(2)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

MAR 03 1992

Dear [redacted]

I am responding to the concerns that you provided to us on July 12, 1991, asserting that while Millstone Unit 2 was in heatup in July, 1991, surveillance procedure SP-2402P for the Reactor Protective System (RPS)/Engineered Safety Feature Actuation System (ESAS) RPS/ESAS was performed unusually fast.

We inspected this concern in NRC Inspection Report 91-18, excerpts of which are attached for your information as promised in our August 14, 1991 letter to you. As stated in our August 14, 1991 letter to you, your concern was unsubstantiated, as the 4 hours expended to complete the test on July 4, 1991 was typical of the time it takes to complete the procedure when few setpoints are found to be out-of-specification or require adjustment. (Please note that report 91-18, Section 5.4 contains a typographical error in the last sentence - The date should be July 4, not July 7). All of the test data were properly taken for satisfactory completion of the procedure. Therefore, no further action is planned by the NRC in this matter, and we consider this concern to be resolved.

We appreciate you informing us of your concerns and feel that we have been responsive. Should you have any additional questions regarding these matters, please call me collect at (215) 337-5225.

Sincerely,

Edward Wenzinger

Edward Wenzinger, Chief
Reactor Projects Branch 4

Attachment: Excerpts from NRC Inspection Report 50-336/91-18 (Detail 5.4).

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 7C
FOIA- 92-162

T/35

[redacted]

bcc /w encl:
Allegation File: RI-91-A-0199
E. Conner's files
W. Raymond/T. Shedlosky
Contractor's office files (Meeker)

concurrences:

RI:DRP	RI:DRP	RI:DRP
Barkley	Kelly	Swenzinger
2/26/92	2/28/92	3/3/92

OFFICIAL RECORD COPY

(31)

ALLEGATION RECEIPT REPORT

Date/Time
Received: July 22, 1991 1050 Allegation No. RI-91-A-203

Name: [ ] Address: [ ]

Phone: [ ] City/St./Zip: [ ]

Confidentiality:
Was it requested? No

Alleger's Employer: NNECO Position/Title: Instrumentation and Control Department Technician

Facility: Millstone Unit 2 Docket No.: 50-336

Allegation Summary: Non-seismic gauge assemblies present on High Pressure Safety Injection Pump suction lines. The licensee is aware of potential seismic deficiencies; and because of this, maintains the instrument root isolation valves shut.

Additionally, the Piping and Instrument Diagram of the High Pressure Safety Injection Pumps, designated an "Operations Critical" drawing depicts these three suction pressure isolation valves as being open.

Number of Concerns: 2

Employee receiving allegation: J. T. Shedlosky

Type of regulated activity: Reactor

Functional Area(s): Operations

Detailed Description of Allegation: The alleger called to inform us that a local pressure indicating gauge, PI-3050, attached to the suction line of the "C" High Pressure Safety Injection (HPSI), is not seismically qualified. Although, he called with a concern of the "C" HPSI pump suction line configuration, the alleger stated that it was typical of the installation of the "A" and "B" HPSI pumps.

the alleger stated that the licensee had outstanding questions in regard to the seismic installation and was maintaining the instrument root stop valve shut.

the alleger described the gauge installation configuration as an approximate one foot long 3/4 inch line attached to the pump suction line to the gauge root isolation valve, an instrument dampening snubber and the 4 1/2 inch liquid filled pressure gauge.

The alleger became aware of this installation configuration on Friday, July 19, 1991 after being assigned to replace PI-3050.

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 7C
FOIA- 92-162

H/36.1

ALLEGATION RECEIPT REPORT

Date/Time
Received: July 22, 1991 1050 Allegation No. RI-91-A-203

Name: [ ]

Additionally, the alleger stated that the instrument isolation valves were depicted on the system P&ID, drawing No. 25203-26015 Sheet 2, as being in the open position.

Immediate On-Site Follow-up: The installation was inspected and was found to be as described by the alleger. The inspectors were not able to visually assess the seismic capacity of the installation. The piping, isolation valve and gauge were supported by their attachment to the HPSI pump suction pipe; there were no additional supports.

The root valves, 2-SI-086, 2-SI-088 and 2-SI-090, were observed to be shut; their position was in agreement with the valve alignment requirements stated in surveillance procedures forms SP 2604E-2, Facility 1 High Pressure Safety Injection System Valve Alignment, Revision 11, dated and SP2604F-2, Facility 2 High Pressure Safety Injection System Valve Alignment, Revision 10, dated April 10, 1991.

The P&ID which is designated as "Operations Critical" depicts these valves in the open position.

The Unit 2 Director, Mr. John S. Keenan, was informed of possible seismic deficiencies in this installation; Mr. Theodore Dubay, an on shift senior licensed Supervising Control Operator was informed of the apparent discrepancy in the P&ID for the positions of valves 2-SI-086, 2-SI-088 and 2-SI-090.

36-2

(1)

ALLEGATION RECEIPT REPORT

Date/Time Received: JULY 30, 1991 3:15 PM

Allegation No. RI-91-A-0209 update RI-91-A-0203 update RI-91-A0156 update (leave blank)

Name: [ ]

Address: [ ]

Phone:

City/State/Zip: [ ]

Confidentiality:

Was it requested?	Yes ___	No ✓
Was it initially granted?	Yes ___	No ___
Was it finally granted by the allegation panel	Yes ___	No ___
Does a confidentiality agreement need to be sent to alleger?	Yes ___	No ___
Has a confidentiality agreement been signed?	Yes ___	No ___
Memo documenting why it was granted is attached?	Yes ___	No ___

Alleger's Employer: NORTHEAST NUCLEAR ENERGY CO.

Position/Title: IC TECH

Facility: MILLSTONE 2

Docket No.: 50-336

(Allegation Summary (brief description of concern(s): ① DISAGREEMENT (91-156) WITH NRC FOLLOW-UP OF PAST CONCERN
no action, file in 156
② ADDITIONAL INFORMATION ON (91-209) SEISMIC QUALIFICATION OF HPSI SUCTION GAUGES ③ NO LOOP (91-209) FOLDERS FOR ANNUNCIATOR POWER SUPPLIES
Reg Initiative
look at specific, followup Inspect later

Number of Concerns: 3

Employee Receiving Allegation: P J HABIGHORST
(first two initials and last name)

Type of Regulated Activity
(a) ✓ Reactor
(b) __ Vendor
(c) __ Materials
(d) __ Safeguards
(e) __ Other: ______ (Specify)

Materials License No. (if applicable):

Functional Area(s):
✓ (a) Operations
__ (b) Construction
__ (c) Safeguards
__ (d) Transportation
__ (e) Emergency Preparedness
__ (f) Onsite Health and Safety
__ (g) Offsite Health and Safety
__ (h) Other:

(NRC Region I Form 207
Revised 10/89)

Information in this record was deleted in accordance with the Freedom of Information Act, exemptions 7C
FOIA- 92-162

T/37.1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

SEP 0 3 1991

Docket No. 50-336

Mr. E. J. Mroczka
Senior Vice President - Nuclear
Engineering and Operations
Northeast Nuclear Energy Company
P.O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

Subject: Millstone Unit 2 Inspection 91-18

This refers to the routine safety inspection conducted by Mr. P. Habighorst of this office on June 23 through August 14, 1991, for Millstone Unit 2 in Waterford, CT. The preliminary findings were discussed with Mr. J. Smith and other members of your staff at the conclusion of the inspection.

Areas examined during the inspection are described in the enclosed report. Within these areas, the inspection focused on issues important to public health and safety, and consisted of performance observations of ongoing activities, independent verification of safety system status and design configuration, interviews with personnel, and review of records.

Overall operation of the facility continued to be satisfactory. Your emergency response organization displayed good teamwork, technical support, and management decision-making during the loss of annunciators on July 26.

Your cooperation with us is appreciated.

Sincerely,

for Eugene M. Kelly
Edward C. Wenzinger, Chief
Projects Branch No. 4
Division of Reactor Projects

Enclosure: NRC Inspection Report 50-336/91-18

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5.4 Performance of SP-2402, Safety Parameter Surveillance

The safety parameter surveillance per SP-2402P was performed by the licensee on July 4 in support of Millstone 2 startup.

The inspector reviewed the surveillance controls to determine if it was completed appropriately and if operability of the reactor protection system and engineered safety actuation system was maintained.

The data sheets for SP-2402P completed on July 5 were reviewed and discussed with the I&C foreman. The work was completed under authorized work order M2-91-04789. The work package shows that three technicians performed the test and the foreman approved the results. The testing was done in four hours, for a total expenditure of 12 man-hours.

All sections of the data sheet were filled in, indicating that the testing for all five safety parameters on all reactor protection system (RPS)/Engineered Safety Feature Actuation System (ESAS) RPS/ESAS channels was completed as required. The data shows that the instruments tested satisfactorily for 190 of 191 individual checks of channel performance. The one exception concerned the low steam generator block removal setpoint on channel "A," which was found out of specification, adjusted to within tolerance, and dispositioned. It is notable that all but four of the points checked were acceptable in the "as-found" condition and required no adjustment. Based on a review of the test data and a discussion of the results with the foreman, the inspector concluded the technical specification requirements were performed completely and satisfactorily.

The inspector reviewed the man-loading, test duration, and total manhours for the last six times SP-2402P was performed. The surveillance is usually done with at least three technicians; four technicians did the test on one occasion. The test statistics were as tabulated below.

TEST	No. of Tech	Manhours	Duration
1	4	15	3.75 hrs
2	3	22	7.03 hrs
3	3	15	5.00 hrs
4	3	12	4.00 hrs
5	3	18	6.00 hrs
6	3	12	4.00 hrs

The licensee stated that test times can vary depending on several variables, such as the number of channels needing adjustment, the number of problems found during testing, operator activities that might cause brief suspension of test activities, etc. Based on the above, the inspector concluded that the July 7 test was completed within normal range of test duration times.

6.0 ENGINEERING/TECHNICAL SUPPORT (IP 37700, 37828)

6.1 Review of Safety Evaluations for Modifications Made in 1990

The licensee's annual report for January 1 to December 1, 1990, for Millstone Unit 2 was reviewed. The report identified 34 plant design changes, 19 plant design change evaluations, 26 procedure changes and 27 jumper-lifted lead-bypass changes. The report provided a summary of each change including a description of each change, a reason for the change, and a short safety evaluation that concluded in every case that the change did not constitute an unreviewed safety question per criteria of 10 CFR 50.59.

A sample of nine plant design change reports (PDCR), five plant design change evaluations (PDCE), five procedure changes, and four jumper lifted lead bypass changes were reviewed in depth to determine if acceptable determinations were performed.

Each of the above files contained a safety evaluation which concluded that the change did not constitute an unreviewed safety question per the criteria of 10 CFR 50.59. The safety evaluations were reviewed against procedure NEO 3.12, "Safety Evaluations," and Unit 2 Engineering Departmental Instruction No. 2-ENG-3.06, "Format for Safety Evaluations and Justifications for Continued Operation (JCO)." Each of these procedures addresses seven topics contained in the three aspects of 10 CFR 50.59 criteria for determining whether or not an unreviewed safety question exists. The seven topics are similar to those stated in document NSAC/125, "Guidelines for 10 CFR 50.59 Safety Evaluations." It was noted that the majority of the safety evaluations provided explicit bases for each of the seven determinations. In some cases, there were bases for only some of the determinations leaving the remainder absent of stated bases. A common weakness of those not providing bases for all determinations was to be silent on the question concerning whether or not there was a reduction of safety margin of the basis of the technical specifications. It was also noted that there was a lack of uniformity in the safety evaluations and the 10 CFR 50.59 determinations. Overall, it was noted that there has been a marked improvement in the 10 CFR 50.59 determinations over previous inspections.

6.2 Steam Generator Tube Repairs

On June 28, 1991, NNECO completed repairs to the steam generator tubes The repairs were in response to non-destructive examinations performed during the forced outage which began on May 25. The scope of the steam generator examinations was previously identified in routine inspection report 50-336/91-15 dated July 12, 1991.

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JUL 0 8 1991

[ ]

On June 19, 1991, in a discussion with Richard Matakas of the Region I Office of Investigations, you alleged that three or four years ago, prior to the implementation of 10 CFR 26, unnamed reactor operators were allowed to perform licensed functions, with the knowledge of Station Management, while not fit to be on duty.

The NRC considered this allegation to be very serious in nature and attempted to corroborate the information that you provided. We found the facts you alleged to be incomplete and as a result, the conclusion you reached was inaccurate. Therefore, we have not been able to substantiate your assertions.

NRC takes its safety responsibilities seriously. Issues such as the one you raised require a prompt response by us. Due to the extremely sensitive nature of fitness for duty issues, it is important that the facts about these matters be provided to us as completely and accurately as possible.

Should you have any further questions, or if I can be of further assistance in these regards, please call me collect at (215) 337-5225.

Sincerely,

Edward Wenzinger

Edward Wenzinger, Chief
Reactor Projects Branch 4

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37.2

BLANCH

NORTHEAST UTILITIES

MEMO

March 6, 1989

PSE-SA-89-061

TO: T.Blanchard
Millstone Unit No. 2 Engineering

FROM: D.L. Coleman D. Coleman
Berlin W021, Ext. 3353

SUBJECT: Millstone Unit No. 2
Evaluation of Pressure Gages for
PDCR-2-112-79.

REFERENCE:

1. Millstone Unit No. 2 PDCR-2-112-79.
2. NUSCO Calculation PDCR-2-112-79-1067, GP Rev.0
3. Memo, R.A. Place to J.F. Smith dated June 20, 1986 (NSE-M-86-59)
4. Memo, S.K. Brinkman to T.J.Mawson, dated July 15, 1986.

Per the requests of references 3 and 4, a structural evaluation was performed on the following pressure gage installations:

GAGE ID; PI-	LOCATED OFF LINE NO.	ROOT VALVE	P&ID No. 25203-
5403	4"-HBD(2)-45	2-CN-96B	26005 Sh3
5405	4"-HBD(2)-45	2-CN-97B	"
5401	6"-HBD(2)-45	2-CN-95B	"
6743	20"-HBD(2)-115	2-RB-111B	26022 Sh1
6745	20"-HBD(2)-115	2-RB-111D	"
6747	20"-HBD(2)-115	2-RB-111F	"
3046	6"-GCB-3	2-SI-090	26015 Sh2
3048	6"-GCB-2	2-SI-088	"
3050	6"-GCB-2	2-SI-086	"
3051	14"-GCB-1	2-SI-093	26015 Sh1
3053	14"-GCB-1	2-SI-091	"
3055	10"-GCB-1	2-CS-032	"
3057	10"-GCB-1	2-CS-030	"
7436	8"-HCC-9	2-RW-126A	26023 Sh2
7462	8"-HCC-9	2-RW-126B	"
8859	2"-JBD-58	2-CHW-7	26027 Sh2
8863	2"-JBD-58	2-CHW-26	"

All of the pressure gages are located off of 3/4" root piping, connected to the line numbers listed above. The root piping, and pressure gages are classified as QA, Seismic Class I. The original design of the root piping was performed in accordance with the methods of Bechtel Standard WC-30.

The pressure gages were installed under reference 1. The pressure gages were installed using several unnecessary fittings, couplings and valves. No apparent design criteria was used. The present configuration is not in accordance with established plant design criteria. However, by engineering judgement, a postulated DBE seismic event would not result in a structural failure that would comprimise the integrity of the associated piping system. It is recommended that the present configurations be modified to ensure acceptable system stresses that meet the current design standards.

The pressure gage assemblies should be modified such that the gage and snubber fitting be assembled directly to the coupling fitting adjacent to the root valve. The attached Figure No. 14 shows the proposed typical configuration for all the pressure gages with the exception of PI 3053. The configuration of PI 3053 is shown on the attached Figure No. 15.

A NUSCO calculation was performed to evaluate the root piping (reference 2). This calculation is based upon the recommended modifications described above. The root piping was evaluated for increased deadweight and DBE seismic loadings due to the additional mass of the pressure gage. The installation of the pressure gage has no affect on thermal loadings conditions. The pressure gage fittings were evaluated for pressure, deadweight and DBE seismic loadings. Stresses were calculated and evaluated in accordance with ASME III, 1974 Edition. This meets or exceeds the original design code requirements.

The root piping for pressure gages PI-6743, 6745 and 6747 include 2 root valves. Root valves 2-RB-111B,D and F constitute an anchor for the pressure gages. Root valves 2-RB-240A,B and C constitute an anchor and problem boundary for instrument tubing to pressure sensors PS-6119A,B and C. The calculation in reference 2 determined that the addition of the pressure gages (assuming installed per Figures 14 and 15) will not affect the ability of the root piping to function as an anchor for the instrument tubing.

37.4

In summary, it is recommended that the pressure gages be modified according to Figures 14 and 15. The root piping and pressure gage fittings have been evaluated in the modified condition for all applicable load cases. All calculated stresses are within the Code allowable limits as defined in ASME III, 1974 Edition and are documented in reference 2.

cc: GED
F.R. Decimo
B.J. Duffy
J. Rosater

37.5

NORTHEAST UTILITIES SERVICE COMPANY

SUBJECT: Evaluation of Pressure Gages for PDCR-2-112-79; PI 8403, 8405, 8401, 6743, 6745, 6747, 3046, 3048, 3050, 3051, 3053, 3055, 3057, 7436, 7662, 8659, 8663.	BY D. Coleman DATE 03-03-89
	CHKD. BY NCC DATE 3/7/89
	CALC. NO. PDCR-2-112-79-10678P REV 0
	SHEET NO. C14 OF C15

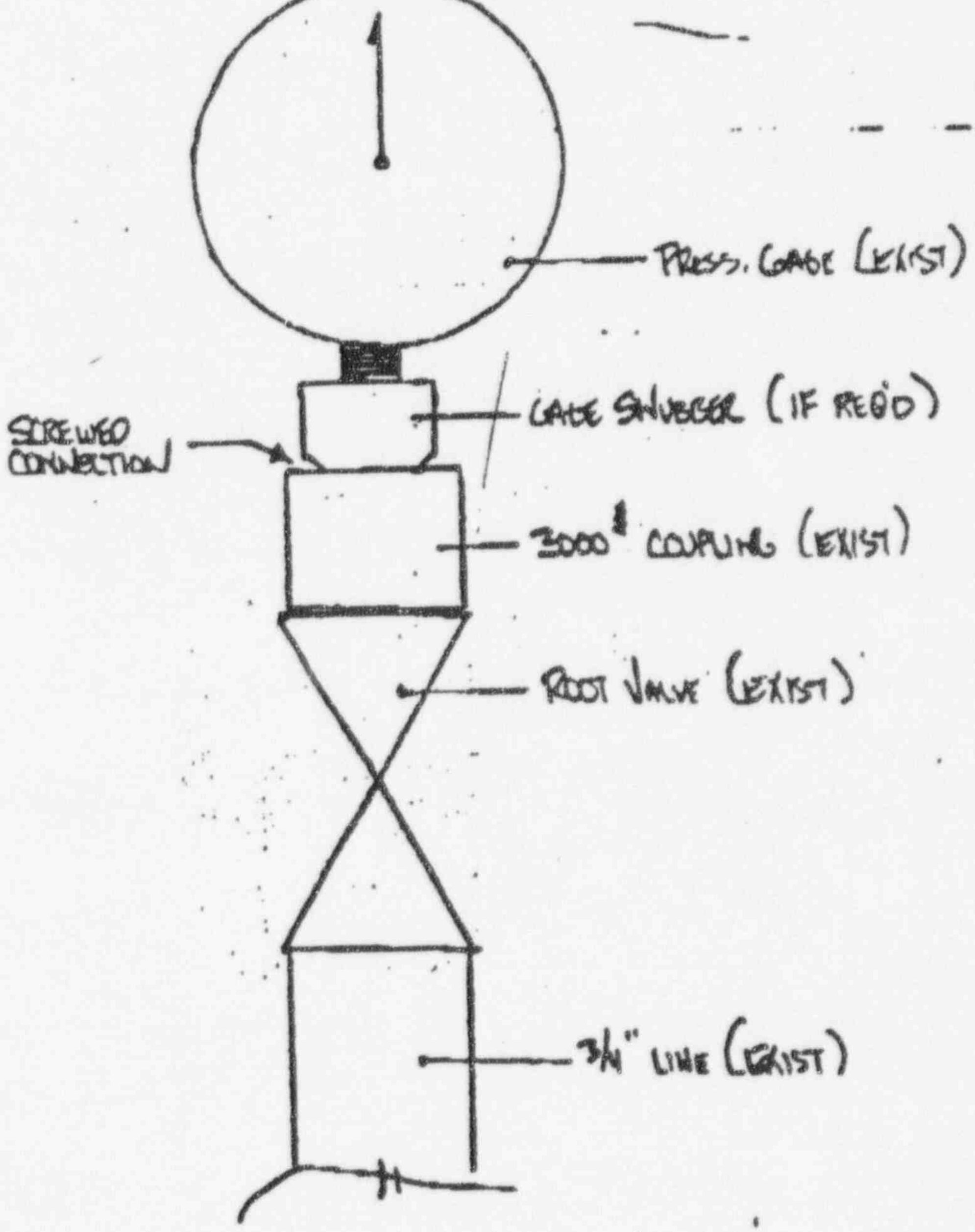

FIGURE NO. 14

37.6

SUBJECT: Evaluation of Pressure Gages for PDCR-2-118-79: PI 5402,5405,5401, 6743,6745,6747,3046,3048,3050,3051, 3052,3055,3057,7436,7662,8859,8863.

BY D.Coleman DATE 03-03-89
CHKD. BY MGC DATE 3/7/89
CALC. NO. PDCR-2-118-79-10676P REV 0
SHEET NO. C15 OF C15

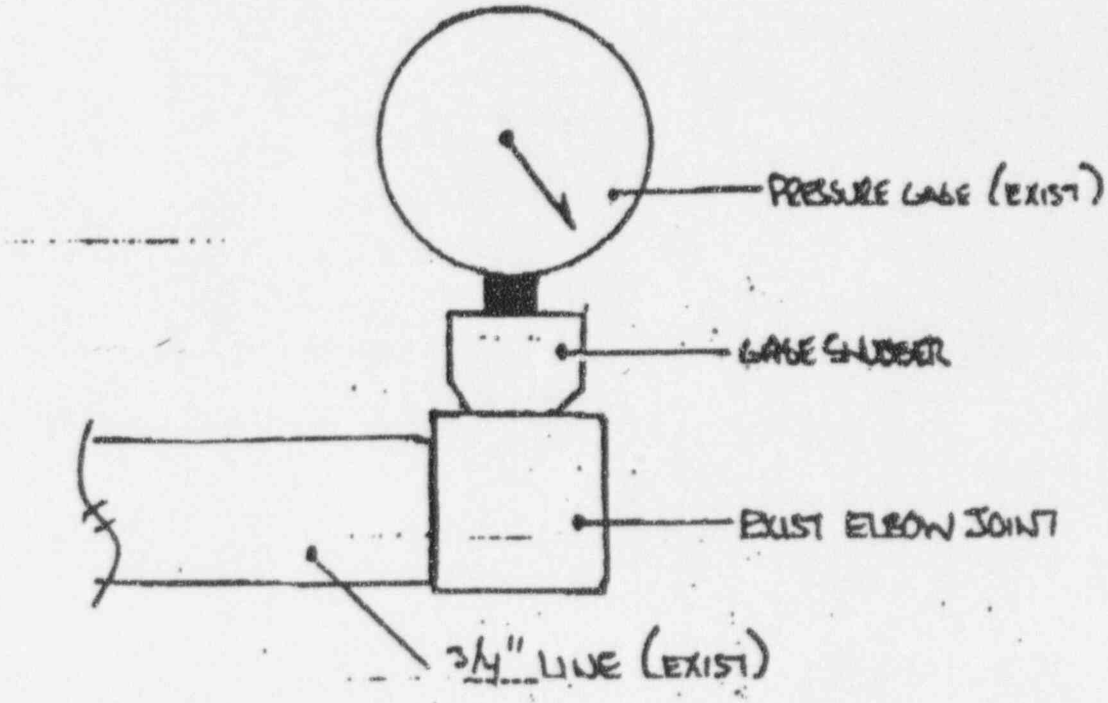

FIGURE NO. 15
PI-3053

37.7