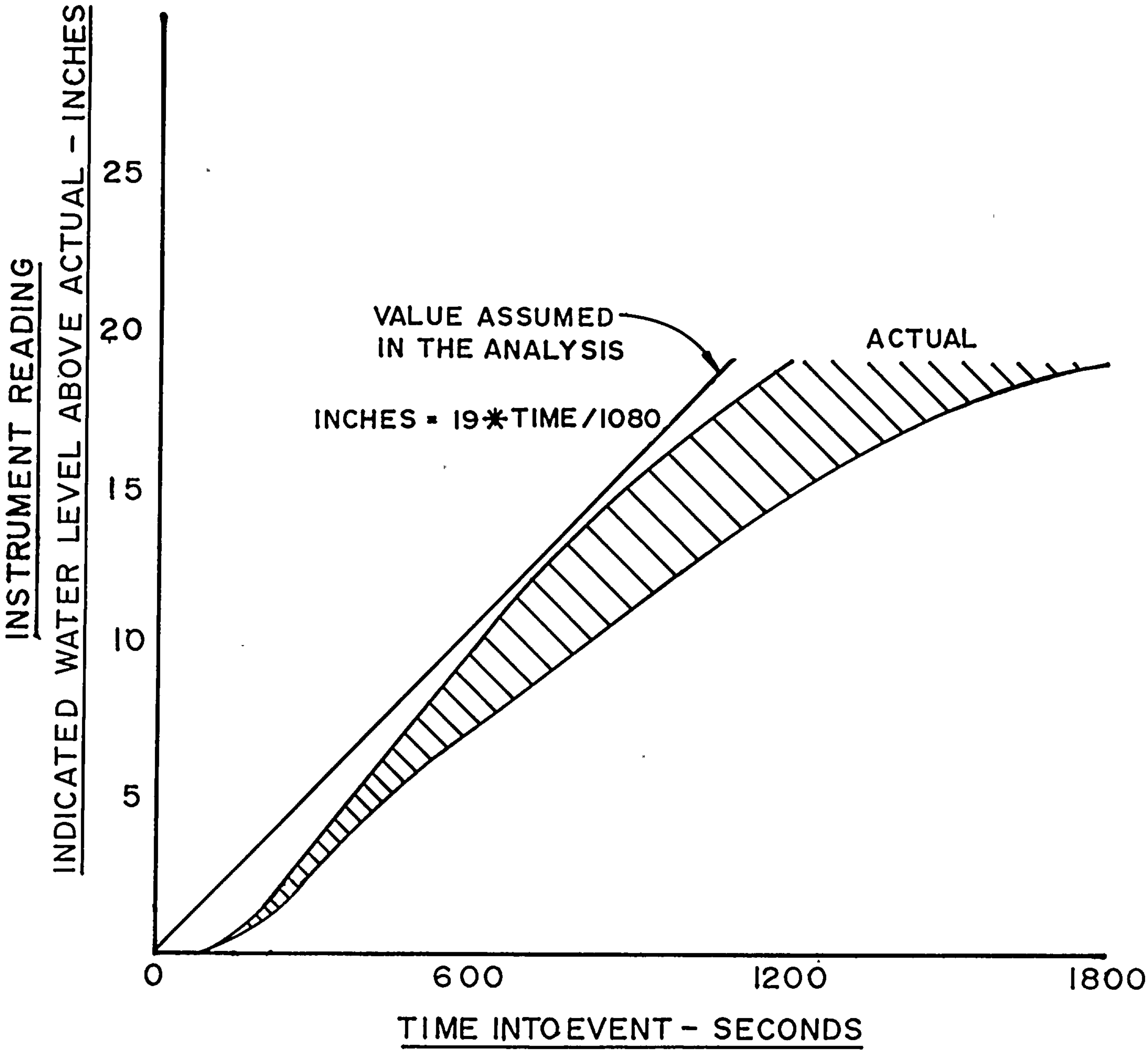
	Break Size*** Ft^2	PCT-F	Time After Accident - Sec L	LL	LLL	Diff. Between Indic. and Act. Water Level-In.	Additional Time Req. to Start Systems-Sec	Projected Additional PCT-F
a.	5.446	2040			< 10	< 1	< 1	< 6
b.	4.357	2040			< 10	< 1	< 1	<10
c.	3.268	2040			< 10	< 1	< 1	<10
d.	1.0	2090			< 15	< 1	< 1	< 6
e.	0.5	2200			< 30	< 1	< 1	< 5
f.	0.2	2195	32	38	< 50	< 1	< 2	<10
g.	0.07	2200	56	70	<105	< 2	< 3	<17
h.	0.275	1860	17	23	< 30	< 1	< 1	< 4
i.	0.567	1300	80	85	< 95	< 2	< 2	~ 0
j.	2.598	1870	48	51	< 55	< 1	< 1	~ 0
k.	0.448	*	245	275	<290	< 6	< 3	~ 0
l.	0.1	*	220	330	<530	<10	<26	**

* No Uncovery

** Although no uncovery occurs in the original case, peak clad temperatures vs. time curves were not available to determine the projected additional peak clad temperatures due to water level instrumentation. However, this change is expected to be < 20°F.

*** Breaks a-g are recirculation line breaks. Break h is a core spray line break; break i is a feedwater line break; break; j is a mainsteam line break; breaks k and l are emergency condenser line breaks.

THERMAL TIME CONSTANT



ASSUMES:

1. INSTANTANEOUS 340°F DRYWELL TEMP.
2. MAINTAIN 340°F FOR COURSE OF ACCIDENT

FIGURE 1

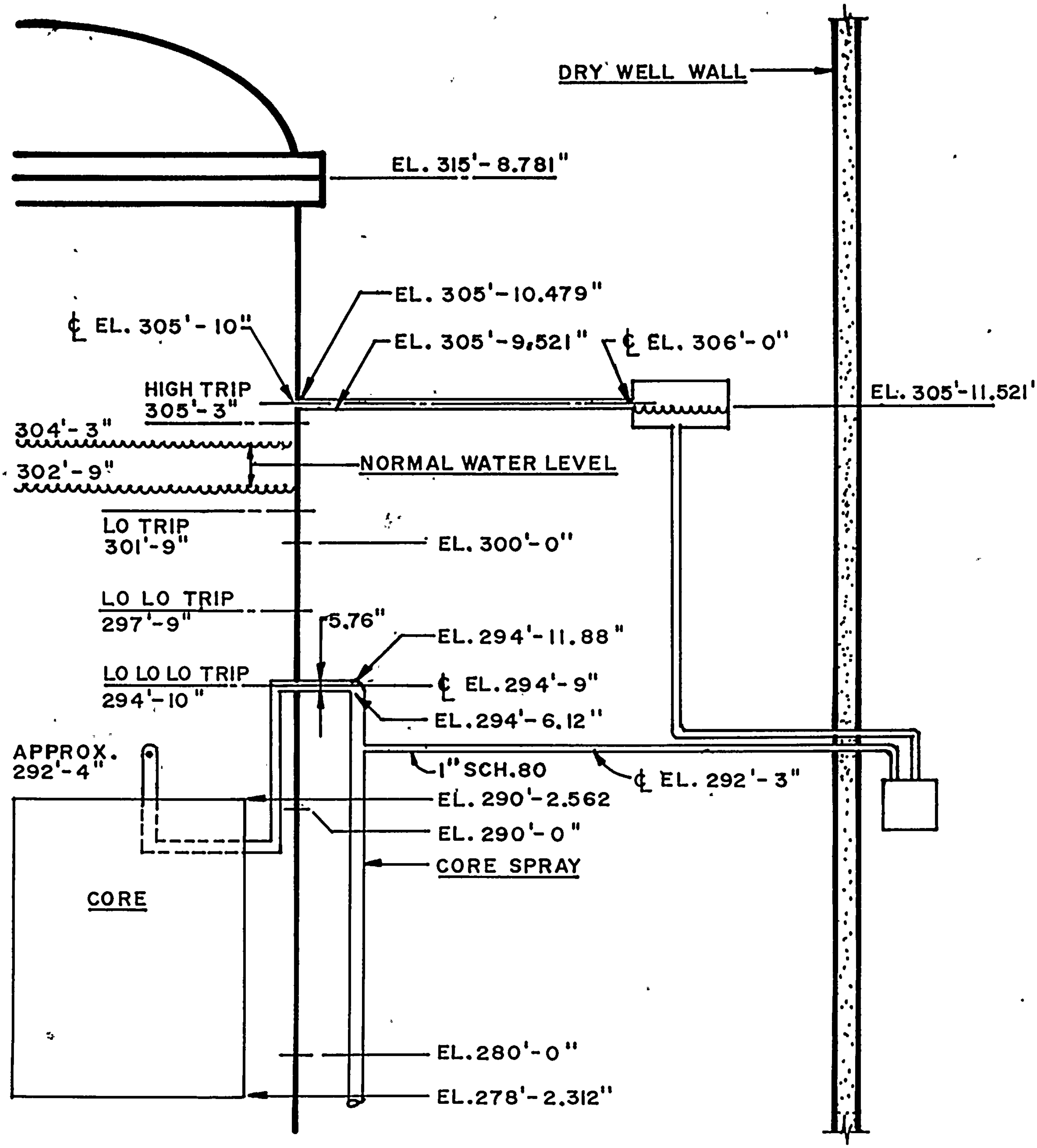

DRY WELL WALL
EL. 315'-8.781"
EL. 305'-10.479"
℄ EL. 305'-10"
EL. 305'-9.521"
℄ EL. 306'-0"
HIGH TRIP
305'-3"
EL. 305'-11.521'
304'-3"
302'-9"
NORMAL WATER LEVEL
LO TRIP
301'-9"
EL. 300'-0"
LO LO TRIP
297'-9"
5.76"
EL. 294'-11.88"
LO LO LO TRIP
294'-10"
℄ EL. 294'-9"
EL. 294'-6.12"
APPROX.
292'-4"
1" SCH. 80
℄ EL. 292'-3"
EL. 290'-2.562
EL. 290'-0"
CORE SPRAY
CORE
EL. 280'-0"
EL. 278'-2.312"

FIGURE 2