Attachment No. 1
Response to Generic Letter No. 83-37
NUREG-0737 Technical Specifications
IPN-84-06

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 Each member of the plant staff shown in Fig. 6.2-2 shall meet or exceed the minimum qualifications of ANSl N18.1-1971 for comparable positions, except for (1) the Radiation and Environmental Services Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the training coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976 with the exception of the training program schedule.

6.4.3 A training program for use of the post-accident sampling system shall be maintained to ensure that the plant has the capability to obtain and analyze reactor coolant and containment atmosphere samples under post-accident conditions.

6.4.4 A training program shall be maintained to ensure that the plant has the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluent during and following an accident.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATING REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The Plant Operating Review Committee shall function to adivse the Resident Manager on all matters related to nuclear safety and all matters which could adversely change the plants environmental impact.

c. A Safety Limit Violation Report shall be prepared by the PORC. This report shall describe (1) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chairman of the SRC and the Senior Vice President-Nuclear Generation by the Resident Manager.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of the Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Post-accident sampling and analysis and maintenance of required equipment.

g. Collection and analysis or measurement of post-accident radioactive iodine and particulates in plant gaseous effluents and maintenance of required equipment.

6.8.2 Temporary changes to procedures above may be made provided:

a. The intent of the original procedures is not altered.

b. The change is approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's license on the unit affected.

c. The change is documented, reviewed by the PORC and approved by the Resident Manager within 14 days of implementation.

6.8.3 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Resident Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

d. Abnormal degradation of systems other than those specified in 6.9.1.7.c above designed to contain radioactive material resulting from the fission process. 7/

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Sealed source leakage in excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Primary coolant activity in excess of limits (Specification 3.1.D)

d. Seismic event analysis (Specification 4.10)

e. Inoperable fire protection and detection equipment (Specification 3.14)

f. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

g. Inoperable plant vent sampling or effluent monitoring capability (Table 3.5-4, Items 5 and 6).

h. Inoperable containment high-range radiation monitors. (Table 3.5-5, Item 22)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipmenmt related to nuclear safety.

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

7/ Sealed sources or calibration sources are not included under this item. Leakage of packing, caskets, mechanical joints and seal welds within the limits for identified leakage set forth in technical specifications need not be reported under this item.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transient cycles.

g. Records of training and qualifications for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the SRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of secondary water sampling and water quality.

6.11 RADIATION AND RESPIRATORY PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure as to maintain exposures as far below the limits specified in 10 CFR Part 20 as reasonable achievable. Pursuant to 10 CFR 20.103 allowance shall be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1 of 10 CFR 20.

3.5.4 In the event of instrumentation channel failure permitted by specification 3.5.2, the minimum degree of redundancy listed in Tables 3.5-2 through 3.5-4 may be reduced by one, but to not less than zero, and the minimum number of operable channels listed in these tables may be reduced by one, but not to less than one (except as noted in Table 3.5-3) for a period of 8 hours while instrument channels are tested. The failed channel may be blocked to prevent an unnecessary reactor trip during this time. In the case of three loop operation, the out-of-service channel is permitted to be bypassed during the test period.

3.5.5 The low pressurizer pressure safety injection trip shall be unlocked when the pressurizer pressure $\geq$ 2000 psig.

3.5.6 At least one source range and one intermediate range nuclear instrument channel shall be operable prior to reactor start-up.

3.5.7 When the reactor is not in the cold shutdown condition, the instrumentation requirements as stated in Table 3.5-5 shall be met.

3.5.8 A minimum of two channels of containment pressure must be operable when Tavg is greater than 350°F.

Basis

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

-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 based upon these signals. 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 System Actuation

The Engineered Safety Features actuation system also initiates 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 (~50% of containment design pressure) than the SIS (~10% of containment design pressure). 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 and coincidence with the S.I. Signal.

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 one steam generator by isolating the steam lines on high containment pressure (Hi-Hi Level) or high steam line flow. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effect of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steam line break inside the containment, by stopping the entry of feedwater.

Setting Limits

1. The Hi-Level containment pressure limit is set at about 10% of containment design pressure. Initiation of Safety Injection protects against loss of coolant(2) or steam line break(3) accidents as discussed in the safety analysis.

2. The Hi-Hi Level containment pressure limit is set at about 50% of containment design pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant(2) or steam line break accidents(3) as discussed in the safety analysis.

3. The pressurizer low pressure limit is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss of coolant accident as shown in the safety analysis(2). The trip is bypassed below 2000 psig to prevent inadvertent actuation of the Engineered Safeguards when the reactor is shutdown.

4. The steam line high differential pressure limit is set well below those differential pressures expected in the event of a large steam line break accident as shown in the safety analysis(3).

5. The high steam line flow measurement Δ P limit is set at approximately 40% of the full steam flow from no load to 20% load. Between 20% and 100% (full) load, the trip setpoint for the flow measurement Δ P is ramped linearly with respect to first stage turbine pressure from 40% of the full steam flow to 110% of the full steam flow. These setpoints will initiate safety injection in the case of a large steam line break accident. Coincident low Tavg setting limit for SIS and steam

line isolation initiation is set below the hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings Provide protection in the event of a large steam line break.(3)

Instrument Operating Conditions

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the Plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels are out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at Power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode, e.g., a two-out-of-three circuit becomes a one-out-of- two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod Protection from NIS, for the channel being tested, and (b) defeating the ΔT protection CHANNEL SET that is being fed from the NIS channel and (c) defeating the power mismatch section of Tavg control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

In the event that either the specified minimum number of operable channels or the minimum degree of redundancy cannot be met, the reactor and the remainder of the Plant is placed, utilizing normal operating procedures in that condition consistent with the loss of protection.

The source range and the intermediate range nuclear instrumentation and the turbine and steam-feedwater flow mismatch trip functions are not required to be operable since they were not used in the transient and safety analysis (FSAR Section 14).

References:

1) FSAR - Section 7.5
2) FSAR - Section 14.3
3) FSAR - Section 14.2.5

TABLE 3.5-5 (Sheet 1 of 3)

TABLE OF INDICATORS AND/OR RECORDERS AVAILABLE TO THE OPERATOR

	PARAMETER	1 NO. OF CHANNELS AVAILABLE	2 MIN. NO. OF CHANNELS REQUIRED**	3 INDICATOR/ RECORDER**
1)	Containment Pressure	6	2	Indicator
2)	Refueling Water Storage Tank Level	2	1	Indicator
3)	Steam Generator Water Level (Narrow Range)	3/Steam generator	*	Indicator
4)	Steam Generator Water Level (Wide Range)	1/steam generator	*	
5)	Steam Line Pressure	3/steam line	1/steam line	Indicator
6)	Pressurizer Water Level	3	2	Indicator/One Channel is Recorded
7)	RHR Recirculation Flow	4	3	Indicator
8)	Reactor Coolant System Pressure (Wide Range)	1	1	Recorder
9)	Cold Leg Temperature (T_c) (Wide Range)	4	1	Recorder
10)	Hot Leg Temperature (T_h) (Wide Range)	4	1	Recorder
11)	Containment Sump Level	2	1	Indicator
12)	Recirculation Sump Level	2	1	Indicator
13)	Temperature Sensors in Penetration Area of Primary Auxiliary Building	3	1	Alarm
14)	Temperature Sensors in Auxiliary Boiler Feedwater Pump Building	2	1	Alarm

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PARAMETER	1 NO. OF CHANNELS AVAILABLE-	2 NO. OF CHANNELS REQUIRED**	3 INDICATOR/ RECORDER**
15) Level Sensors in Lower Level of Tubine Building	2	1	Alarm
16) Reactor Coolant System Subcooling Margin Monitor	1	1	Recorder
17) PORV Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indicator
18) PORV Position Indicator (Limit Switch)	1/Valve	1/Valve****	Indicator and Alarm
19) PORV Block Valve Position Indicator (Limit Switch)	1/Valve***	1/Valve	Indicator
20) Safety Valve Position Indicator (Acoustic Monitor)	1/Valve	1/Valve	Indicator
21) Auxiliary Feedwater Flow Rate	1/Pump	1/Pump	Indicator
22) High-Range Containment***** Radiation Monitors (R25, R26)	2	1	Alarm
23) Core Exit Thermocouples	4/quadrant	2/quadrant	Indicator

* One level channel per steam generator (either wide range or narrow range) with at least two wide range channels.

** Column 2 and 3 may be modified to allow the instrument channels to be inoperable for up to 7 days and/or the recorder to be inoperable for up to 14 days.

*** Except at times when valve operator control circuit is de-engerized.

**** Except when the respective block valve is closed.

***** If the high-range containment monitor is determined to be inoperable when the reactor is above the cold shutdown condition, then restore the monitoring capability within 7 days, and

a) Initiate an alternate monitoring capability as soon as practical, but no later than 72 hours after identification of the failure of the monitor. If the monitor is not restored to operable status within 7 days, then,

b) Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system.

Table 3.5-5 (Sheet 3 of 3)

With the exception of the High-Range Containment Radiation Monitors, if the minimum number of channels required are not restored to meet the above requirements within the time periods specified, then:

1. 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.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

3. In either case, if the requirements of Columns 2 and 3 are not satisified within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

TABLE 3.5-4 (Sheet 2 of 3)
INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. NUMBER OF OPERABLE CHANNELS	4 MIN. DEGREE OF REDUNDANCY	5 OPERATOR ACTION IF CONDITIONS OF COL. 3 OR 4 CANNOT BE MET
3.	FEEDWATER LINE ISOLATION					
	a. Safety Injection	See Item No. 1 of Table 3.5-3				
4.	CONTAINMENT VENT AND PURGE					
	a. Containment Radioactivity-High (R11 and R12 monitor)	2	1	1	0	close all containment vent and purge valves when above cold shutdown
5.	PLANT EFFLUENT RADIOIODINE/ PARTICULATE SAMPLING (sample line common with monitor R13)	1	NA	1	0	(see note 3)
6.	Wide Range Plant Vent Monitor (R27)	1	NA	1	0	(see note 3)

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Table 3.5-4 (Sheet 3 of 3)

Notes

1. If the conditions of Columns 3 or 4 cannot be met, the reactor shall be placed in the hot shutdown condition, utilizing normal operating procedures, within 4 hours of the occurrence. If the conditions are not met within 24 hours of the occurrence, the reactor shall be placed in the cold shutdown condition, or the alternate condition if applicable, within an additional 24 hours.

2. Main steam isolation valves may be closed in lieu of going to cold shutdown if the circuitry associated with closing the valves is the only portion inoperable.

3. If the plant vent sampling capability or the wide-range vent monitor is determined to be inoperable when the reactor is above the cold shutdown condition, then restore the sampling/monitoring capability within 72 hours or:

   a) Initiate a pre-planned alternate sampling/monitoring capability as soon as practical, but no later than 72 hours after identification of the failure. If the capability is not restored to operable status within 7 days, then,

   b) Submit a Special Report to the NRC pursuant to Technical Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system.

TABLE 4.1-1 (SHEET 4 of 4)

CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
29. Reactor Coolant System Subcooling Margin Monitor	D	R	N.A.	
30. PORV Position Indicator (Limit Switch)	N.A.	R	R	
31. PORV Position Indicator (Acoustic Monitor)	D	R	R	
32. Safety Valve Position Indicator (Acoustic Monitor)	D	R	R	
33. Auxiliary Feedwater Flow Rate	N.A.	R	N.A.	
34. Plant Effluent Radioiodine/ Particulate Sampling (sample line common with monitor R 13)	N.A.	N.A.	R	
35. Wide-range Plant Vent Monitor (R27)	D	R	Q	
36. High-Range Containment Radiation Monitoring (R25, R26)	D	R	Q	
37. Core Exit Thermocouples	D	N.A.	N.A.	

S - Each Shift
D - Daily
W - Weekly
M - Monthly
P - Prior to each startup if not done previous week
Q - Quarterly
R - Each Refueling Outage
NA - Not Applicable

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Attachment No. 2
Response to Generic Letter No. 83-37
NUREG-0737 Technical Specifications
IPN-84-06

POWER AUTHORITY OF THE STATE OF NEW YORK
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286

Section I - Description of Change

The proposed changes to the Technical Specifications are shown in Attachment No. 1. Sections 3.5, 4.1, 6.4, 6.8, and 6.9 have been revised to reflect the guidelines and recommendations of Generic Letter No. 83-37 for specified, completed NUREG-0737 Items.

Section II - Purpose of the Change

The proposed changes expand the current IP-3 Technical Specifications to include the appropriate Limiting Conditions for Operation, Surveillance Requirements, and Administrative Controls for specified, completed NUREG-0737 Items, including the post-accident sampling system, the unit vent gaseous effluent monitor, and inadequate core cooling equipment.

Section III - Impact of Change

The proposed changes to the Technical Specifications will not alter the conclusions reached in the FSAR and SER nor will they impact the ALARA, Fire Protection Program or the Emergency Plan. The proposed change does not in any way lessen the existing LCO or Surveillance Requirements but rather serves to impose additional requirements to reflect completion of specified NUREG-0737 Items.

The Authority considers that the proposed changes can be classified as not likely to involve significant hazards considerations since the proposed changes "constitute an additional limitation, restriction, or control not presently included in the technical specifications". (Example (ii), Federal Register, Vol. 48, No. 67 dated April 6, 1983, page 14870).

Section IV - Conclusion

The incorporation of these modifications: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section V - References

(a) IP-3 FSAR
(b) IP-3 SER
(c) Generic Letter No. 83-37