ATTACHMENT A

LASALLE COUNTY STATION UNIT 1

TECHNICAL SPECIFICATION CHANGE REQUEST

NPF-4/82-11

Subject: Unit 1 RCIC Flow Test at Steam Pressure 150 ± 15 psig.

References (a): G. I. Zwarich letter to R. H. Holyoak dated August 6,1982.

(b): Sargent & Lundy Calculation No. RI-16, Rev. 1, approved August 6, 1982.

(c): Sargent & Lundy Calculation No. RI-17, Rev. 0, approved August 6, 1982.

BACKGROUND

The proposed Technical Specification change has been requested as a result of the inability of the RCIC turbine to meet the literal words of Technical Specification 4.7.3.c.2. During initial Startup Test STP-14, it was determined that RCIC cannot develop enough power at 150 psig to push 600 gpm though the Test Flowpath due to the large pressure drop in the test line.

Discussion

It is requested that Technical Specification 4.7.3.c.2 and Bases 3/4.7.3 be changed as indicated on the enclosed marked up pages. These changes provide that the turbine's capability be demonstrated by pumping through the test flow path at a different flow and pressure. The test flow path data will be demonstrated to be equivalent to or greater than that which is required to provide 600 gpm to the reactor vessel after correcting for line losses.

The current 1E51-F022 globe valve manufactured by Anderson Greenwood was initially assumed to have approximately 80 ft. of pressure drop in 1975. However, manufacturer's data supplied after 1975 indicated the valve has approximately 115 pounds of pressure drop. In addition to this difference in pressure drop, the RCIC test line is being repaired at this time due to underground piping problems. The new line will be longer and has additional 90 degree bends that will increase the present test line pressure drops. Therefore, it is necessary that the Technical Specification change be made.

8208170187 820811
PDR ADOCK 05000373
P PDR

Initial Startup Test data in Table 1 and plotted on Figure 1 indicates the current test line permits on 590 gpm flow at a discharge pressure of 430 pounds. Line losses, as calculated in References (a), (b), and (c) indicate that a pressure of 258 psig or greater at the Test Line will ensure a flow of 600 gpm into the reactor vessel at a reactor pressure of 165 psig. The equivalent pressure at the "T" at 590 gpm is 416 psig. Therefore, from Table 1 and Figure 1 it can be demonstrated that approximately 150 psig excess (416-258) is available for injection to the vessel. Actual Startup Test data will be taken to determine the vessel injection line loss and flow capacity.

Tables 2 and 3 are provided for additional information. Table 3 indicates RCIC has no problem in the test flow path at 940 psig reactor pressure.

Conclusion

Commonwealth Edison Company finds no unreviewed safety questions in this Technical Specification change.

4730N

no change

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.#

ACTION:

a. With a RCIC discharge line "keep filled" pressure alarm instrumentation channnel inoperable, perform Surveillance Requirement 4.7.3.a.1 at least once per 24 hours.

b. With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water,

   2. Performance of a CHANNEL FUNCTIONAL TEST of the discharge line "keep filled" pressure alarm instrumentation, and

   3. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

   4. Verifying that the pump flow controller is in the correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

#See Special Test Exception 3.10.7.

SURVEILLANCE REQUIREMENTS

c. At least once per 18 months by:

1. Performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve in the flow path actuates to its correct position, but may exclude actual injection of coolant into the reactor vessel.
2. Verifying that the system will develop a flow of greater than or equal to 600 gpm in the test flow path when steam is supplied to the turbine at a pressure of 150 + 15 psig.*
3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint to be ≥ 62 psig.

VERIFYING THAT THE SYSTEM IS CAPABLE OF PROVIDING A FLOW OF GREATER THAN OR EQUAL TO 600 gpm TO THE REACTOR VESSEL WHEN STEAM IS SUPPLIED TO THE TURBINE AT A PRESSURE OF 150 ± 15 PSIG.* USING THE TEST FLOW PATH

d. By demonstrating MCC-121y and the 250-volt battery** and charger** OPERABLE:

1. At least once per 7 days by verifying that:
   a) MCC-121y is energized, and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts.
   b) The electrolyte level of each pilot cell is above the plates,
   c) The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.200, and
   d) The overall battery voltage is greater than or equal to 250 volts.
2. At least once per 92 days by verifying that:
   a) The voltage of each connected battery is greater than or equal to 250 volts under float charge and has not decreased more than 12 volts from the value observed during the original test,
   b) The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.195 and has not decreased more than 0.05 from the value observed during the previous test, and
   c) The electrolyte level of each connected cell is above the plates.
3. At least once per 18 months by verifying that:
   a) The battery shows no visual indication of physical damage or abnormal deterioration, and
   b) Battery terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

*The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the tests.

**The Unit 2 RCIC system battery and charger may be in service to supply Unit 1 RCIC system provided that they are demonstrated OPERABLE per Unit 1 Technical Specifications. This footnote shall be deleted upon issuance of an Operating License for Unit 2.

3/4.7.1 CORE STANDBY COOLING SYSTEM - EQUIPMENT COOLING WATER SYSTEMS

The OPERABILITY of the core standby cooling system - equipment cooling water systems and the ultimate heat sink ensure that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM AND AUXILIARY ELECTRIC EQUIPMENT ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room and auxiliary electric equipment room emergency filtration system ensures that the rooms will remain habitable for operations personnel during and following all design basis accident conditions. The OPERABILITY of this system in conjunction with room design provisions is based on limiting the radiation exposure to personnel occupying the rooms to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50. Cumulative operation of the system with the heaters OPERABLE for 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig even though the LPCI mode of the the residual heat removal (RHR) system provides adequate core cooling up to 350 psig.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCICS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

Initial startup test program data may be used to determine equivalent turbine / pump capabilities between the test flow path and the vessel injection flow path.

no change

3/4.7.4 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emmitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.5 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, deluge and/or sprinklers, CO_2 systems, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

TABLE 1

LASALLE COUNTY STATION: UNIT 1

IE51-F022, full flow test line globe valve, was throttled to record the following pressures, speeds and flows. Flow controller set at 600 gpm. Reactor pressure 163 psig.

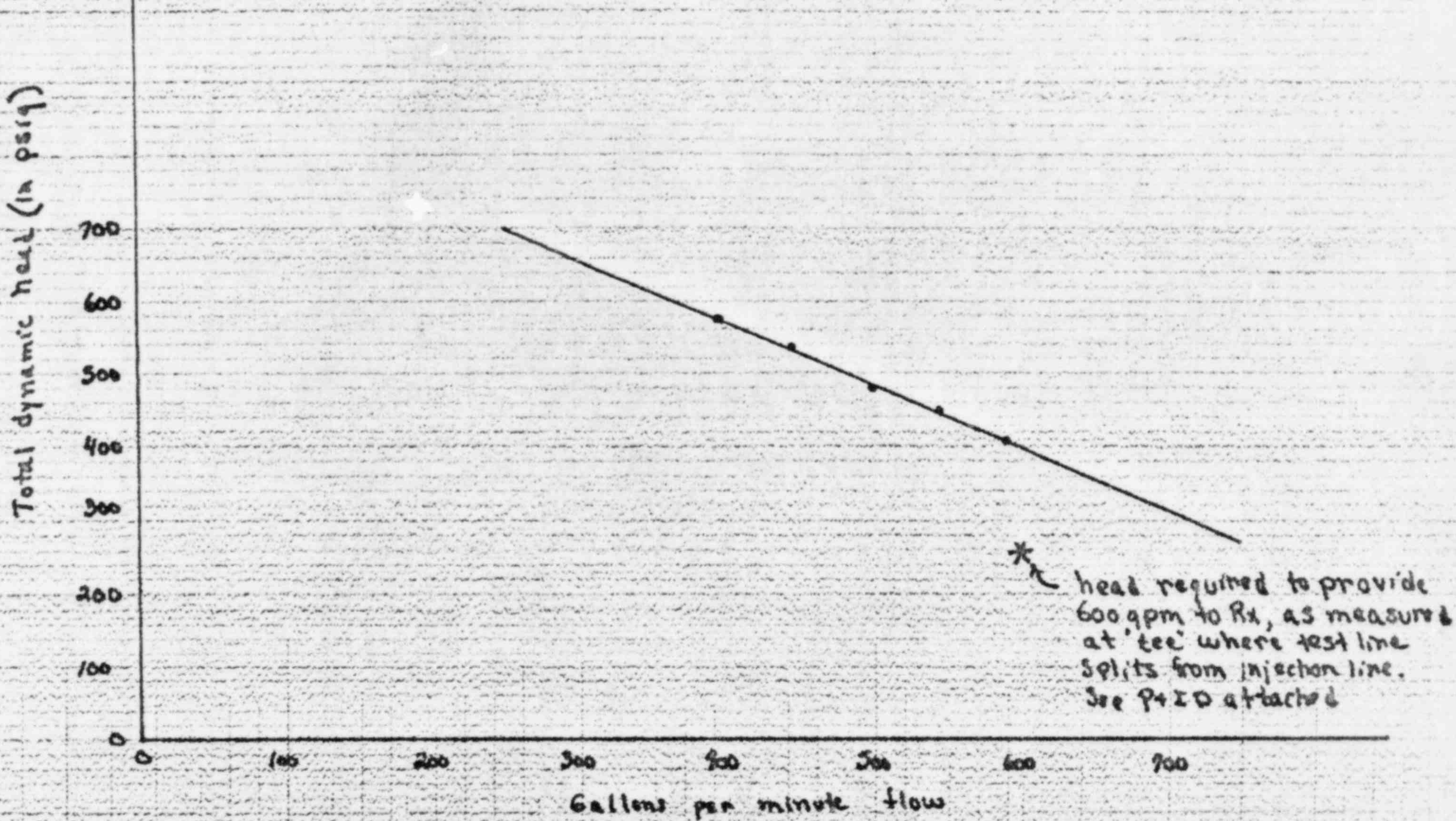
Flow gpm	Discharge Pressure psig	Suction Pressure psig	Speed rpm
590	430	21	2850
545	470	21	2900
500	500	21	3000
445	560	22	3100
395	610	22	3200

4730N

Fig 1

LaSalle County Station ; Unit One ; RCIC

Data plotted from Table 1. (attached)



8/10/82

TABLE 2

LASALLE COUNTY STATION: UNIT 1

IE51-F022, full flow test line globe valve, throttling valve, full open while flow was veried using "tape" setpoint of RCIC flow controller. Reactor pressure 160 psig.

Flow gpm	Discharge Pressure psig	Suction Pressure psig	Speed rpm
590	430	21	2900
570	400	21	2800
540	370	21	2650

TABLE 3

LASALLE COUNTY STATION: UNIT 1

RCIC flow controller set at 610 gpm; in automatic. Discharge pressure, suction pressure, and speed recorded while throttling IE51-F022, the test line globe valve. First reading taken with F022 fully open. Reactor pressure 940 psig.

Discharge Pressure	Suction Pressure	Speed	Flow
490 psig	22 psig	3100 rpm	610 gpm
575 psig	22 psig	3300 rpm	610 gpm
705 psig	21 psig	3550 rpm	610 gpm
760 psig	21 psig	3750 rpm	610 gpm
900 psig	21 psig	4050 rpm	610 gpm
1000 psig	21 psig	4200 rpm	610 gpm
1100 psig	21 psig	4450 rpm	610 gpm

SARGENT & LUNDY
ENGINEERS
FOUNDED 1891
55 EAST MONROE STREET
CHICAGO, ILLINOIS 60603
(312) 269-2000

August 6, 1982
Project No. 4266-00

Commonwealth Edison Company
LaSalle County Station - Unit 1

RCIC System Test

Mr. R. H. Holyoak
LaSalle County Station
Commonwealth Edison Company
R. R. 1
Marseilles, IL 61341

Dear Mr. Holyoak:

During initial testing of the RCIC system at low reactor pressure (≈150 psig) a flow of 600 gpm through the test line could not be reached. The maximum flow reached was 590 gpm. This is because of the high pressure drop in the new longer, heavier wall, underground return piping and also because of the high pressure drop through the fully open valve 1E51-F022.

In order to evaluate what was happening in the system and to reduce the inaccuracies of the calculations for the return line, a pressure gauge was installed at double block valve 1E51-F351 & 1E51-F352. The pressure reading at that point was 400 psig with a flow of 590 gpm going through the test line.

The test tap for valves 1E51-F351 & 1E51-F352 is at elevation 742'. Since there is no flow through this line, the only pressure difference between this point and the "T" is due to elevation differences.

$$\Delta Pel = \frac{(742-704)}{2.313}$$

$$\Delta Pel = 16 \text{ psi}$$

The equivalent pressure at the "T" or intersection of 1RI24A-6" and the test line 1RI30A-4" is then 416 psig when the flow is 590 gpm.

From calculation RI-16 Rev. 1 it can be seen that a pressure of 258 psig or greater at this "T" will ensure a flow of 600 gpm into the vessel at a reactor pressure of 165 psia.

Since there is very little flow loss between the pump discharge and the "T", the reduction in pressure due to the difference in flow between 600 gpm and 590 gpm is inconsequential in comparison to the existing pressure margin of 158 psi (416-258). For these reasons a flow of 600 gpm to the vessel is assured even though only 590 gpm was reached in the test line. This will also be verified by the actual injection into the vessel during the RCIC preoperational test.

To resolve the apparent discrepancy between the Technical Specification and the system design some possible actions are:

1. Revise the Technical Specifications to account for the present system design.

2. Revise the RCIC return line by replacing valve 1E51-F022 with another valve with a lower pressure drop.

3. Revise the RCIC return line by adding a bypass line with another valve in it in parallel with 1E51-F022.

4. Another possibility is to add another test return line to the suppression pool with appropriate valving and logic.

If you wish us to pursue any of these options, please contact us.

Yours very truly,

G. I. Zwarich
Mechanical Engineer

GIZ:cb
In duplicate
Copies:
B. R. Shelton
T. E. Watts
J. E. Renwick
R. L. Scott
J. M. Dameron
D. C. Haan
W. G. Schwartz
R. H. Pollock
E. R. Weaver
B. L. Pandit
File 3.2.4

SARGENT LUNDY ENGINEERS CHICAGO

Calcs. For	ORIFACE SIZING OF 1E51-D006 & 1E51-D307
X Safety-Related	Non-Safety-Related
Calc. No.	RI-16
Rev.	1
Date	8-4-82
Page	1 of 7
Client	CECo
Project	LaSalle
Proj. No.	4266-00
Equip. No.	1E51-D006
Prepared by	S.J. Evarian
Date	8-4-82
Reviewed by	[illegible]
Date	8-5-82
Approved by	David Okan
Date	8-6-82

This review was accomplished by a detailed review of the original calculation
[ ] comments as noted [X] no comments
Reviewer [illegible] Date 08-05-82

This revision to calc RI-16 is being done for three reasons. First, to account for the new RI test return line which is both longer and of heavier wall; second, to use the manufacturer data for 1E51-F022; and third, to use the test conditions actually encountered during the test. The methodology used will be the same as in Rev. 0.

Assumptions and Data

1. Test Temperature 75°F
2. elevation of intersection of 1RI 34A 6" and 1RI 30A-4" is 704'-0"
3. Valve 1E51-F356 is full open & 40-13
4. Valve 1E51-F022 has a Cv of 55 at 60F
5. VALVES 1E51-F065 + 1E51-F066 have ΔP of .33 psi at [illegible] (manufacturer data)

Form GQ 3.08.1 Rev. 2

6 RPV nozzle pressure required is 178 psia (GE Process Diagram)

7 nozzle elevation is 831.3'

Calculations

A. From "T" to RPV 6" NPS Sched 120 (ID 5.501)

1. Take Off & Equivalent Length

straight Pipe	317'	317
20 LR 90° ELL'S	=> 20(20)(5.501/12)	183
8 - LR 45° ELL'S	= (8)(12)(5.501/12) =	44
1E51-F013	1(13)(5.501/12) =	6
1E51-F065 + 1E51-F066	(SEE BELOW)	-
1 - 6"x6x6 "T" RUN	= 1(20)(5.501/12) =	9
1/2 - 6x4x6 "T" RUN	= 1/2(20)(3.624/5.501)(5.501/12) =	3
2 - 6"x1.5"x6" "T" RUN	= 2(20)(1.234/5.501)(5.501/12) =	4
		566'

2 CHECK VALVES 1E51-F065 & F066 ΔP = .66 psi

3 RPV PRESSURE REQUIRED AT NOZZLE = 178 psia

B. From "T" to CY Tank

1. 4" NPS SCHED 120 (ID 3.624)

straight pipe	5	5
5 - 90° LR ELL'S	5(20)(3.624/12)	30
1/2 6x4x6 T BRANCH	(1/2)(20)(3.624/5.501)(5.501/12)	9
SUB TOTAL		45

SARGENT LUNDY ENGINEERS CHICAGO

Calcs. For	
Safety-Related	Non-Safety-Related
Calc. No. R216	
Rev. 1	Date
Page 3	of 7
Client	
Project	
Proj. No. 4266-00	Equip. No.
Prepared by	Date
Reviewed by	Date
Approved by	Date

SUBTOTAL CARIED FWD.		45
2-4 ×× ×× BRANCH T	= 2×(60) $\left(\frac{3.624}{12}\right)$ =	36
1-1.5" STUB RUN	= 1(20) $\left(\frac{1.500}{3.624}\right)\left(\frac{3.624}{12}\right)$	3
2-GATE VALVES OPEN	= 2(13) $\left(\frac{3.624}{12}\right)$	8
TOTAL 4" SCHED 120		92'

1E51-F022, $C_V = 55$ at 60°F

$$\Delta P = \left(\frac{Q}{C_V}\right)^2 \frac{\rho}{62.4}$$

$$\Delta P = \left(\frac{600}{55}\right)^2 \frac{62.26}{62.4}$$

$$\Delta P = 119 \text{ psi}$$

4" NPS SCHED 40 INDOOR PIPING (ID 4.026)

STRAIGHT PIPE	32'	32
4-LR 90° ELLS	4(20) $\left(\frac{4.026}{12}\right)$ =	27
TOTAL		59'

4" NPS SCHED 40 OUTDOOR PIPE ID = (4.026)

STRAIGHT PIPE	1081'	1081
10-LR 90° ELL'S	10(20) $\left(\frac{4.026}{12}\right)$	67
5-LR 45° ELL'S	5(12) $\left(\frac{4.026}{12}\right)$	20
TOTAL		1168

EXIT LOSS K = 1.0

Find velocity + friction factor in each line size

$v = 0.408 \frac{Q}{d^2}$ $R_e = 123.9 \frac{d v \rho}{\mu}$; $\rho = 62.26$, $\mu = 0.9$

Form GQ 3.08.1 Rev. 2

1. 6" NPS Sched 120 ID 5.501

$v_6 = 0.408 \frac{600}{(5.501)^2}$

$v_6 = 8.09$ fps

$Re = 123.9 \frac{D v \rho}{\mu}$

$Re_6 = \frac{(123.9)(5.501)(8.09)(62.26)}{0.9}$

$Re_6 = 3.81 \times 10^5 \Rightarrow f_6 = 0.0168$

2. 4" NPS Sched 120 ID = 3.624

$v_{4_{120}} = \frac{(0.408)(600)}{(3.624)^2}$

$v_{4\,120} = 18.64$ fps

$Re_{4_{120}} = \frac{(123.9)(3.624)(18.64)(62.26)}{0.9}$

$Re_{4_{120}} = 5.79 \times 10^5 \Rightarrow f_{4_{120}} = 0.0176$

3. 4" NPS Sched 40 ID = 4.026

$v_{4.40} = \frac{(0.408)(600)}{(4.026)^2}$

$v_{4.40} = 15.1$ fps

$Re_{4.40} = \frac{(123.9)(4.026)(15.1)(62.26)}{(0.9)}$

$Re_{4.40} = 5.21 \times 10^5 \Rightarrow f_{4.40} = 0.0174$

SARGENT LUNDY ENGINEERS CHICAGO	Calcs. For		Calc. No. [illegible]
	Safety-Related	Non-Safety-Related	Rev. 1 Date
			Page 5 of 7

Client		Prepared by	Date
Project		Reviewed by	Date
Proj. No. 4266-00	Equip. No.	Approved by	Date

Find pressure drop/ft of pipe for each size

$$\Delta P = 2.16 \times 10^{-4} \left(\frac{FL\rho Q^2}{D^5} \right)$$

$$\Delta P_6 = \frac{(2.16 \times 10^{-4})(1.68 \times 10^{-2})(1)(62.26)(600^2)}{(5.501)^5}$$

$$\Delta P_6 = 1.6146 \times 10^{-2} \text{ psi/ft}$$

$$\Delta P_{4,120} = \frac{(2.16 \times 10^{-4})(1.76 \times 10^{-2})(1)(62.26)(600)^2}{(3.624)^5}$$

$$\Delta P_{4,120} = 1.363 \times 10^{-1} \text{ psi/ft}$$

$$\Delta P_{4,40} = \frac{(2.16 \times 10^{-4})(1.74 \times 10^{-2})(1)(62.26)(600)^2}{(4.026)^5}$$

$$\Delta P_{4,40} = 7.964 \times 10^{-2} \text{ psi/ft}$$

Find Pressure required at "T" from 6" line (to RPV)

$$P_{req} = (566)(1.6146 \times 10^{-2}) + (0.66) + 178 + \frac{(821.3 - 704)}{2.313}$$

P_{req} = 242.84 psia or 562' fa or 257.54 psig or 596 ft.

Find pressure loss from T to CY Tank

$$P = 92(.1363) + 119 + 54(.07964) + 1168(.07964) + \frac{1(15.1)^2}{(64.4)2.31} + 14.7 + \frac{7[illegible]}{2.3[illegible]}$$

P = 258.459 psig or 598 ft

SARGENT LUNDY ENGINEERS CHICAGO	Calcs. For		Calc. No. RI 16	
	Safety-Related	Non-Safety-Related	Rev. 1	Date
			Page 6	of 7

Client	Prepared by	Date
Project	Reviewed by	Date
Proj. No. 4266-00 Equip. No.	Approved by	Date

Conclusion; Oriface 1E51-D006 should be made full bore and is not really required.

1E51-D307

Assume Reactor at 1000 psig dome pressure

Required head at "T" is then

$$P_{req} : (566)(1.6146\times10^{-2}) + 0.66 + 1000 + (178-165) + \frac{831.3-704}{2.313}$$

P_{req} = 1077.84 psig or 2493 ft

Pressure drop from "T" to tank assuming full bore oriface for 1E51-D006 and valve 1E51-F356 closed

P = 258.457 psig − 55(.1363)

P = 251 psi or 580'

The Pressure drop required for 1E51-D307 is then

P. 2493 − 580 or 1913' or 827 psi

Since this is the theoretical maximum and assume no throttling at all for 1E51-F022 the oriface will be sized

Form GQ 3.08.1 Rev. 2

SARGENT LUNDY ENGINEERS CHICAGO	Calcs. For		Calc. No. RI 16
	Safety-Related	Non-Safety-Related	Rev. 1 Date
			Page 7 of 7

Client	Prepared by Date
Project	Reviewed by Date
Proj. No. 4266-00 Equip. No.	Approved by Date

for a loss of about 500 psi

$$Q = 236\, d_o^2\, C \sqrt{\frac{\Delta P}{\rho}}$$

$$d_o^2 = \frac{Q}{236\, C \sqrt{\frac{\Delta P}{\rho}}}$$

$$d_o^2 = \frac{600}{236(.6)\sqrt{\frac{500}{62.36}}}$$

$$d_o^2 = 1.5$$

$$d_o = 1.23''$$

These calculation are conservative in that we are not assuming enough pressure drop since the flow test return line has many areas of fitting to fitting installation which causes higher losses than the single fittings separated by ample straight piping. For this reason and because of the high pressure loss design of valve 1E51-F022 the size orifice 1E51-D307 will be increased to

$d_o = 1.50$ inches.

Form GQ.3.08.1 Rev. 2

SARGENT LUNDY
ENGINEERS
CHICAGO

Calcs. For	JUSTIFICATION OF LOW RCIC TEST FLOW
X Safety-Related	Non-Safety-Related

Calc. No.	RI-17
Rev. 0	Date [illegible]
Page 1	of 3

Client	COMMONWEALTH EDISON	
Project	LA SALLE	
Proj. No.	4266-00	Equip. No.

Prepared by [illegible]	Date [illegible]
Reviewed by [illegible]	Date [illegible]
Approved by [signature]	Date 86-82

This review was accomplished by a detailed review of the original calculation
☐ comments as noted ☒ no comments
Reviewer [illegible] Date [illegible]

Purpose

During initial testing of the RCIC system at low reactor pressure (≈ 150 psig) a flow of 600 gpm through the test line could not be reached. The maximum flow reached was 590 gpm. As seen in calculation RI 16 rev. 1 this is because of the high pressure drops in the new larger, heavier walled underground return piping and also because of the high pressure drop through the fully open valve 1E51-F022.

In order to evaluate what was happening in the system and to reduce the inaccuracies of the calculations for the return line, a pressure guage was installed at double block valve 1E51-F351 & F352. The pressure

Client		Prepared by	Date
Project	—	Reviewed by	Date
Proj. No. 4266-00	Equip. No.	Approved by	Date

reading at that point was 400 psig with a flow of 590 gpm going through the test line.

Calculation

The test tap for valves 1E51-F351 & F352 is at elevation 742'. Since there is no flow through this line, the only pressure difference between this point and the "T" is due to elevation difference

$$\Delta P_{el.} = \frac{(742 - 704)}{2.313}$$

$$\Delta P_{el.} = 16.4 \text{ psi}$$

The equivalent pressure at the "T" or intersection of 1FI 24A-6" and the test line, 1RI 50A-4" is then 416.4 psi when the flow is 590 gpm.

From calculation RI-16 rev. 1 it can be seen that a pressure of 250 psig or greater at this "T"

SARGENT LUNDY ENGINEERS CHICAGO

Calcs. For	
Safety-Related	Non-Safety-Related
Calc. No. RD-17	
Rev. 0	Date
Page 3 of 5	
Client	
Project	
Proj. No. 4266-00	Equip. No.
Prepared by	Date
Reviewed by	Date
Approved by	Date

will ensure a flow of 600 gpm into the vessel at a reactor pressure of 165 psia.

Since there is very little flow loss between the pump discharge and the "T", the reduction in pressure due to the difference in flow between 600 gpm and 590 gpm is inconsequential in comparison to the existing pressure margin 416-258 or 158 psi.

For these reasons a flow of 600 gpm to the vessel is assumed even though only 590 gpm was reached in the test line.

Form GQ-3.08.1 Rev. 2

ATTACHMENT B

Status of Tech Spec Change Requests

#	Topic	Submitted	NRC Action
NPF-11/82-7	SRM Countrate (Required prior to source decay below 3 cps).	6/14/82	Issued Am. 2 7/09/82
NPF-11/82-8	Revise RCIC suction delta p alarm setpoint (Required prior to pressurization).	7/02/82	Issued Am. 2 7/09/82
NPF-11/82-9	Add commitment to complete torque checks on bolting on S/R valves outside containment (NRC Requested to be submitted ASAP).	7/14/82	Issued Am. 3 7/15/82
NPF-11/82-10	Add ECCS delta p specification to reflect modification (License Condition 2.C.C)	----	
NPF-11/82-11	Revise RCIC surveillance to account for flow differences between test flow and normal flow paths (Required prior to restart after next shutdown).	8/11/82	

4730N