UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 03 1985

MEMORANDUM FOR: Ashok Thadani, Chief
Reliability and Risk Assessment Branch, DST

FROM: Brian W. Sheron, Chief
Reactor Systems Branch, DSI

SUBJECT: MAKEUP TO MITIGATE LARGE LOCAS OUTSIDE OF CONTAINMENT AT SHOREHAM

In your memorandum of June 14, 1985, you requested that the Reactor Systems Branch perform an analysis for Shoreham to determine a) the adequacy of a makeup of 1000 gpm from the condensate system for the case of a large LOCA outside of primary containment and b) whether credit could be given for the use of the control rod drive hydraulic system for such LOCAs because of the harsh environment in the Reactor Building with unisolated LOCA outside of primary containment. Our response to this request is provided in Enclosure A.

Wayne Hodges for
Brian W. Sheron, Chief
Reactor Systems Branch, DSI

Enclosure: As stated

cc: RSB Section B Members
R. Caruso
F. Chow

CONTACT: C. Graves, RSB
x29404

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ENCLOSURE A

MAKEUP FROM CONTROL ROD DRIVE HYDRAULIC SYSTEM AND

CONDENSATE PUMPS FOLLOWING LOCA OUTSIDE OF PRIMARY CONTAINMENT

In reference 1, the Reactor Systems Branch was requested to perform an analysis to determine a) whether credit should be given for use of the control rod drive hydraulic system because of the harsh steam environment in the Reactor Building following an unisolated LOCA outside of primary containment and b) the adequacy of 1000 gpm from the condensate system following a large LOCA outside of primary containment. The concern with respect to the supply from the condensate system was bypassing of the supply from the core through the break . Our response to this request is provided below.

I. CREDIT FOR CRD HYDRAULIC SYSTEM AND CONDENSATE SYSTEM FOLLOWING LOCA OUTSIDE PRIMARY CONTAINMENT

The CRD hydraulic system could supply a makeup of roughly 100 gpm to the lower part of the reactor vessel. Although small, this makeup could be significant later in the event. For example, if the supply temperature is 100F, the nominal makeup to match boil-off from decay heat is roughly 260, 205, 135 and 85 gpm at times 1/2, 1, 4 and 24 hours after scram for a 2436 MWt plant such as Shoreham. These numbers do not reflect the importance of the early and rapid supply of ECCS water to quench core heatup for some breaks but are pertinent in consideration of long term decay heat removal.

The ECCS at Shoreham is located at the lowest level (8 ft. elevation) in the Reactor Building which is not compartmented. The CRD hydraulic system, which is located at the 40 ft. level, is in communication with the lowest level (e.g. stairwells). Hence, although it would not be subjected to flooding, it should be subjected to a harsh steam environment for some postulated large breaks in the Reactor Building. Although the CRD hydraulic system should be operating at the time of the postulated event, the pump motor has not been qualified for a harsh steam environment (ref. 2). Hence, no credit for extended operation of the CRD pumps and this source of makeup water should be given.

The postulated 1000 gpm from the condensate pumps is significant with respect to core inventory makeup and vessel pressure for some of the LOCAs outside of primary containment. The temperature of this water is assumed to be 100°F, although the initial temperatures from the main feedwater lines would range from the normal feedwater inlet temperature of 420°F down to the hotwell temperature of 96F and a nominal condensate storage tank temperature of 80°F. As noted above, for 100°F supply water temperature, nominal makeup to match decay heat is roughly 260, 205, 135 and 85 gpm at 1/2, 1, 4 and 24 hours after scram. Since the supply rate is much higher than these boil-off rates, this source can have a significant impact on reduction in vessel pressure. For example, the flow rates at which steam produced by decay heat would be condensed in raising this water to the saturation temperature at 50 psia are roughly 1450, 1130, 740 and 470 gpm at 1/2, 1, 4 and 24 hours after scram. However, this supply may not be effective for some postulated LOCAs outside of primary containment since it may be bypassed

from the vessel by the break. This diversion of the supply may occur even for relatively small breaks. The review of various breaks with respect to diversion is discussed in part II.

II: LOCAS OUTSIDE OF PRIMARY CONTAINMENT AND DIVERSION OF CONDENSATE MAKEUP WATER

High energy line breaks in the Reactor Building and interfacing LOCAs were reviewed to identify those LOCAs outside of primary containment which would result in partial or complete loss of the postulated 1000 gpm makeup from the condensate pumps. The following LOCAs were considered:

Interfacing LOCAS

- LPCS Injection Line
- RHR Letdown Line from RCS
- RHR Heat Exchanger Steam Supply Line
- RHR Head Spray Line
- RHR/LPCI Injection Line

High Energy Line Breaks in the Reactor Building

- HPCI Steam Supply Line
- HPCI Pump Discharge Line*
- RCIC Steam Supply Line
- RCIC Pump Discharge Line*

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* Appendix A of Reference 3 apparently does not include failure of the pump discharge lines although the lines are mentioned in the discussion on page 3-157.

Reactor Water Cleanup Line from RCS

Reactor Water Cleanup Line to Main Feedwater Line

Main Steam Line

Main Feedwater Line

The CRD system and small lines such as the SLC lines, drain lines and pump seal injection lines were not considered.

The following break locations were found to result in partial or complete loss of the postulated 1000 gpm from the condensate system:

RHR Letdown Line

RHR/LPCI Injection Line

RWCU Line from RCS

Main Feedwater Line in Reactor Building

HPCI and RCIC pump discharge line and RWCU Return to Main Feedwater Line**

A) Interfacing LOCAs Resulting in Partial or Complete Diversion of Condensate Supply

Interfacing LOCAs associated with the RHR letdown line and the RHR/LPCI injection lines involve partial or complete diversion of the postulated condensate water supply.

** These breaks are included here but would not result in significant loss of vessel inventory through the break if one of two main feedwater check valves closed as intended.

a) RHR Letdown Line

The 20" RHR letdown line connects to the 28" suction leg of recirculation loop B. This line has inboard and outboard isolation valves (gate valves F008 and F009) which have interlocks to prevent opening or to close the valves if the RCS pressure exceeds a setpoint. Failure of these valves and a resulting break in the low pressure portion of the RHR system could result in a large break in the Reactor Building. Following closure of the recirculation pump discharge valves, the leakage path from the RPV,as shown in Figure 1, is the recirculation pump suction line of loop B which connects to the vessel near the bottom of the downcomer. This leakage path would result in the loss of essentially all of the 1000 gpm from the condensate pumps which enters the pressure vessel at the main feedwater sparger. Note that the postulated flow of subcooled condensate water is sufficient to condense the steam generated by decay heat within a short time after blowdown. However, the condensed steam and the condensate supply are lost through the break.

b) RHR/LPCI Injection Lines

The 24" LPCI injection lines connect to the 28" pump discharge legs of the recirculation loops. Each of the two LPCI injection lines has a testable check valve (F050A or F050B) inside primary containment and a normally closed outboard isolation valve (F015A or F015B). In the event of a complete failure of these valves and

the low pressure piping, the initial limiting flow area would be large but would be reduced to that for 10 jet pump nozzles (about 1/2 ft2) in the affected recirculation loop if the recirculation pump discharge valve closes as intended after the vessel pressure has reduced to roughly 300 psi. After this valve closes, the leakage path and the postulated makeup water path from the condensate pumps are as illustrated in Figure 1. The subcooled condensate falling from the feedwater sparger would be expected to reach the saturation temperature during the approximately 15 ft. drop to the jet pump nozzles. The supply and leakage paths are such that the downcomer region up to about the jet pump nozzle elevation would be expected to be kept full by the makeup water. However, the makeup water would have to enter the core via the throats of the jet pumps in both loops. Since the reduction in subcooling of the makeup water is initially insufficient to condense all of the steam produced by decay heat, the net steam generated would exit the vessel via the jet pump nozzles. The fraction of makeup water swept out to the break through the jet pump nozzles is dependent on a complicated two phase flow division in the region of the jet pump nozzles which should be exhausting a two phase flow under choked flow conditions at upstream pressures of roughly 50 to a few hundred psi simply to remove decay heat after the blowdown. The only experimental data found for this flow situation in the vicinity of the jet pump nozzles were for single phase flow. In view of the uncertainty in the prediction of flow division, one might assume that the 1000 gpm makeup from the condensate pumps is

diverted initially and inadequate. However, as decay heat decreases, the postulated condensate makeup of 1000 gpm is sufficient to decrease the net steam flow enough to ensure that enough flow enters the jet pumps to maintain the level inside the pumps to about the jet pump nozzles and maintain core cooling.

B) <u>High Energy Line Break in Reactor Building Causing Partial or Complete Loss of Condensate Water</u>

a) <u>RWCU Lines From the Recirculation Loops</u>

The 4" lines from each recirculation line suction leg join to a common 6" line inside primary containment which has motor-operated isolation valves (F001 and F004). Lines in the system considered for breaks range from this 6" line down to 3"

A break in the RWCU line involves considerations similar to those discussed for the break in the RHR letdown line since the connection is to the suction leg of the recirculation loop and the vessel leakage path after closure of the pump discharge valve is the same. A 6" break (0.2 ft. 2) is relatively small. However, we assume loss of HPCI and RCIC and depressurization via ADS to a vessel pressure of about 50 psia, the lowest pressure at which the SRVs at Shoreham can be kept open. This break would result in the loss of all of the 1000 gpm from the condensate pumps. When subcooled condensate water gets to the vessel, it would be heated to the saturation temperature during the fall from the feedwater

sparger to the bottom of the downcomer. Hence, this flow could depressurize the vessel to about containment pressure. However, the break flow would still result in loss of essentially all of this makeup water.

b) <u>HPCI and RCIC Pump Discharge Lines and RWCU Discharge Line</u>

The 14" HPCI pump discharge line, 4" RCIC pump discharge line and the 4" RWCU discharge lines are connected to the main feedwater lines at points upstream of the outboard feedwater line check valve. The injection points are at about the 87 foot elevation whereas the main feedwater sparger in the reactor vessel is at about the 132 foot elevation. Assuming at least one of the feedwater check valves operates as intended, loss of inventory from the reactor vessel should cease shortly after the break. In view of the relatively low feedwater temperature and the break locations, the environment in the Reactor Building might not be sufficiently harsh to postulate loss of all mitigating equipment. However, if the vessel were depressurized and condensate pumps used to provide makeup, the break location, minimum vessel pressure via ADS and differences in elevation between the break and the feedwater sparger are such that all of the 1000 gpm from the condensate pumps should be lost from the break. Although the condensate flow rate is sufficient to condense all steam produced by decay heat the condensation process would be limited by stratification well before the water level reached the level of the feedwater sparger.

c) Main Feedwater Line Break

Any main feedwater line break of significance in the Reactor Building should result in the loss of all of the makeup condensate water as discussed above under HPCI and RCIC pump discharge lines.

C) Other Breaks Not Resulting in Diversion of Condensate Water

a) Low Pressure Core Spray

The 10" injection lines from each of the two independent loops of the Core Spray System contain an inboard testable check valve (F006A, F006B) and a normally closed outboard gate valve (F005A, F005B). For a postulated large failure in the valves and low pressure portion of the piping in a given loop, the limiting flow area is about 0.28 ft. 2 (associated with the spray sparger) and the leak path from the vessel is from the core spray sparger in the core exit plenum. Analyses of this type of break for a break inside containment have indicated a relatively mild depressurization until actuation of the ADS. In view of the relatively small break area and the leak path from the vessel (above the core) the 1000 gpm from the condensate pumps should be effective, provided initiation of this source of makeup is timely.

b) RHR Head Spray Line

The RHR head spray line has a check valve, F019, and a normally closed isolation valve, F022, inside containment and a normally closed outboard isolation valve, F023. If this combination of

valves failed, the maximum break size (about .088 ft. 2) and the RPV leak path (upper head) are such that a mild depressurization and relatively small vessel inventory loss is expected. Manual depressurization and makeup via the condensate booster pumps should be adequate.

c) RHR Steam Supply Line

Reactor steam for operation of the RHR system in the steam condensing mode is supplied to each RHR heat exchanger via 8" and 10" lines that come from the 10" HPCI steam supply line. Each line has two isolation valves (F051A and F052A, F051B and F052B). The RHR supply line take-off point from the HPCI steam supply line is downstream of the HPCI isolation valve, F003. Hence, if a break occurs in the RHR steam supply line during normal operation, the break flow rate would be limited by the 1" bypass line around valve F003 and the discussion under the HPCI steam line applies.***

*** LILCo. has stated that the steam condensing mode of operation of the RHR system will not be used.

REFERENCES

Reference 1:

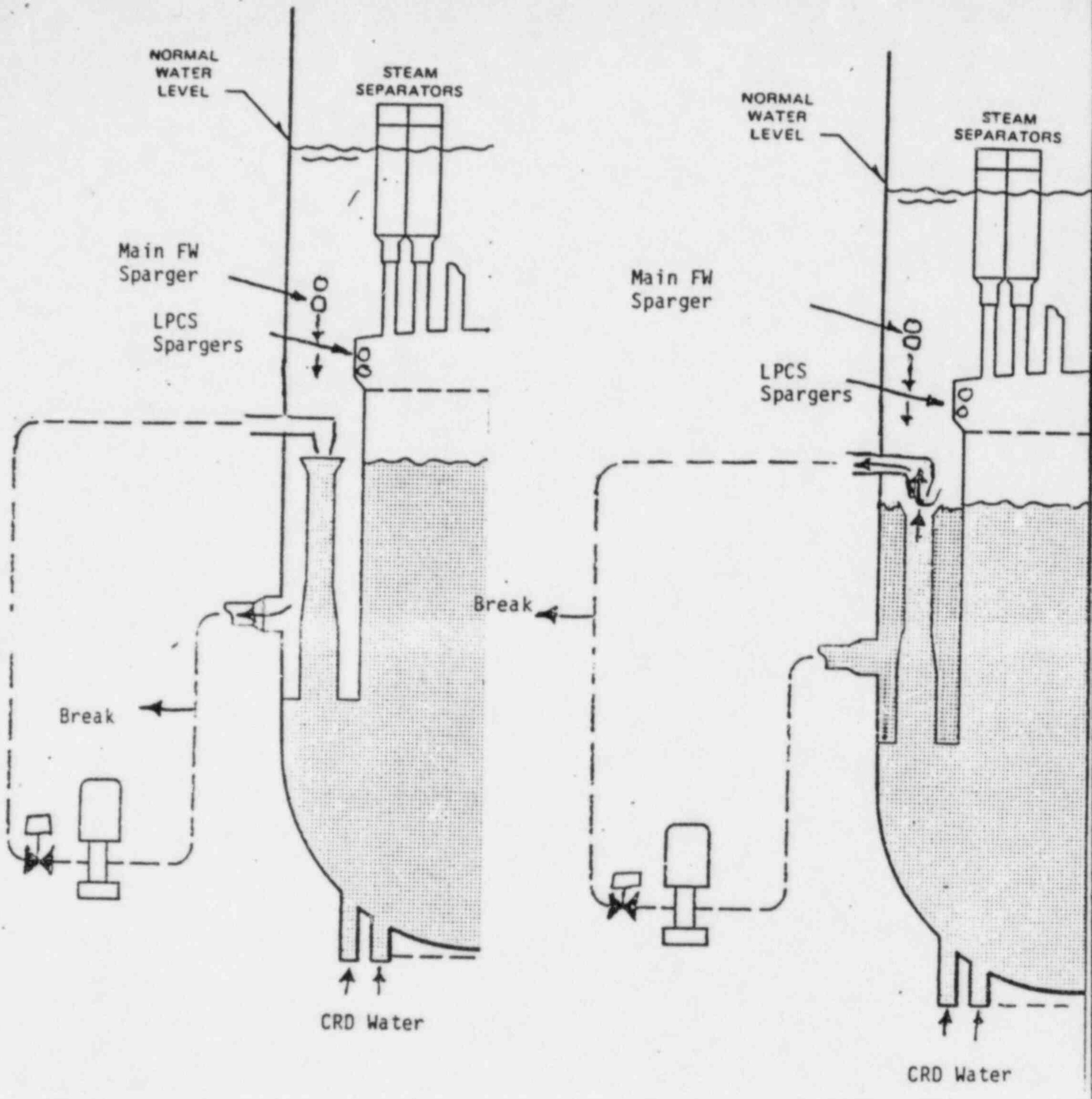
Memo from A. Thadani to B. Sheron, "Makeup to Mitigate Large LOCAs Outside of Containment at Shoreham," June 14, 1985.

Reference 2:

"Environmental Qualification Report for Class IE Equipment for Shoreham Nuclear Power Station Unit 1, Long Island Lighting Company, Revision 5, June 1983," Stone and Webster

Reference 3:

"Final Report, Probabilistic Risk Assessment Shoreham Nuclear Power Station," Science Applications Inc., June 24, 1983.



a) Break in RWCU Line from RCS or RHR Letdown Line

b) Break in RHR/LPCI Injection Lines

Figure 1 Break Paths

LILCO
LONG ISLAND LIGHTING

LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

JOHN D. LEONARD, JR.
VICE PRESIDENT - NUCLEAR OPERATIONS

NOV 16 1985

SNRC-1213

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Unisolated LOCA Outside Drywell
Shoreham Nuclear Power Station
Docket No. 50-322

Reference:
(1) NRC letter (A. Schwencer) to LILCO (J. D. Leonard, Jr.) dated May 6, 1985
(2) LILCO letter (J. D. Leonard, Jr.) to NRC (H. R. Denton) dated June 28, 1985
(3) NRC telcon (R. Caruso) to LILCO (R. Grunseich) September 23, 1985
(4) LILCO telcon (G. Gisonda, et al) to NRC (R. Caruso, et al) dated October 22, 1985
(5) NRC telcon (R. Caruso) to LILCO (J. V. Woodford) dated October 22, 1985

Dear Mr. Denton:

This letter is in response to R. Caruso's request to LILCO to summarize and document the supplemental information provided by LILCO on the above subject matter. This supplemental information was provided in response to an informal request from the NRC in the Reference (3) telephone conversation.

Reference (1) contained the NRC's request for supporting information from LILCO to justify the assumption of successful isolation by high pressure isolation valve closure thus making it possible to estimate core damage frequency during a LOCA outside of the drywell at the Shoreham plant.

In Reference (2) LILCO provided the requested information; however, additional questions were developed by the NRC which required a supplemental response. This was handled informally during the Reference (3) and (4) telephone conversations. The contents of the telephone conversations were as follows:

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- <u>Question 1</u>

  Provide the model numbers of the Velan valves under discussion.

  <u>Response</u>

  The model numbers have been provided during a prior telephone conversation and they are as follows:

  B10-07054B-02WN
  B12-07054B-02WN
  B14-07054B-02WN
  10-B16-0754B-265P

- <u>Question 2</u>

  Did the specification state that the valves should be capable of opening against maximum differential pressure?

  <u>Response</u>

  The specifications do require that capability. This requirement is contained on page 1-11 (copy attached) of Specification No. SH1-088V, Carbon Steel Valves MOV(Cat I).

- <u>Question 3</u>

  Has Velan performed stress analysis on the parts of the valve affected by operating under Δ P conditions?

  <u>Response</u>

  Static stress analysis was performed on the stem, wedge and seat.

- <u>Question 4</u>

  Could the disc deform, flutter or cock in some way so that it would not close?

  <u>Response</u>

  The valve is designed to avoid being subject to phenomena which could preclude valve closure. The wedge guide is the main design feature used to eliminate these phenomena.

LILCO also advised the NRC, during the Reference (4) telephone conversation, that full-flow tests at a 1500 psi pressure differential on an 8" valve of similar design were performed for another utility and that the tests were successful.

It is our understanding, based on the Reference (5) telephone conversation, that the NRC was satisfied with the LILCO responses and clarifications and had no further questions on the issue.

I believe this letter satisfactorily closes out this issue, however, if you or your staff require additional information in the future, please contact my office.

Very truly yours,

J. D. Leonard, Jr.
Vice President - Nuclear Operations

JVW:ck

Attachment

cc: J. A. Berry

permit the motor to attain full speed before the load is 9.38
encountered and load should be shared equally by the two 9.39
lugs making up the hammerblow device.

Bearings shall be ball or roller type throughout, 9.41
designed to facilitate renewal, except guide bushings which 9.42
shall be sleeve type. Seals shall prevent lubricant escape 9.43
or foreign particle entrance

Each motor operator shall be removable from the 9.45
valve for maintenance without dismantling the valve. 9.46

The motor operator shall be weatherproof 9.48
construction, unless otherwise noted. 9.49

All a-c motors shall be rated for operation at 9.51
460 v, 3 phase, 60 Hz. The motors shall have 15 min short- 9.52
time rating and conform to NEMA Standard MG1, June, 1972, 9.53
Sections 10.35, 12.41 and 12.42, unless otherwise noted.
The motors shall operate under all service conditions with 9.54
terminal voltage variation between 10 percent above and 9.55
30 percent below.

All d-c motors shall be rated for operation at 9.57
125 v d-c. The motors shall have 15 min short-time rating 9.58
and conform to NEMA Standard MG1, June, 1972, 9.59
Sections 10.63, 10.64 and 12.62, unless otherwise noted.
The motors shall operate under all service conditions with 10.1
terminal voltage variation between 10 percent above and 10.2
20 percent below.

The motor operator shall open and close gate valves 10.4
against full differential pressure at a rate of 12 in. per 10.5
minute and globe valves at a rate of 4 in. per minute, 10.6
unless otherwise specified on the data sheet.

Each operator shall be designed so that operating 10.8
time can be changed in the field within motor limits without 10.9
disassembling the main gear case. This change can be 10.10
accomplished by changing the motor gear set ratio.

Terminals shall be provided for connecting remote 10.12
control stations or relay contacts to provide for remote 10.13
operation of the valves.

A position transmitter, of the slidewire type, 10.15
shall be supplied if specified. Remote position indicator 10.16
will be provided by the Purchaser.

A limit switch assembly shall be supplied with each 10.18
operator. The device shall be an intermittent geared 10.19
assembly mounted integrally with the operator. The limit 10.21
switches shall be of the open contact rotor type consisting