ATTACHMENT B

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
April,1981

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LIST OF TABLES

ΔT trip setpoint for three loop operation has been set in accordance with specification 2.3.1.B-4.

d. Reactor operation with one of the four loops out of service will be permitted for up to 24 hours. If the fourth loop can not be returned to service within 24 hours, the reactor will be put in a hot shutdown condition using normal procedures.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor is critical and the average coolant temperature is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable, or an opening greater than or equal to the size of one code safety valve flange shall be provided to allow for pressure relief, whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with $\pm 1\%$ allowance for error.

4. Power Operated Relief Valves (PORVs)/Block Valves

a. Whenever the reactor coolant system is above 350°F, the PORVs and their associated block valves shall be operable with the block valves either open or closed.

b. If a PORV becomes inoperable when above 350°F, its associated block valve shall be maintained in the closed position.

c. If a PORV block valve becomes inoperable when above 350°F, the block valve shall be closed and deenergized.

d. If the requirements of specifications 3.1.A.4.a, 3.1.A.4.b or 3.1.A.4.c above cannot be satisfied, compliance shall be established within one (1) hour, or the reactor shall be placed in the hot shutdown condition within the next six (6) hours and subsequently cooled below 350°F.

5. Pressurizer Heaters

a. Whenever the reactor coolant system is above 350°F, the pressurizer shall be operable with at least 150kw of pressurizer heaters.

b. If the requirements of specification 3.1.A.5.a cannot be met, restore the required pressurizer heater capacity to operable status within 72 hours or place the reactor in hot shutdown within the next 6 hours and subsequently cool below 350°F.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only (1); hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

Three loop operation is allowed over a 24 hour period to permit corrective action to return the fourth loop to service and limit the number of unnecessary shutdown cycles. During these periods of three loop operation, the reactor coolant system parameters will be maintained within the limits described for three loop operation in Section 2.1 and 3.1 of the Technical Specifications.

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperatures and pressure.(2)

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for overpressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load(3) without a direct trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove decay heat after a reactor shutdown.

All pressurizer heaters are supplied electrical power from an emergency bus. The requirement that 150kw of pressurizer heaters and their associated controls be operable when the reactor coolant system is above 350°F provides assurance that these heaters will be available and can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) can operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to provide a relief path when desirable and to ensure the ability to seal off possible RCS leakage paths. Both the PORVs and the PORV block valves are subject to periodic valve testing for operability in accordance with the ASME Code Section XI as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program.

Reference

1) FSAR Section 14.1.6
2) FSAR Section 9.3.1
3) FSAR Section 14.1.10

3.5.5 The cover plate on the rear of the safeguards panel, in the control room, shall not be removed without authorization from the Watch Supervisor.

3.5.6 When the reactor coolant system is above 350°F, the instrumentation requirements as stated in Table 3-5 shall be met.

Basis

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features.[(1)(4)]

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safety Features have been designed to sense the effects of the Loss of Coolant accident by detecting low pressurizer pressure and generate signals actuating the SIS active phase.

The SIS active phase is also actuated by a high containment pressure signal (Hi-Level) brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure signal actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between any two steam generators or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line break accident is also provided by low pressurizer pressure signals actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brought about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features actuation system also initiate containment spray upon sensing a high containment pressure signal (Hi-Hi Level). The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (2.0 psig). Since spurious actuation of containment spray is to be avoided, it is automatically initiated only on coincidence of Hi-Hi Level containment pressure sensed by both sets of two-out-of-three containment pressure signals.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line stop valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than

TABLE 3-1 (Continued)

ENGINEERED SAFETY FEATURES INITIATION INSTRUMENT SETTING LIMITS

No.	FUNCTIONAL UNIT	CHANNEL	SETTING LIMITS
6.	Steam Generator Water Level (low-low)	Auxiliary Feedwater	$\geq$5% of narrow range instrument span each steam generator
7.	Station Blackout (Undervoltage)	Auxiliary Feedwater	$\geq$40% nominal voltage
8a.	480v Emergency Bus Undervoltage (Loss of Voltage)	--	$\geq$220V with $\leq$4 sec. time delay
8b.	480v Emergency Bus Undervoltage (Degraded Voltage)	--	$\geq$396V with $\leq$180 sec. time delay

Amendment No.

TABLE 3-3

INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-DANCY	5 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1	SAFETY INJECTION					
a.	Manual	2	1	1	0	Cold Shutdown
b.	High Containment Pressure (Hi Level)	3	2	2	1	Cold Shutdown
c.	High Differential Pressure Between steam Lines	3/steam line	2/steam line	2/steam line	1/steam line	Cold Shutdown
d.	Pressurizer Low Pressure*	3	2	2	1	Cold Shutdown
e.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	2/line	1/2 in any 2 lines	1/line in each of 3 lines	2	Cold Shutdown
		4 T_{avg} Signals	2	3	2	
		4 Pressure Signals	2	3	2	
2	CONTAINMENT SPRAY					
a.	Manual	2	1	1	0	Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	2 sets of 3	2 of 3 in each set	2 per set	1/set	Cold Shutdown

* Permissible bypass if reactor coolant pressure less than 2000 psig.

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TABLE 3-3 (Continued)
INSTRUMENTATION OPERATING CONDITION FOR ENGINEERED SAFETY FEATURES

No.	FUNCTIONAL UNIT	1	2	3	4	5
3.	LOSS OF POWER					
	a. 480v Emergency Bus Undervoltage (Loss of Voltage)	2/bus	1/bus	1/bus	0	Cold Shutdown
	b. 480v Emergency Bus Undervoltage (degraded Voltage)	2/bus	2/bus	1/bus	0	Cold Shutdown
4.	AUXILIARY FEEDWATER					
	a. Stm Gen. Water Level-Low-Low					
	i. Start Motor Driven Pumps	3/stm gen	2 in any stm gen.	2 chan. in each stm gen	1	Reduce RCS temperature such that T $<$ 350°F
	ii. Start Turbine-Driven Pump	3/stm. gen	2/3 in each of two stm. gen.	2 chan. in each stm. gen.	1	T $<$ 350°F
	b. S.I. Start Motor-Driven Pumps	(All safety injection initiating functions and requirements)				
	c. Station Blackout Start Motor-Driven and Turbine-Driven Pumps	2	1	1	0	T $<$ 350°F
	d. Trip of Main Feedwater Pumps start Motor-Driven Pumps	2	1	1	0	Hot Shutdown

Amendment No.

TABLE 3-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

		1	2	3	4	5
NO.	FUNCTIONAL UNIT	NO. OF CHANNELS	NO. OF CHANNELS TO TRIP	MIN. OPERABLE CHANNELS	MIN. DEGREE OF REDUN-ANCY	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 or 4 CANNOT BE MET
1.	CONTAINMENT ISOLATION					
a.	Automatic Safety Injection (Phase A)	See Item No. 1 of Table 3-3				Cold Shutdown
b.	Containment Pressure (Phase B)	See Item No. 2 of Table 3-3				Cold Shutdown
c.	Manual					
	Phase A one out of two	2	1	1	0	Cold Shutdown
	Phase B one out of two	2	1	1	0	Cold Shutdown
2.	STEAM LINE ISOLATION					
a.	High Steam Flow in 2/4 Steam Lines Coincident with Low T_{avg} or Low Steam Line Pressure	See Item No. 1(e) of Table 3-3				Cold Shutdown
b.	High Containment Pressure (Hi Hi Level)	See Item No. 2b of Table 3-3				Cold Shutdown
c.	Manual	1/loop	1/loop	1/loop	0	Cold Shutdown
3.	FEEDWATER LINE ISOLATION					
a.	Safety Injection	See Item No. 1 of Table 3-3				
4.	CONTAINMENT PURGE AND PRESSURE RELIEF ISOLATION					
a.	Containment Radioactivity-High (R-11/R-12)	2	1	*	0	*

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* See Specification 3.1.F.

TABLE 3-5

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED (1)	3 INDICATOR/ RECORDER (1)
1. Pressurizer Water Level	3	2	Indicator/One Channel is recorded
2. Reactor Coolant System Subcooling Margin Monitor (2)	1	1	Indicator
3. PORV Position Indicator (Limit Switch)	1/Valve	1/Valve	Indicator and alarm
4. PORV Block Valve Position Indicator (Limit Switch)	1/Valve (3)	1/Valve (3)	Indicator (3)
5. Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indicator
6. Auxiliary Feedwater Flow Rate	1/S.G.	1/S.G.	Indicator

Footnotes:

(1) Except as specified in another footnote, columns 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or recorder(s) to be inoperable for up to 14 days. If the minimum number of channels required is not restored to meet the above requirements within the time periods specified, then:

a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

b. If the requirements of Columns 2 and 3 are not satisfied within an additional 48 hours, the reactor shall be cooled to below 350°F utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

TABLE 3-5 (Continued)

Footnotes (Cont'd):

(2) If the subcooling margin monitor is inoperable for more than seven (7) days, plant operation may continue for an additional thirty (30) days provide that steam tables are continuously maintained in the control room and the subcooling margin is determined and recorded once a shift.

(3) Except at times when the valve operator is deenergized in accordance with technical specification 3.1.A.4.c.

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B. During power operation, the following components may be inoperable:

1. Power operation may continue for seven days if one diesel is inoperable provided the 138 kv and the 13.8 kv sources of off-site power are available and the remaining diesel generators are tested daily to ensure operability and the engineered safety features associated with these diesel generator buses are operable.

2. Power operation may continue for 24 hours, if the 138 kv or the 13.8 kv source of power is lost, provided the three diesel generators are operable. This operation may be extended beyond 24 hours provided the failure is reported to the NRC within the subsequent 24-hour period with an outline of the plans for restoration of off-site power.

3. If the 138 KV power source is lost, in addition to satisfying the requirements of specification 3.7.B.2 above, the 6.9 KV bus tie breaker control switches 1-5, 2-5, 3-6, and 4-6 in the CCR shall be placed in the "pull-out" position and tagged to prevent an automatic transfer of the 6.9 KV buses 1,2,3 and 4.

4. One battery may be inoperable for 24 hours provided the other battery and two battery chargers remain operable with one battery charger carrying the dc load of the failed battery's supply system.

C. Gas Turbine Generators:

1. At least one gas turbine generator (GT-1, GT-2 or GT-3) and associated switchgear and breakers shall be operable at all times.

2. A minimum of 54,200 gallons of fuel for the operable gas turbine generator shall be available at all times.

3. If the requirements of 3.7.C.1 or 3.7.C.2 cannot be met, then, within the next seven (7) days, either the inoperable condition shall be corrected or an alternate independent power system shall be established.

4. If the requirements of 3.7.C.3 cannot be satisfied, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.7.C.3 cannot be met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Conditions of a system-wide blackout could result in a unit trip. Since normal off-site power supplies as required in Specification 3.7.A are not available for startup, it is desirable to be able to blackstart this unit within on-site power supplies as a first step in restoring the system to an operable status and restoring power to customers for essential service. Specification 3.7.D.1 provides for startup using the on-site gas turbine to supply the 6.9 KV loads and the diesels to supply the 480-volt loads. Tie breakers between the 6.9 KV and 480-volt systems are open so that the diesels would not be jeopardized in the event of any incident and would be able to continue to supply 480-volt safeguards power. The scheme consists of starting two reactor coolant pumps, one condensate pump, 2 circulating water pumps and necessary auxiliaries to bring the unit up to approximately 10% power. At this point, loads can be assumed by the main generator and power supplied to the system in an orderly and routine manner.

This Specification (3.7.D.2) is identical with normal start-up requirements as specified in 3.7.A except that off-site power is supplied exclusively from gas turbines with a minimum total power of 37 MW (nameplate rating) which is sufficient to carry out normal plant startup.

As a result of an investigation of the effect components that might become submerged following a LOCA may have on ECCS, containment isolation and other safety-related functions, a fuse and a locked open circuit breaker were provided on the electrical feeder to emergency lighting panel 218 inside containment. With the circuit breaker in the open position, containment electrical penetration H-70 is de-energized during the accident condition. Personnel access to containment may be required during power operation. Since it is highly improbable that a LOCA would occur during this short period of time, the circuit breaker may be closed during that time to provide emergency lighting inside containment for personnel safety.

When the 138 KV source of offsite power is out of service, the automatic transfer of 6.9 KV Buses 1,2,3 and 4 to offsite power after a unit trip could result in overloading of the 20 MVA 13.8 KV/6.9 KV auto-transformer. Accordingly, the intent of specification 3.7.B.3 is to prevent the automatic transfer when only the 13.8 KV source of offsite power is available. However, this specification is not intended to preclude subsequent manual operations or bus transfers once sufficient loads have been stripped to assure that the 20 MVA auto-transformer will not be overloaded by these manual actions.

References
1) FSAR-Section 8.2.1
2) FSAR-Section 8.2.3

TABLE 4.1-1 (CONTINUED)

Channel Description	Check	Calibrate	Test	Remarks
22. Accumulator Level and Pressure	S	R	N.A.	
23. Steam Line Pressure	S	R	M	
24. Turbine First Stage Pressure	S	R	M	
25. Logic Channel Testing	N.A.	N.A.	M	
26. Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	M	
27. Control Room Ventilation	N.A.	N.A.	R	Check damper operation for accident mode with isolation signal
28. Control Rod Protection (for use with LOPAR fuel)	N.A.	R	*	
29. Loss of Power				
a. 480v Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
b. 480v Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
30. Auxiliary Feedwater:				
a. Steam Generator Water Level (Low-Low)	N.A.	R	R	

* Within 31 days prior to entering a condition in which the Control Rod Protection System is required to be operable unless the reactor trip breakers are manually opened during RCS cooldown prior to T_{cold} decreasing below 350°F and the breakers are maintained open during RCS cooldown when T_{cold} is less than 350°F.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST
b. Station Blackout (Undervoltage)	N.A.	R	R
c. Trip of Main Feed-water Pumps	N.A.	N.A.	R
31. Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.
32. PORV Position Indicator (Limit Switch)	N.A.	R	R
33. PORV Block Valve Position Indicator (Limit Switch)	N.A.	R	R
34. Safety Valve Position Indicator (Acoustic Monitor)	N.A.	R	R
35. Auxiliary Feedwater Flow Rate	N.A.	R	R

Amendment No.

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the Auxiliary Feedwater System.

Objective

To verify the operability of the Auxiliary Feedwater System and its ability to respond properly when required.

Specification

A. The following surveillance tests shall be performed at refueling intervals:

(1) Verification of proper operation of auxiliary feedwater system components and initiating logic upon receipt of test signals for each mode of automatic initiation.

(2) Verification of the capability of each auxiliary feedwater pump to deliver full flow to the steam generators.

B. The above tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Basis

The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements. Testing of the auxiliary feedwater system will verify its operability. These specifications establish those surveillance tests to be performed at refueling intervals to verify operability of both the automatic initiation circuitry and the individual components necessary for proper functioning of the auxiliary feedwater system. This testing will verify proper component actuation upon receipt of all required automatic initiation signals and will verify that adequate system flow rates and pressures are obtained with proper valve positioning and pump full flow operation. Both control room instrumentation and visual observation of the equipment will be used to verify proper component operation.

The periodic "operational readiness" testing required by the ASME Code Section XI for pumps and valves in the auxiliary feedwater system is conducted as specified in the Indian Point Unit No. 2 Inservice Inspection and Testing Program and is therefore not included in these specifications.

References

FSAR - Sections 10.4, 14.1.9 and 14.2.5

ATTACHMENT C

Safety Evaluation

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

April, 1981

Safety Evaluation

By letter dated July 2, 1980, the NRC requested all pressurized water reactor licensees to propose new license conditions and technical specification revisions to incorporate various requirements relative to the TMI-2 Lessons Learned Category "A" Items. Consolidated Edison's February 26, 1981 letter to the NRC stated that these proposed changes would be submitted prior to the unit's return to service from the 1980-1981 refueling/maintenance outage.

The proposed licensed conditions are contained in Attachment A to this Application and the proposed technical specification changes are contained the Attachment B to this Application. These proposed changes will establish new requirements in the areas of pressurizer heaters, PORVs and associated block valves, critical valve position indication, inadequate core cooling instrumentation, containment isolation, auxiliary feedwater, leakage integrity/measurement program for systems outside containment and improved iodine measurement capability. The proposed changes contained in this Application have been based on the model technical specifications and model license conditions provided with the NRC's July 2, 1980 letter and will provide additional assurance that the Indian Point Unit No. 2 facility is operated within the limits determined appropriate following implementation of the TMI-2 Lessons Learned Category "A" Items.

Proposed changes with regard to requirements for the Shift Technical Advisor (STA) are not included in this Application. The changes requested by the NRC's July 2, 1980 letter for the STA were incorporated into Consolidated Edison's March 11, 1981 license amendment application relative to pending organizational and administrative changes at Consolidated Edison. Such changes have since been issued by the NRC Regulatory Staff as Amendment No. 68 to DPR-26 (dated March 27, 1981).

By letter dated December 31, 1980, Consolidated Edison committed to providing proposed technical specification revisions as part of the ongoing degraded grid voltage study for Indian Point Unit No. 2. These proposed changes are included in Attachment B to this Application and when implemented will establish requirements for the undervoltage (UV) protection relays associated with the 480V safeguards power buses.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.