Peter Zarakas
Vice President

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4 Irving Place, New York, NY 10003
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August 11, 1980

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
ATTN: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Varga:

As required by your June 13, 1980 letter which forwarded the NRC Regulatory Staff's interim safety evaluation of the Indian Point Unit No. 2 auxiliary feedwater system, the information regarding items C.1, C.3 and D of that evaluation is provided in Attachments A, B and C, respectively, to this letter. In addition, further information requested by members of the Regulatory Staff regarding auxiliary feedwater automatic initiation, indication, and channelization is provided in Attachment D to this letter.

Should you or your staff have any further questions, please contact us.

Very truly yours,

Peter Zarakas

Peter Zarakas
Vice President

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ATTACHMENT A

Redundant position indication system for the two series manual valves between the condensate storage tank and the auxiliary feedwater suction manifold (Ref: Section C.1, Recommendation GL-2, of the June 13, 1980 Interim Safety Evaluation):

The redundant position indication system, described in Consolidated Edison's April 14, 1980 submittal and addressed in item C.1 of the June 13, 1980 NRC Interim Safety Evaluation, has been installed at Indian Point Unit No. 2. In addition, as required by the subject recommendation, plant procedures have been revised to require that the auxiliary feedwater pumps be placed in the manual start mode in the event of an alarm indicating that either one or both of the two series suction line valves is not in the open position. These procedures further require that the auxiliary feedwater system be maintained in the manual start mode until the valve is returned to the open position or is verified to be open.

Finally, proposed Technical Specifications for the series valves in the single line from the Condensate Storage Tank have been submitted under separate cover letter dated August 11, 1980.

ATTACHMENT B

Results of the evaluation of the capability of the present design to withstand internally generated missiles (Ref: Section C.3 of the June 13, 1980 Interim Safety Evaluation):

Plant Specific Recommendation (Long Term) No. 1 (part b) of the Commission's November 7, 1979 letter, regarding auxiliary feedwater systems, required that Consolidated Edison evaluate the capability of the Indian Point Unit No. 2 Auxiliary Feedwater System (AFWS) design to withstand internally generated missiles.

In our December 19, 1979 letter, we committed to evaluate the capability of the AFW system design to withstand internally generated missiles and to provide the results of this evaluation. The results of this evaluation are also required by the Staff to complete their evaluation of the auxiliary feedwater system reliability for Indian Point Unit No. 2 as stated in the interim safety evaluation of the Unit 2 AFWS dated June 13, 1980.

In compliance with your recommendation and our commitment, Consolidated Edison and the Power Authority jointly contracted EDS Nuclear, Inc. to evaluate the capability of the auxiliary feedwater system to withstand internally generated missiles.

The purpose of the study was to:

1. Determine if a missile could be generated from the steam driven AFW pump, drive line components or turbine, and

2. If a missile could be generated, to determine the potential effects of the missile on vital AFWS components within the AFW pump room.

Preliminary investigation indicated that the auxiliary feedwater pump turbine could be the only potential source of a missile and, therefore, a detailed analysis of the turbine was undertaken. The rated speed of the turbine is 3,927 RPM, with trip speed, following a governor failure, of 4,500 RPM. For conservatism, the missile analysis assumed the worst case condition consisting of simultaneous loss of load, governor failure and trip valve failure. Consequently, the turbine disc was assumed to reach destructive overspeed of 14,047 RPM.

The turbine disc has only one keyway and, therefore, it was considered necessary to perform a parametric study of various disc sizes ranging from 10° to 170° segments. The results of the study showed that, at the burst speed of 14,047 RPM, a 30° segment would cause a casing failure due to stage 1 perforation. Other size missiles would also cause casing failure, with the worst stage 2 failure being a 134° segment.

The results of this analysis showed that a turbine missile generated at destructive overspeed of 14,047 RPM could escape the auxiliary feed pump turbine casing.

To determine the effect of the missile, a detailed study of the AFW pump room was conducted to identify critical components which could be potential targets. The method used involved superimposing a 50° angle of revolution (Reg. Guide 1.15) about the turbine midplane on applicable piping, instrument and electrical layout drawings. This information was verified by actual plant walk-throughs. After a target was identified, the failure of the equipment was assumed and the impact on impairment of the AFWS was analyzed.

The results of the study showed that there was a sufficient amount of vital equipment within the target area such that a missile could disable the AFWS.

Based on the above, Consolidated Edison is presently considering implementing one of the following corrective measures:

1. The installation of a protective missile shield around the pump turbine for protection against missiles generated at destructive overspeed. (A preliminary investigation shows a shield thickness of approximately 3 inches of steel would be required).

2. Upgrading of the auxiliary feed pump turbine overspeed protection system by equipment modifications and/or inspection and testing so that credit may be taken for its functionability.

3. Rerouting of all safety-related equipment which could be potential targets of a pump turbine missile.

4. Investigating the feasibility of using alternate means to provide feedwater to the steam generators independent of the AFWS.

Consolidated Edison will notify the Commission as to which alternative will be implemented as soon as the evaluation of the above alternatives is completed. At that time, a full description of the evaluation and supporting documentation will be provided for review. It is our intention to effect any necessary modifications compatible with a planned plant outage but no later than our next scheduled refueling/maintenance outage.

ATTACHMENT C

Basis for Auxiliary Feedwater
System Flow Requirements
(Ref: Enclosure 2 of NRC
November 7, 1979 letter)

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
August, 1980

ATTACHMENT C

Question 1:

a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1. Loss of Main Feed (LMFW)
2. LMFW w/loss of offsite AC power
3. LMFW w/loss of onsite and offsite AC power
4. Plant cooldown
5. Turbine trip with and without bypass
6. Main steam isolation valve closure
7. Main feed line break
8. Main steam line break
9. Small break LOCA
10. Other transient or accident conditions not listed above.

b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

1. Maximum RCS pressure (PORV or safety valve actuation)
2. Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
3. RCS cooling rate limit to avoid excessive coolant shrinkage
4. Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

Response to 1.a:

The Auxiliary Feedwater System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. The Auxiliary Feedwater System is directly relied upon to prevent core damage and system overpressurization in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following plant transients.

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the generated steam either to the condensers through the steam dump valves or to the atmosphere through the steam generator safety valves or the power-operated relief valves. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and continuation of the decay heat removal process. The water level is maintained under these circumstances by the Auxiliary Feedwater System which delivers an emergency water supply to the steam generators. The Auxiliary Feedwater System must be capable of functioning for extended periods, allowing time either to restore normal feedwater flow or to proceed with an orderly cooldown of the plant to the reactor coolant temperature where the Residual Heat Removal System can assume the burden of decay heat removal. The Auxiliary Feedwater System flow and the emergency water supply capacity must be sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown.

DESIGN CONDITIONS

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for Indian Point Unit No. 2:

- Loss of Main Feedwater Transient
  - Loss of main feedwater with offsite power available
  - Station blackout (i.e., loss of main feedwater without offsite power available)

- Rupture of a Main Steam Line

- Loss of all AC Power

- Loss of Coolant Accident (LOCA)

- Cooldown

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

- Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system

- Loss of offsite power or blackout with the consequential shutdown of the system pumps, auxiliaries, and controls

Loss of main feedwater transients are characterized by a rapid reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following reactor trip from high power, the power quickly falls to decay heat levels. The water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam either through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. The reactor coolant temperature increases as the residual heat in excess of that dissipated through the steam generators is absorbed. With increased temperature, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater, further expansion will result in water being discharged through the pressurizer safety and relief valves. Hence, the timely introduction of sufficient auxiliary feedwater is necessary to arrest the decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and eventually to establish stable hot shutdown conditions. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected.

The blackout transient differs from a simple loss of main feedwater in that emergency power sources must be relied upon to operate vital equipment. The loss of power to the electric driven condenser circulating water pumps results in an undervoltage condition on the associated 6.9 Kv Bus which trips the affected circulating water pumps with loss of operability of the condenser steam dump valves. Over time, condenser vacuum is lost; hence, steam formed by decay heat is relieved through the steam generator safety valves or the power-operated relief valves. The calculated transient is similar for both the loss of main feedwater and the blackout, except that reactor coolant pump heat input is not a consideration in the blackout transient following loss of power to the reactor coolant pump buses.

The station blackout transient serves as the basis for the minimum flow (400 gpm) required for Indian Point Unit No. 2. The system is designed so that a minimum of 600 gpm is provided against the steam generator safety valve set pressure (with 3% accumulation) to prevent water relief from the pressurizer. This is accomplished in the Indian Point Unit No. 2, even considering the effect of the throttling of the individual flow control valves and assuming a coincident single failure.

Rupture of a Main Steam Line

Because the rupture of a main steam line may result in the complete blowdown of one steam generator, a partial loss of the plant heat sink is a concern. The main steamline rupture accident conditions are characterized initially by plant cooldown, and hence, auxiliary feedwater flow is not needed during the early stage of the transient to remove decay heat from the Reactor Coolant System. Emergency procedures and the necessary equipment exist to allow termination of flow to the faulted loop and to provide flow to the intact steam generators during the controlled cooldown following the steamline break accident.

Loss of All AC Power

The loss of all AC power is postulated as resulting from accident conditions wherein not only onsite and offsite AC power is lost but also AC emergency power is lost as an assumed common mode failure. Battery power for operation of protection circuits is assumed available. The impact on the Auxiliary Feedwater System is the necessity for providing both an auxiliary feedwater pump power and control source which are not dependent on AC power and which are capable of maintaining the plant at hot shutdown until AC power is restored. The steam-driven auxiliary feedwater pump No. 22 provides such a capability (Also see Table 1B-1). In addition, as part of the Unit 2 alternate shutdown system planned for installation during the 1981 refueling outage, an independent power feed will be provided to one motor-driven auxiliary feedwater pump independent of all existing Unit 2 offsite and onsite AC power feeds. This independent power feed will normally be from a Unit 1 offsite AC power feed and is provided with emergency backup power from gas turbines.

Loss-of-Coolant Accident (LOCA)

The loss of coolant accidents do not impose on the auxiliary feedwater system any flow requirements in addition to those required by the other accidents addressed in this response. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the auxiliary feedwater system in this transient.

Small LOCA's are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCAs is basically the same as the system's function during hot shutdown or following a spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The auxiliary feedwater system is utilized in a system cooldown and depressurization following a small LOCA while bringing the reactor to a cold shutdown condition.

Cooldown

The cooldown function performed by the Auxiliary Feedwater System is a partial one since the reactor coolant system is reduced from normal zero load temperatures to a hot leg temperature of approximately 350°F. The latter is the maximum temperature recommended for placing the Residual Heat Removal System (RHRS) into service. The RHR system completes the cooldown to cold shutdown conditions.

Cooldown may be required following expected transients, following an accident such as a main feedline break, or it may be a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level (maximum decay heat), the AFWS is capable of delivering sufficient AFW to remove decay heat and reactor coolant pump (RCP) heat following reactor trip while maintaining the steam generator (SG) water level. Following transients or accidents, the recommended cooldown rate is consistent with expected needs and at the same time does not impose additional requirements on the capacities of the auxiliary feedwater pumps, considering a single failure. In any event, the process consists of being able to dissipate plant sensible heat in addition to the decay heat produced by the reactor core.

Response to 1.b:

Table 1B-1 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a, above. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2.

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability for heatup accidents following reactor trip to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are bounded by the analysis of the rupture of a main steam pipe transient. The maximum flow requirements determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.

TABLE 1B-1

Criteria for Auxiliary Feedwater System Design Basis Conditions

Condition or Transient	Classification*	Criteria*	Additional Design Criteria
Loss of Main Feedwater	Condition II	Peak RCS pressure not to exceed design pressure. No consequential fuel failures	
Station Blackout	Condition II	(same as LMFW)	Pressurizer does not fill with 400 gpm delivered to 2 SGs.
Loss of all A/C Power	N/A	Note 1	
Loss of Coolant	Condition III	10 CFR 100 dose limits 10 CFR 50 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 50 PCT limits	
Cooldown	N/A		$100^{o}F$/hr $547^{o}F$ to $350^{o}F$

*Ref: ANSI N18.2 (This information provided for those transients performed in the FSAR).

Note 1 Although this transient establishes the basis for AFW pump powered by a diverse power source, this is not evaluated relative to typical criteria since multiple failures must be assumed to postulate this transient.

Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a above including:

a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

b. Time delay from initiating event to reactor trip.

c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

d. Minimum steam generator water level when initiating event occurs.

e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out of 4?

h. RC flow condition -- continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removel due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

Response to 2 :

Analyses have been performed for the Loss of Main Feedwater and the loss of offsite AC power to the Station, the transients which define the AFWS performance requirements. These analyses have been provided in the FSAR and approved by NRC.

In addition to the above analyses, calculations have been performed specifically for Indian Point Unit No. 2 to determine the plant cooldown flow (storage capacity) requirements. The LOCA analysis, as discussed in response 1.b, incorporates the system flows requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response.

Loss of Main Feedwater (Blackout)

A loss of feedwater, assuming a loss of power to the reactor coolant pumps, was performed in FSAR Section 14.1.9 for the purpose of showing that for a station blackout transient, with a minimum of 400 gpm delivered to two steam generators does not result in filling the pressurizer. Furthermore, the peak RCS pressure remains below the criterion for Condition II transients and no fuel failures occur (refer to Table 1B-1). Table 2-1 summarizes the assumptions used in this analysis. The transient analysis begins at the time of reactor trip. This can be done because the trip occurs on a steam generator level signal, hence the core power, temperatures and steam generator level at time of reactor trip do not depend on the event sequence prior to trip. Although the time from the loss of feedwater until the reactor trip occurs cannot be determined from this analysis, this delay is expected to be 20-30 seconds. The analysis assumes that the plant is initially operating at 102% (calorimetric error) of the Engineered Safeguards design (ESD) rating shown on the table, a very conservative assumption in defining decay heat and stored energy in the RCS. The reactor is assumed to be tripped on steam/feed mismatch coincident with low steam generator level, allowing for level uncertainty. The FSAR shows that there is a considerable margin with respect to filling the pressurizer. A loss of normal feedwater transient with the assumption that the two smallest auxiliary feedwater pumps operate with the reactor coolant pumps running results in even more margin.

This analysis establishes the minimum flow requirements and also establishes train association of equipment so that this analysis remains valid assuming the most limiting single failure.

Plant Cooldown

Minimum flow requirements from the previously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tankage size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in response 1.a, the auxiliary feedwater system partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table 2-1 shows the assumptions used to determine the cooldown heat capacity of the auxiliary feedwater system.

The cooldown is assumed to commence at the maximum rated power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFWS. See Table 2-2 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid source which are normally aligned.

TABLE 2-1

Summary of Assumptions Used in AFWS Design Verification Analyses

	Transient	Loss of Feedwater (station blackout)	Cooldown
a.	Max reactor power	102% of ESD rating (102% of 3216.5 MWt)	2813 MWt
b.	Time delay from event to Rx trip	2 sec	2 sec
c.	AFWS actuation signal/time delay for AFWS flow	lo-lo SG level 1 minute	NA
d.	SG water level at time of reactor trip	(Lo SG level + steam-feed mismatch) 30% NR span	NA
e.	Initial SG inventory	45,600 lbm/SG (at initiation of event)	56,035 lbm/SG @ 514.8°F
	Rate of change before & after AFWS actuation	See FSAR Section 14.1.9	N/A
	decay heat	ANS + 20%	
f.	AFW pump minimum delivery pressure	1133 psia	1112 psia
g.	Minimum # of SGs which must receive AFW flow	2 of 4	N/A
h.	RC pump status	Tripped @ reactor trip	Tripped
i.	Maximum AFW temperature	100°F	100°F
j.	Operator action	none	N/A
k.	MFW purge volume/temp.	476 ft^3/410.6°F	100 ft^3/SG @415.6°F
l.	Normal blowdown	none assumed	none assumed
m.	Sensible heat	Table 2-2	Table 2-2
n.	Time at standby/time to cooldown to RHR	2 hr/4 hr	2 hr/4 hr
o.	AFW flow rate	400 GPM - constant (min. requirement)	variable

TABLE 2-2

Summary of Sensible Heat Sources

Primary Water Sources (initially at rated power temperature and inventory)
- RCS fluid
- Pressurizer fluid (liquid and vapor)

Primary Metal Sources (initially at rated power temperature)
- Reactor coolant piping, pumps and reactor vessel
- Pressurizer
- Steam generator tube metal and tube sheet
- Steam generator metal below tube sheet
- Reactor vessel internals

Secondary Water Sources (initially at rated power temperature and inventory)
- Steam generator fluid (liquid and vapor)
- Main feedwater purge fluid between steam generator and AFWS piping.

Secondary Metal Sources (initially at rated power temperature)
- All steam generator metal above tube sheet, excluding tubes.

Question 3:

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Response to 3:

Figure 3-1 schematically shows the major features and components of the Auxiliary Feedwater System for Indian Point Unit No. 2. Flow rates for all of the design transients described in Response 2 have been met by the system for the worst single failure. The flows for those single failures considered are tabulated for the various transients in Table 3-1, including the following:

A. Turbine Driven Pump Failure
B. Motor Driven Pump Failure
C. AFWS check valve failure (failure to close on reverse flow).

There is approximately 15% margin for the turbine-driven auxiliary feed pump which allows for a continuous recirculation flow and bearing cooling flow, as well as any seal leakage and pump wear. The motor-driven auxiliary feed pumps do not utilize pump discharge flow for bearing cooling. Also, the recirculation flow path is automatically isolated when pump flow increases above 55 gpm (maximum recirculation flow), thus providing flow only to the steam generators at design conditions. Pump wear and seal leakage are considered to be negligible.

Table 3-1 includes the effect of throttling of the auxiliary feedwater system. As can be seen in all cases the minimum flow requirement of 400 gpm is clearly exceeded.

TABLE 3-1

Auxiliary Feedwater Flow (1) to Steam Generators
Following an Accident/Transient
With Selected Single Failure - GPM

	Single Failure		
Accident/Transient	TD Pump Failure	MD Pump Failure	CV(2) Failure
	A	B	C
1. Loss of Main FW	600	900	1200
2. Blackout	600	900	1200
3. Cooldown	600	900	1200

Notes:

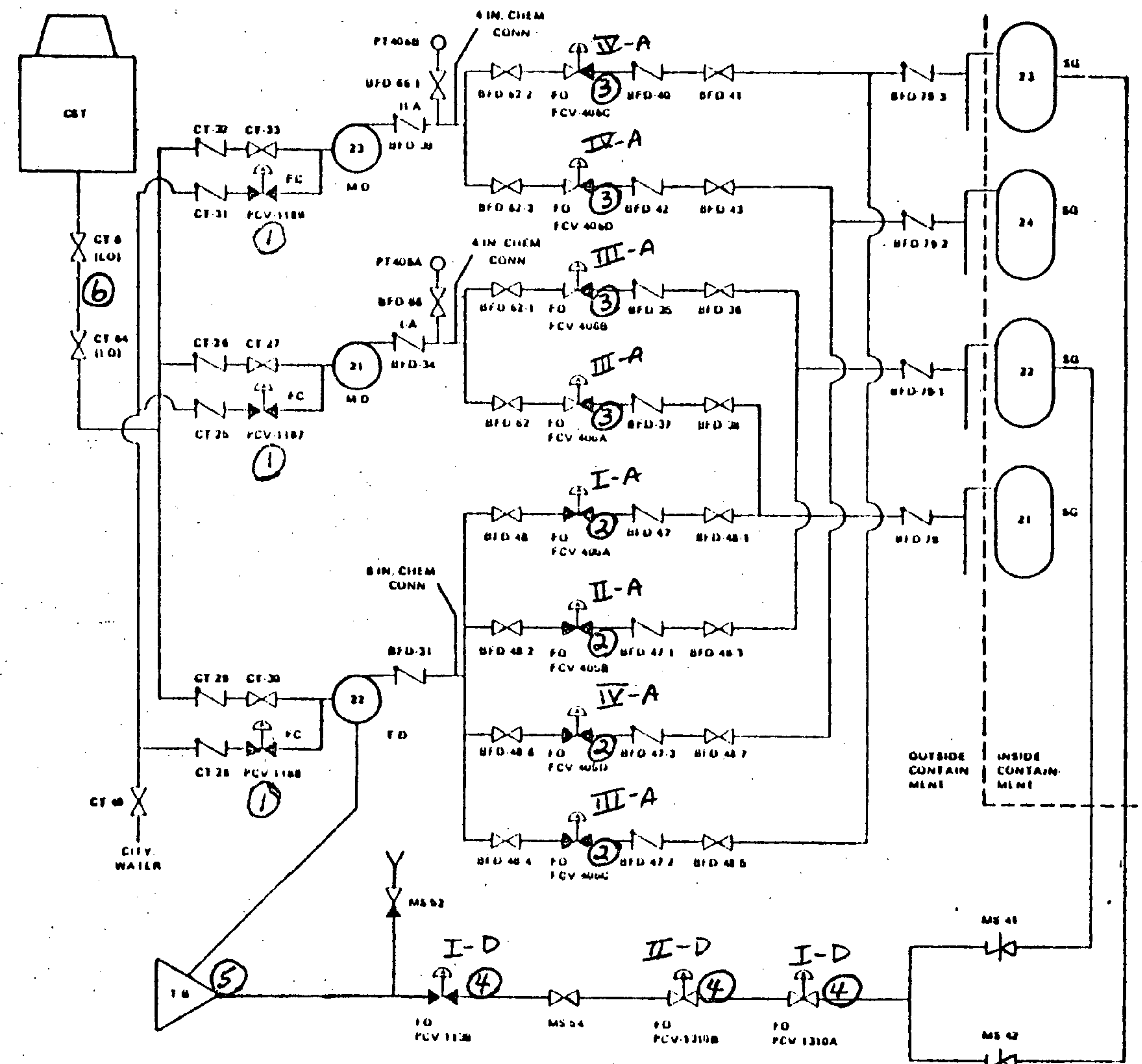
(1) Items 1 thru 3 are minimum expected flows to intact loops.

(2) Including only those CVs in the AFWS. "Failure" is interpreted as failure to close on reverse flow; failure of the CV to open to permit flow in the normal direction is not considered.

AUXILIARY FEEDWATER SYSTEM - INDIAN POINT 2

Legend:

MD - Motor Driven
TD - Turbine Driven
- Normally Open
- Normally Closed
[M] - Motor Operated
(A) - Air Operated
- Partially Open
- Stop Check
SG - Steam Generator
I,II,III,IV - Power Divisions
A - Alternating Current
D - Direct Current
TB - Turbine
FO - Fail Open
FC - Fail Close
[LO] - Locked Open



1. PCV-1187, 1188, 1189 - Powered from Non-IE Bus. Fail Closed on loss of solenoid power or air.

2. FCV-405A, B, C, D - Powered from Class-IE Buses. Fail open on loss of power or air. Normally Closed.

3. FCV-406A, B, C, D - Powered from Class-IE Buses. Partially open. Fail fully open on loss of power or air.

4. All steam supply valves are controlled by DC solenoids powered from Class IE batteries and will fail open on loss of DC power or air. PCV-1139 (steam supply reg. valve) normally closed, and PCV-1310 A and B (high temperature shutoff valves) normally open.

5. HCV-1118 - Pneumatic Hand Control Valve - Permits operator speed control from CCR. It cannot isolate steam flow.

6. CT-6 and CT-64 locked open with redundant position indication for each valve.

Figure 3-1

ATTACHMENT D

Additional information regarding auxiliary feedwater automatic initiation, indication and channelization:

I. AFWS Initiation-Control and Protection Interaction:

Signals from auxiliary feedwater initiation channels to control systems are transmitted through isolation devices. No failure in the control system can prevent proper actuation of the auxiliary feedwater initiation protection system.

II. AFWS Testing:

As committed in Consolidated Edison's October 17, 1979 and December 14, 1979 submittals to NRC, a periodic test program for all automatic AFWS initiation signals has been developed and surveillance testing will be performed at each refueling interval.

III. AFWS Bypass Indication:

In addition to the positive indication provided in the control room for the three auxiliary feedwater pumps and associated regulating valves, steam inlet valves and suction valves, separate alarms are provided to annunciate the following:

(1) Pump control switches in "pullout" ("Safeguards Equipment Locked Open"),

(2) "Remote/local" switches in "local" ("Control Transferred to Local"), and

(3) A closed AFWS series suction valve from the condensate storage tank.

All of the above indications supplement existing administrative controls which include the use of written Maintenance Work Requests and Shift Turnover Checklists to assure that the operator is aware of auxiliary feedwater system status at all times.

IV. AFWS Flow and Steam Generator Level Indication:

As indicated in Consolidated Edison's December 31, 1979 submittal, the single failure criterion for safety grade auxiliary feedwater flow indication (Lessons Learned Item 2.1.7b) is being satisfied through the use of diversity.

We have installed new safety grade auxiliary feedwater flow transmitters which have satisfied testing requirements of IEEE-323-1971 and IEEE-344-1971. All four flow channels are powered from a separate and independent Class IE 120VAC Instrumentation and Control Bus (Instrument Buses 21, 22, 23 and 24 power the auxiliary feedwater flow channels for steam generators 21, 22, 23 and 24, respectively). These power supplies were specifically selected to maximize independence from the multiple level channels for each steam generator.

Also as indicated in the December 31, 1979 submittal, the existing safety grade steam generator level indication system can provide the diverse means for determining adequate auxiliary feedwater flow to a steam generator in the event that the auxiliary feedwater flow indication channel for that steam generator has failed. Each steam generator has three redundant narrow range level indication channels and one wide range level indication channel, all powered from the previously mentioned four Class-IE 120VAC Instrument Buses. In particular, the three narrow range level channels for steam generators 21 and 24 are powered from Instrument Buses 22,23 and 24, and the three narrow range level channels for steam generators 22 and 23 are powered from Instrument Buses 21, 23 and 24. The wide range level channels for each of the four steam generators are powered from Instrument Bus 23. The narrow range level channels themselves provide redundant indication of adequate water level in each steam generator. This level indication provides a diverse check on the delivery of adequate auxiliary feedwater flow to each steam generator.

As indicated in our recent July 30, 1980 submittal, the discharge regulating valves for each motor-driven auxiliary feedwater pump are "unitized" on the instrument bus within their associated pump's power train (i.e., Instrument Bus 23 for motor-driven auxiliary feed pump 21 and Instrument Bus 24 for motor-driven auxiliary feed pump 23). The discharge regulating valves for the steam turbine-driven auxiliary feed pump 22 are powered from all four instrument buses so that a single failure will only affect regulating capability from the steam-driven pump to one steam generator. Furthermore, the power supplies for these four regulating valves have been "unitized" with the auxiliary feedwater flow channelization for each steam generator.

The net effect is an overall design which maximizes the availability of both indication and regulating capability of the auxiliary feedwater system. The minimum requirement for Indian Point Unit No. 2 is auxiliary feedwater flow to two of four steam generators. Therefore, even considering a faulted steam generator loop and the additional single failure of one of the four redundant, independent auxiliary feedwater flow channels, the minimum requirement of flow to two intact steam generators can still be verified solely with the new flow indication system.