50-247 CON ED CO. INDIAN POINT, UNIT 2

CHANGE NO. 2 TO THE TECH SPECS dtd 8-9-73

INDIAN POINT UNIT NO. 2

AEC DOCKET NO. 50-247

FACILITY OPERATING LICENSE DPR-26

CHANGE NO. 2 TO THE

TECHNICAL SPECIFICATIONS AND BASES

INSTRUCTION SHEET

Change No. 2 to the Technical Specifications and Bases (as issued originally as Appendix A to the Amendment No. 1 to Facility Operating License DPR-26) consists of corrections and additional information to be included in the Specifications. The new and revised pages should be added to the Technical Specifications and Bases as listed below.

Remove Old Sheet (front/back)	Insert Revised Sheet (front/back)
i/ii	i/ii
iii/iv	iii/iv
2.3-3/2.3-4	2.3-3/2.3-4
3.1-17/3.1-18	3.1-17/3.1-18
3.3-3/3.3-4	3.3-3/3.3-4
3.3-5/3.3-6	3.3-5/3.3-6
3.5-1/3.5-2	3.5-1/3.5-2
TABLE 3-3/TABLE 3-4	TABLE 3-3/TABLE 3-4
pages 3.9-1 through 3.9-6 (this entire specification is being replaced)	pages 3.9-1 through 3.9-14
TABLE 4.1-2 (2 pages)	TABLE 4.1-2 (6 pages)

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Remove Old Sheet	Insert Revised Sheet
(front/back)	(front/back)
	pages 4.11-1 through 4.11-4
pages 6.1-1 through 6.9-1 (this entire specification is being replaced)	pages 6.1-1 through 6.15-1

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1. For $(q_t - q_b)$ within the range between ΔI_1 and ΔI_2 given in the table below, $f(\Delta I) = 0$ (where q_t and q_b are percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power).

2. For each percent that $(q_t - q_b)$ is less than ΔI_1, the Delta-T trip setpoint shall be automatically reduced by 4.5% of its value at rated power. For each percent that $(q_t - q_b)$ is greater than ΔI_2, the Delta-T trip setpoint shall be automatically reduced by 2% of its value at rated power.

ΔI_1 and ΔI_2 are linear functions of the gain K_4. The proper limits on ΔI_1 and ΔI_2 shall be obtained from the following table which gives the allowable values corresponding to the actual value of K_4.

K_4	ΔI_1	ΔI_2
$\leq$ 1.01	$\geq$ -16.0	$\leq$ +16
1.04	$\geq$ -15.33	$\leq$ +14.5
1.07	$\geq$ -14.66	$\leq$ +13
1.10	$\geq$ -14.0	$\leq$ +11.5
1.13	$\geq$ -13.33	$\leq$ +10
1.16	$\geq$ -12.66	$\leq$ +8.5
1.19	$\geq$ -12	$\leq$ +7

(6) Low reactor coolant loop flow:

(a) $\geq$90% of normal indicated loop flow

(b) Low reactor coolant pump frequency - $\geq$ 57.5 cps

(7) Undervoltage - $\geq$ 70% of normal voltage

C. Other reactor trips

(1) High pressurizer water level - $\leq$ 92% of span

(2) Low-low steam generator water level - $\geq$ 5% of narrow range instrument span.

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

A. The reactor trips on low pressurizer pressure, high pressurizer level, and low reactor coolant flow for two or more loops shall be unblocked when:

1) Power range nuclear flux $\geq$ 10% of rated power, or

2) Turbine first stage pressure $\geq$ 10% of equivalent full load.

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates $\leq$ 60% of rated power. The single loop loss of flow reactor trip may be bypassed below 75% of rated power only when the overtemperature ΔT trip setpoint has been adjusted to the three-loop operation value given in 2.3.1.B-4 above. The resetting of the overtemperature ΔT trip shall be performed by the I & C Repair Unit under the direct supervision of the Operations Staff of Consolidated Edison Company. 2

Basis

The high flux reactor trips provide redundant protection in the power range for a power excursion beginning from low power. This trip was used in the safety analysis.(1)

The power range nuclear flux reactor trip high set point protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.(2) (3)

The source and intermediate range reactor trips do not appear in the specification as these settings are not used in the transient and accident analysis (FSAR Section 14). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of rated full power.

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 REDUNDANCY	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 and Low Level*	3**	1**	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 2
		4 T_{avg} Signals	2	3	2	
		4 Pressure Signals	2	3	2	
2	CONTAINMENT SPRAY					
a.	Manual	2	2	2	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.
** Each channel has two separate signals.
*** Must actuate 2 switches simultaneously.

TABLE 3-4

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MIN. OPERABLE CHANNELS	4 MIN. DEGREE OF REDUN-ANCY	5 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 Phase B	 2 See Item 2a of Table 3-3	 1	 1	 0	 Cold Shutdown 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				

F. LEAKAGE OF REACTOR COOLANT

Specification

1. If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation, a follow up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the indicated leakage is substantiated and is evaluated as unsafe or is determined to exceed 10 GPM, reactor shutdown shall be initiated as soon as practicable but no later than within 24 hours after the leak was first detected.

3. The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soom thereafter as practicable. The safety evaluation shall assure that the exposure to offsite personnel to radiation from the primary system coolant activity is within the guidelines of CFR 20. The reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected. 2

4. When the reactor is critical and above 2% power, two reactor coolant leak detection systems of different principles shall be in operation, with one of the two systems sensitive to radioactivity may be out-of-service for 48 hours provided two other systems are available.

Basis:

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; and therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks in the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Plant Operating Staff and will be documented in writing and approved by either the General Superintendent or his designated alternate. Under these conditions, an allowable primary system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also well within the capacity of one-charging pump and makeup would be available even under the loss of off-site power condition.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low leak rates. The rates of reactor coolant leakage to which the instrument

a. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours and the remaining two pumps are demonstrated to be operable.

b. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours and the other residual heat removal pump is demonstrated to be operable.

c. One residual heat removal exchanger may be out of service provided that it is restored to operable status within 48 hours.

d. Any valve required for the functioning of the system during and following accident conditions may be inoperable provided that it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

e. One channel of heat tracing may be out of service for 48 hrs. 2

B. Containment Cooling and Iodine Removal Systems

1. The reactor shall not be made critical unless the following conditions are met:

a. The spray additive tank contains not less than 4000 gal. of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. During power operation, the requirements of 3.3.B-1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the

requirements of 3.3.B-1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.B-1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

a. **Fan cooler unit 23, 24, or 25 may be non-operable during normal reactor operation for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.** 2

OR

Fan cooler unit 21 or 22 may be non-operable during normal reactor operation for a period not to exceed 7 days provided both containment spray pumps are demonstrated daily to be operable. 2

b. One containment spray pump may be out of service during normal reactor operation, for a period not to exceed 24 hours, provided the five fan cooler units are operable and the remaining containment spray pump is demonstrated to be operable.

c. Any valve required for the functioning of the system during and following accident condition may be inoperable provided it is restored to operable status within 24 hours and all valves in the system that provide the duplicate function are demonstrated to be operable.

C. Isolation Valve Seal Water System

The isolation valve seal water system shall be operable when the reactor is critical.

D. Weld Channel and Penetration Pressurization System

The weld channel and penetration pressurization system shall be operable when the reactor is critical.

E. Component Cooling System

1. The reactor shall not be made critical unless the following conditions are met:

   a. Two component cooling pumps on busses supplied by different diesels together with their associated piping and valves are operable.

   b. Two auxiliary component cooling pumps together with their associated piping and valves are operable.

   c. Two component cooling heat exchangers together with their associated piping and valves are operable.

2. During power operation, the requirements of 3.3.E-1 may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the conditions of 3.3.E-1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.E-1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

   a. One of the two operable component cooling pumps may be out of service provided the pump is restored to operable status within 24 hours.

   b. One auxiliary component cooling pump may be out of service provided the pump is restored to operable status within 24 hours and the other pump is demonstrated to be operable.

   c. One component cooling heat exchanger or other passive component may be out of service for a period not to exceed 48 hours provided the system may still operate at design accident capability.

F. Service Water System

1. The reactor shall not be made critical unless the following condition is met:

   Three service water pumps on the designated essential header together with their associated piping and valves are operable.

2. If during power operation one of the three service water pumps on the designated essential header or any of their associated piping or valves is found inoperable, the operator shall immediately proceed to place in service an essential service water system which meets the requirements of 3.3.F-1. If an essential service water system can not be restored within eight hours, the reactor shall be placed in cold shutdown condition.

G. Hydrogen Recombiner System

1. The reactor shall not be made critical unless the following conditions are met:

   a) Both hydrogen recombiner units together with their associated piping, valves, oxygen supply system and control system are operable, with the exception of one recombiner unit's equipment located outside of the containment which may be inoperable, provided it is under repair and can be made operable if needed. 2

   b) The containment atmosphere sampling system including the sampling pump, piping and valves are operable.

   c) Hydrogen and oxygen supplies shall not be connected to the hydrogen recombiner units except under conditions of an accident or those specified in 4.5.I.C.1.

3.5 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability:

Applies to plant instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

3.5.1 When the plant is not in the cold shutdown condition, the Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3-1. 2

3.5.2 For on-line testing or instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 3-2 through 3-4. No more than one channel of a particular protection channel set shall be tested at the same time. By definition, an instrumentation channel failure shall not be regarded as a channel being tested.

3.5.3 In the event the number of channels of a particular function in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 5 of Tables 3-2 through 3-4.

3.5.4 In the event of sub-system instrumentation channel failure permitted by specification 3.5.2, Table 3-2 through 3-4 need not be observed during the short period of time the operable sub-system channels are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

3.5.5 The cover plate on the rear of the safeguard panel, in the control room shall not be removed without the authorization from the operations staff. If a cover is removed, the event must be reported in the Semi-Annual Station Operation Report in accordance with Specification 6.6.4.D.

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 level and generates signals actuating the SIS active phase based upon the coincidence of 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 and level 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 of low steam line pressure.

3.9 EFFLUENT RELEASE - RADIOACTIVE MATERIALS

Applicability

Applies to the controlled release of radioactive liquids and gases from the plant.

Objective

To define the limits and conditions for the controlled release of radioactive effluents to the environs to ensure that these releases are as low as practicable.

Specification

A. Liquid Effluents

1. The concentrations of radioactive liquid effluents at the discharge canal - Hudson River confluence shall not exceed the values specified in 10CFR20 Appendix B, Table II, Column 2 when averaged over one year.

2. The release from the site of radioactive liquids excluding tritium and dissolved noble gases, shall not exceed 20 curies in three consecutive months.

3. Steam generator blowdown radioactivity shall be continuously monitored. Whenever the monitor is inoperable, grab samples shall be taken and analyzed at least five days per week with detectable concentrations equal to those required for weekly analysis in Table 4.1-2.

Steam generator blowdown radioactivity shall normallly be recorded.

4. During release of liquid radioactive waste from a Waste Condensate Tank and/or a monitor tank, the following conditions shall be met:

   a) The effluent control monitor shall be set to alarm and automatically close the waste discharge valve prior to exceeding the limits specified in 3.9.A.1 above.

   b) Liquid waste radioactivity and flow rate shall be continuously monitored during release. If this requirement cannot be met, continued release of liquid effluents shall be permitted only during the succedding 48 hours provided that during this 48-hour period, a sample of each tank shall be analyzed and valving shall be checked prior to the discharge.

5. All equipment which has been installed for the purpose of reducing the activity levels in all liquid effluents from the site shall be maintained and, when it has been projected that quarterly liquid effluent activities, excluding tritium and dissolved noble gases, will exceed 2.5 curies from the site, all treatment equipment which could significantly reduce effluent activity shall be utilized.

6. If, at any time, the activity in a Waste Condensate Tank or a monitor tank, excluding tritium and dissolved noble gases, exceeds 10 curies, the contents of that tank shall promptly be routed to the Liquid Waste Holdup Tank or the CVCS holdup Tank.

7. When the release of radioactive liquids, excluding tritium and dissolved noble gases, exceeds 5 curies from the site during a consecutive three months period, the licensee shall report such as per Section 6.12.2.b.4 of the Technical Specifications.

B. Gaseous Effluents

1. The maximum release rate for gross radioactivity of gaseous effluents from the site shall not exceed:

$$\left[\frac{\bar{E}_{\gamma 1} Q_1}{8.3 \times 10^{-2}} + \frac{(\bar{E}_{\gamma 2} + \bar{E}_{\beta 2}) Q_2}{1.3 \times 10^{-2}} \right] \leq 1.0$$

where: Q_1 is the measured release rate from Unit 1 (Ci/sec).

$\bar{E}_{\gamma 1}$ is the average gamma energy per disintegration for the gaseous effluents from Unit 1 (Mev/dis).

Q_2 is the measured release rate from Unit 2 (Ci/sec).

$\bar{E}_{\gamma 2}$ is the average gamma energy per disintegration for the gaseous effluents from Unit 2 (Mev/dis).

$\bar{E}_{\beta 2}$ is the average beta energy per disintegration for gaseous effluents from Unit 2 (Mev/dis).

2. a) The maximum release rate for I-131 in the gaseous effluents shall not exceed:

$$\frac{Q_1}{8.5 \times 10^{-6}} + \frac{Q_2}{1.8 \times 10^{-6}} \leq 1.0$$

b) The maximum release rate for particulates with half lives longer than eight days in the gaseous effluents from Unit 2 shall not exceed $8 \times 10^{1} \overline{MPC}_a$ Ci/sec, where $\overline{MPC}_a$ is the composite maximum permissible concentration in air as defined in Appendix B, Table II, Column 1 of 10CFR Part 20 and note 1 thereto.

3. a) The quarterly release of gaseous radioactivity, excluding I-131 and particulates with half-lives longer than eight days, shall be limited to 16% of the limits specified in 3.9.B.1 above.

b) The quarterly release of I-131 and particulates with half-lives longer than eight days shall be limited to 8% of the limits specified in 3.9.B.2 above.

4. During releases of gaseous wastes from the waste gas decay tanks, the following conditions shall be met:

a) The gross activity monitor and particulate activity monitor shall be operable.

b) Automatic isolation devices capable of limiting gaseous release rates to within the values specified in 3.9.B.1 and 3.9.B.2 above shall be operable.

5. The maximum activity to be contained in one waste gas decay tank shall not exceed 11,400 Ci at Unit 1 and and 16,500μCi at Unit 2 (i.e. equivalent Xe-133 curies).

6. When it has been projected that the quarterly release of airborne radioactive effluents will exceed 12.5% of the limits specified in Specification 3.9.B.3, the appropriate equiptment shall be used and procedures followed to significantly reduce airborne effluent activity.

7. When the release of radioactive gases exceeds 50% of the limits specified in Specification 3.9.B.3, the licensee shall report such as per Section 6.12.2.b.5 of the Technical Specifications.

<u>Basis</u>

Although it is expected that annual releases of liquid radioactive effluents will not result in exceeding a small fraction of the concentration limits of 10CFR20, Appendix B, Table II, Column 2, Specification 3.9.A.1 permits the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may result in releases higher than the design objective levels.

Dilution, in the discharge canal common to Indian Point Unit Nos. 1 and 2, of all radioactive liquid effluents is accomplished by the Circulating Water System. Unit No. 1 is equipped with two 140,000 gpm circulating water pumps and six smaller nuclear and conventional service water pumps. Unit No. 2 is equipped with six 140,000 gpm circulating water pumps and six smaller service water pumps. The actual circulating water flow under various operating conditions will be calculated from the head differential across the pumps and the manufacturer's head-capacity curves.

Specification 3.9.A.2 establishes an upper limit for the release from the site of radioactive liquids, excluding tritium and dissolved gases, of 20 curies during three consecutive months. The intent of this Specification is to aim towards the design objectives while permitting the licensee the flexibility of operation to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the levels normally achievable when the plant and the liquid radwaste equipment are functioning as designed. Releases of up to 20 curies during any quarter will result in concentrations of radioactive material in liquid effluents at small percentages of the limits specified in 10 CFR Part 20.

When fuel failure and steam generator tube leakage exist concurrently, steam generator blowdown can be a significant source of radioactivity released to the environment. Continuous monitoring of this potential source of activity is intended to assure operational attention to compliance with Specification 3.9.A.1 and 3.9.A.2 above. In the event that the continuous monitor and recorder become inoperable, every effort will be made to repair or replace them and, in the interim, frequent sampling of the liquid blowdown will provide the means to limit radioactive releases as specified in 3.9.A.1 and 3.9.A.2.

Specification 3.9.A.4 requires that suitable equipment to dilute and monitor the releases of radioactive materials in liquid effluents are operating during any period these releases are taking place.

Specification 3.9.A.5 requires that the licensee maintain all liquid effluent treatment equipment and that the suitable purification equipment be utilized in an attempt to meet the objectives stated above concerning control of radioactive liquid effluents. All liquid effluent streams whose activity can be significantly reduced should be treated when it is apparent that lack of such treatment by existing systems will result in exceeding the limits of these objectives.

Specification 3.9.A.6 limits the amount of radioactivity that may be inadvertently released to the environment to an amount such that the limits of specification 3.9.A.2 are not exceeded. Every effort will be made to insure that activity as high as 10 curies does not enter any of these tanks at any one time.

In addition to the limiting conditions for operation listed under Specification 3.9.A.2, the reporting requirements of Specification 3.9.A.7 require that the licensee shall identify the cause whenever the release rate of radioactive effluents, excluding tritium and noble gases, exceeds 5.0 curies during any quarter and describe the propose program of action to reduce such release rate. This report must be filed within 30 days following the quarter in which the 5.0 curie release occurred.

Detailed dose calculations for several locations offsite have been made. These calculation consider site meteorology, buoyancy characteristics, and isotopic content of the effluent of each unit. Independent dose calculations for several locations offsite have been made by the AEC staff, and the most critical one was chosen to set the release limit. The controlling distance is 760 meters to the south for Unit 1 and 510 meters to the south-southwest for Unit 2 at the site boundary. The use and occupancy of any portions of its site will be controlled in such a manner that no member of the public using or occupying any such portion will receive a dose to the whole body from the radioactive effluents from the site in excess of 0.5 Rems in any calendar year. Suitable onsite monitoring stations and procedures will be maintained by the licensee so that appropriate action may be taken to carry out this commitment.

The method utilized by the staff is described in Section 7-5.2.5 of "Meteorology and Atomic Energy - 1968," equation 7.63 being used . The results of these calculations are conservative and thus chosen to be used as the basis of establishement of the limits. Based on these calculations, a continuous release rate of gross radioactivity in the amount of $8.3 \times 10^{-2}/\overline{E}_{\gamma}$ curies/sec from the Unit 1 plant stack will not result in offsite annual doses in excess of the limits specified in 10 CFR Part 20. The $\bar{E}_\gamma$ determination need consider only the average gamma energy per disintegration since the controlling whole body dose is due to the cloud passage over the receptor and not cloud submersion in which the beta dose could be additive.

The dose analysis performed by the AEC staff for radioactive releases from the Unit 2 ventilation stack included an evaluation of the beta dose as well as the gamma dose. The staff assumed that such releases would be equivalent to ground level releases which could result in a beta dose from cloud submersion. The methods utilized are the same as used for the Unit 1 releases to determine the gamma dose contribution while the beta dose contribution was determined using the method described in Section 7.4 of "Meteorology and Atomic Energy - 1968," equation 7.21 being used. Therefore, the gamma dose contribution was determined on the basis of a finite cloud passage, and the beta dose contribution on the basis of a semi-infinite cloud submersion both for a ground level release. The beta dose contribution was reduced from an air dose to a depth dose equivalent to 200 mg/cm^2 penetration (depth distance for lens of the eye) by a 4 Mev beta particle (maximum beta energy of noble gas releases) which the staff considers to represent conservatively a whole body dose equivalence for beta particles, i.e., lens of the eye dose. The beta depth dose determination method utilized is described in Section 7.4.1.2 of "Meteorology and Atomic Energy - 1968," equation 7.25 being used. The calculated reduction factor on the beta dose for depth penetration by the staff was 0.3 for the above assumptions which was used in the beta dose equation 7.21. Based on these calculations, a continuous release rate of gross radioactivity in the amount of $1.3 \times 10^{-2} / (\bar{E}_{\gamma} + \bar{E}_{\beta})$ curie/sec from the Unit 2 reactor building ventilation stack will not result in offsite annual doses in excess of the limits specified in 10 CFR Part 20.

Since the calculated and measured dose contribution from Units 1 and 2 are additive for purposes of meeting the limits specified in 10CFR Part 20, the equation given in Specification 3.9.B.1 combines the release rates from Units 1 and 2.

The average gamma energy per disintegration used in the equation of Specification 3.9.B.1 will be based on the average composition of gases determined from the Unit 1 plant stack and Unit 2 ventilation stack which will assure that offsite doses are not in excess of the limits specified in 10 CFR Part 20.

The gamma energy per disintegration for those radioisotopes determined to be present from the isotopic analyses shall be as given in "Table of Isotopes," C. M. Lederer, J. M. Hollander, and I. Perlman, Sixth Edition 1967. Other documents may be used for the average gamma energy per disintegration for radioisotopes provided such documents reference the above works. For Kr-89 and Xe-138, the gamma energy per disintegration shall be as given in "Energy Release from the Decay of Fission Products," Nuclear Science and Engineering: 3,726-746 (1958) until values are published in "Table of Isotopes." Using these reference gamma energies per disintegration with the compostion of radiogases in the releases, the average gamma energy per disintegration, $\bar{E}_{\gamma}$ shall be determined for each unit.

The beta energy per disintegration for those radioisotoped determined to be present from the appropriate isotopic analysis for Unit 2 releases shall be as given in USNRDL-TR-802, II. Spectra of Individual Negatron Emitters (Beta Spectra), H. Hogan P. E. Zigman, and J. L. Mockin. Other documents may be used for the average beta energy per disintegration for radioisotopes provided such document reference the above works. The average beta energy shall be used as the beta energy per disintegration for each radioisotope evaluated. Using these reference beta energies per disintegration with the appropriate composition of radiogases in the Unit 2 ventilation stack, the average beta energy per disintegration, $\bar{E}_{\beta}$, shall be determined.

Detailed calculations of ground level air concentrations of I-131 and particulates with half lives greater than 8 days at several offsite locations have been made. These calculations consider site meteorology for Unit 2 releases. Since Unit 1 has an elevated release while Unit 2 has an equivalent ground level release, the controlling sector and distance for air concentrations of I-131 and particulates with half lives greater than 8 days at ground level are different for the two release points.

The controlling sector for Unit 1 releases is the north sector at a distance of 3840 meters (X/Q= 3.5×10^{-7} sec/m^3) for the inhalation dose with a limiting release of 280μCi/sec of I-131. The controlling sector for Unit 2 releases is the south-southwest sector at a distance of 510 meters (X/Q= 1.8×10^{-5} sec/m^3) for the inhalation dose with a limiting release rate of 5.5μCi/sec of I-131. The nearest milk cow is located in the south-southwest sector at a distance of 10200 meters (7 miles). In accordance with Regulatory Guide 1.42, the applicable X/Q at the nearest milk cow is 2.4×10^{-7} sec/m^3 for Unit 2 and 5.1×10^{-8} sec/m^3 for Unit 1. Therefore, the cow-milk-child thyroid cycle remains controlling for both units. The release of I-131 from Unit 1 and Unit 2 were considered additive while particulates with half-lives greater than eight days from Unit 1 and Unit 2 were not considered to be additive as a basis for operation but would be additive for the purpose of meeting 10CFR Part 20 requirements.

The assumptions used by the AEC staff for these calculations were: (1) onsite meteorological data were used for the most critical 22.5 degree sector; (2) building wake credit was used; and (3) to consider possible reconcentration effects, a reduction factor of 233 was applied to the I-131 cow-milk-child thyroid cycle (Regulatory Guide 1.42) and a reduction factor of 700 for possible ecological chain effects from radioactive particulate releases.

The releases limits for I-131 and particulates with half lives greater than eight days have been analyzed separately since the most probable critical organ for the radioiodines will be the thyroid while for the predominent radioactive particulates (Cs-137, Sr-89, Sr-90, etc.) will not be the thyroid and would not be additive to the same organ.

Specification 3.9.B.3 established an upper limit for the release from the site of radioactive gaseous effluents in a consecutive 3 month period. The intent of this Specification is to aim towards the design objectives while permitting the licensee the flexibility of operation to assure that the public is provided a dependable source of power under unusual operating conditios which may temporarily result in releases higher than the levels normally achievable when the plant is functioning as designed.

Releases within the limits of this specification will not exceed a small fraction of 10CFR20 with respect to concentration of airborne effluents at any point on the site boundary or at a point where any dairy cow may pasture.

Specification 3.9.B.4 requires a that sutiable equipment to monitor and limit the radioactive gaseous releases is operating duirng any period these releases are taking place.

Specification 3.9.B.5 limits the maximum offsite dose to below the limits of 10 CFR Part 20, postulating that the rupture of a wast gas decay tank holding the maximum activity release all of the contents to the atmosphere.

Specification 3.9.B.6 requires that the licensee utilize the gaseous radwaste equipment avaible when necessary to attempt to meet the design objectives concerning control of radioactive airborne effluents.

In addition to the limiting conditions for operation listed under 3.9.B.1, 3.9.B.2 and 3.9.B.3 the reporting requirements of Specification 3.9.B.7 require that the licensee shall identify the cause whenever the radioactive gaseous release rate exceeds twice-the-annual design objective averaged over a consecutive 3 month period, and describe the proposed program of action to reduce such release rate. The report must be filed within 30 days following the quarter in which more than twice the design release rate occurred.

4.11 RADIOACTIVE MATERIALS

Applicability

Applies to the periodic test and record requirements and sampling and monitoring methods used for facility effluents.

Objective

To ensure that radioactive liquid and gaseous releases from the facility are maintained as low as practicable and within the limits specified in Specification 3.9.A and 3.9.B.

Specification

A. Liquid Effluents

1. Radioactive liquid waste sampling and activity shall be performed in accordance with Table 4.1-2.

2. Prior to the release of each batch of liquid effluent from an isolated tank a sample shall be taken from that tank for determination of gross radioactive content and a portion will be set aside for a latter composite analysis of significant gamma emitters to demonstrate compliance with Specification 3.9.A.3 & 4.

3. Records and reports of the sampling and analyses results shall be maintained in accordance with Specification 6.13.2.k. Estimates of the error associated with these analyses should be included.

4. The liquid waste discharge radiation monitor shall be calibrated at least quarterly by means of a check source and annually with a known radioactive source. The monitor, as described, shall also have an instrument channel test monthly and a sensor check daily.

5. The performance of automatic isolation valves and discharge tank selection valves shall be checked monthly.

B. Airborne Effluents

1. Radioactive gaseous waste sampling and activity analysis shall be performed in accordance with Table 4.1-2.

2. The waste gas decay tank effluent monitor shall be tested prior to any release of radioactive gas from a decay tank and shall be calibrated at refueling intervals. The calibration procedure shall consist of exposing the detector to a referenced calibration source in a controlled reproducible geometry. The source and geometry shall be referenced to the original monitor calibration which provides the applicable calibration curves.

3. During power operation, the condenser vacuum pump discharge (steam jet air ejector) shall be continuously monitored for gross radiogas activity. The monitor shall not be inoperable for more than a week. Whenever this monitor is inoperable, grab samples shall be periodically taken daily and analyzed for gross radioactivity (β, γ).

4. Records and reports of the sampling and analysis results shall be maintained in accordance with Specification 6.13.2.k. Estimates of the error associated with these analyses should be included.

5. At least annually, automatic initiation and closure capability of waste gas system shall be verified.

6. All waste gas monitors shall be calibrated at least quarterly by means of a check source and annually with a known radioactive source. Each monitor shall have an instrument channel test at least monthly and sensor check at least daily.

Basis

The surveillance requirements given under Specification 4.11.A provide assurance that liquid wastes are properly controlled and monitored during any planned release of radioactive materials in liquid effluents. (A batch of discharge is defined as the volume of liquid released over a period of not more than one week.) These surveillance requirements provide the data for tha licensee and the Commission to evaluate the plant's performance relative to radioactive liquid waste released to the environment. Reports on the quantities of radioactive materials released in liquid effluents shall be furnished to the Commission on the basis of Section 6 of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The surveillance requirements given under Specification 4.11.B provides assurance that radioactive gaseous effluents from the plant are properly controlled and monitored over the life of the plant. These surveillance requirements provide the data for the licensee and the Commission to evaluate the plant's performance relative to radioactive gaseous wastes released to the environment. Reports on the quantities of radioactive materials released in gaseous effluents shall be furnished to the Commission on the basis of Section 6 of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the Licensee or others, the Commission may from time to time require the Licensee to take such action as the Commission deems appropriate.

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
1.	Reactor Coolant Samples	Gross Activity (1)	5 days/week (1)	3 days
		Radiochemical (2)	Monthly	45 days
		$\bar{E}$ Determination	Semi-annually (3)	30 weeks
		Tritium Activity	Weekly (1)	10 days
		F, Cl & O_2	Weekly	10 days
2.	Reactor Coolant Boron	Boron Concentration	Twice/week	5 days
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Monthly	45 days
4.	Boric Acid Tank	Boron Concentration	Twice/week	5 days
5.	Boron Injection Tank	Boron Concentration	Monthly	45 days
6.	Spray Additive Tank	NaOH Concentration	Monthly	45 days
7.	Accumulator	Boron Concentration	Monthly	45 days
8.	Spent Fuel Pit	Boron Concentration	Prior to Refueling	NA*
9.	Secondary Coolant	Iodine-131	Weekly (4)	10 days
10.	Liquid Radwaste Discharge Line Monitor	Gross β and γ Activity	Continuous (5)	NA
11.	Composite Discharge Canal Sampler	Radioactivity Analysis	Continuous Composite (6)	NA
12.	Liquid Radwaste Montor Tanks	Radioactivity Analysis	Prior to Each Batch Release	NA
		Dissolved Noble Gases (7)	One Batch/Month	45 days
		Ba 140, La 140, I 131 (8)	Weekly Proportional Composite (12)	10 days

Change No. 2

TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
12.	Liquid Radwaste Monitor Tanks (Continued)	γ Emitters (9)	Monthly Proportional Composite (12)	45 days
		H-3 (7)	Monthly Proportional Composite (12)	45 days
		Gross α (10)	Monthly Proportional Composite (12)	45 days
		Sr 89, Sr 90 (11)	Quarterly Proportional Composite (12)	15 weeks
13.	Environmental Release Point (Plant Vent)	Iodine 131 and Particulate Activity	Continuous (5)	NA
		Gross β and γ Activity (gas) (15, 16)	Continuous (5)	NA
		H-3 as HTO (gas) (17)	Quarterly	15 weeks
		I 131 (charcoal) (19)	Weekly (10)	10 days
		I 133, I 135 (charcoal) (19)	Quarterly	15 weeks
		Gross β and γ Activity (particulate) (20)	Weekly (18)	10 days
		Ba 140, La 140, I 131 (particulate) (20)	Weekly (18)	10 days
		Individual γ Emitters (particulate) (19)	Monthly Composite of Weekly Samples	45 days
		Gross β and γ Activity (particulate) (20)	Monthly Composite of Weekly Samples	45 days
		Sr 89, Sr 90 (particulate) (20)	Quarterly Composite of Monthly Samples	15 weeks
		Gross (long lived α) (20)	One week Sample per Quarter	15 weeks
14.	Containment Iodine-Particulate Monitor or Gas Monitor	Iodine 131 and Particulate Activity or Gross Gaseous Activity	Continuous When Operating at Power (21)	NA

TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests
15.	Gas Decay Tank Release	H-3 as HTO (17)	Each Tank Release	NA
		Individual γ Emitters (15)	Each Tank Release	NA
16.	Containment Purge Release	H-3 as HTO (17)	Each Purge	NA
		Individual γ Emitters (15)	Each Purge	NA
17.	Condenser Air Ejector Release	H-3 as HTO (17)	Quarterly	15 weeks
		Individual γ Emitters (15)	Monthly (16)	45 days
18.	Steam Generator Blowdown	Gross β and γ Activity (10)	Weekly	10 days
		Ba 140, La 140, I 131 (8)	Weekly	10 days
		Dissolved Noble Gases (7)	One Sample/Month	45 days
		γ Emitters (8)	Monthly Proportional Composite (22)	45 days
		H-3 (7)	Monthly Proportional Composite (22)	45 days
		Gross α (10)	Monthly Proportional Composite (22)	45 days
		Sr 89, Sr90 (11)	Quarterly Proportional Composite (22)	15 weeks

TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES:

* NA- Not Applicable

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/cc, and when the activity levels exceed 10% of limits specified in 3.9.A.1 and 3.9.B.1, the sampling frequency shall be increased to a minimum of once each day.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

(3) $\bar{E}$ determination will be started when the gross analysis indicates $\geq$ 10 μCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 μCi/cc in accordance with Specification 3.1.D.

(4) When the iodine-131 activity exceeds 10% of the limit in Specification 3.4.A, the sampling frequency shall be increased to a minimum of once each day.

(5) Except as indicated in Specification 3.9.

(6) Only when releasing radioactive effluents; sampling once/shift when discharging may be substituted when monitors are inoperable.

(7) Detectable concentration = 10^{-5} μCi/ml (24)

(8) Detectable concentration = 10^{-6} μCi/ml

(9) Detectable concentration = 10^{-7} μCi/ml (13)

(10) Detectable concentration = 10^{-7} μCi/ml

(11) Detectable concentration = 10^{-8} μCi/ml (14)

TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES (Continued)

(12) A proportional sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged from the plant.

(13) For certain mixtures of gamma emitters, it may not be possible to measure radionuclides in concentrations near their sensitivity limits when other nuclides are present in the sample in much greater concentrations. Under these circumstances, it will be more appropriate to calculate the concentrations of such radionuclides using observed ratios with these radionuclides which are measureable.

(14) One quarterly proportional composite sample will be collected and analyzed for Sr 89 and Sr 90. The proportional inputs to this sample will be from the monitor tank and steam generator blowdown.

(15) Detectable concentration = 10^{-4} μCi/cc (23)

(16) For certain mixtures of gamma emitters, it may not be possible to measure radionuclides at levels near their sensitivity limits when other nuclides are present in the sample at much higher levels. Under these circumstances, it will be more appropriate to calculate levels of such radionuclides using observed ratios with those radionuclides which are measureable.

(17) Detectable concentration = 10^{-4} μCi/cc for HTO. Sample points will also be studied for gaseous releases to determine fractions of releases of tritium in other forms.

(18) When the iodine or particulate release rate is greater than the release rate given in Specification 3.9.B.3, the sample shall be taken and analyzed five days a week until a steady release level has been established.

(19) Detectable concentration = 10^{-10} μCi/cc

(20) Detectable concentration = 10^{-11} μCi/cc

(21) Except as indicated in Specification 3.1.F.5

Change No.2

TABLE 4.1-2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

FOOTNOTES (Continued)

(22) Since these potential sources of liquid radioactive waste are discharged on a continuous rather than batch basis, the volume of liquid to be used as a basis for obtaining proportional samples from secondary blowdown and leakage is that amount discharged over the period of one week.

(23) Analyses shall also be performed following each refueling, startup, or similar operational occurrence which could alter the mixture of radionuclides.

(24) The above detectability limits for activity analysis are based on technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable and when nuclides are measured below the stated limits, they should also be reported.

SECTION 6

ADMINISTRATIVE CONTROLS

6.0 INTRODUCTION

Administrative controls are the means by which Station operators are subject to management control. Measures specified in this section provide for the assignment of responsibilities, Station organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls and reporting requirements. Each of these measures is necessary to ensure safe and efficient facility operation.

SPECIFICATIONS

6.1 RESPONSIBILITY

6.1.1 The Chief Engineer shall have direct responsibility for the safe operation of his assigned nuclear unit. This responsibility shall be expressly delegated to a specified member of the Station management staff during any off-duty status period of the Chief Engineer.

6.1.2 The Station Manager shall have direct responsibility for the safe operation and maintenance of all facilities comprising Indian Point Station. This responsibility shall be expressly delegated to a specified member of the Station management staff during any off-duty status period of the Station Manager.

6.1.3 The Manager, Nuclear Services, shall have direct responsibility for the development and implementation of the Station's programs in the areas of radiation safety, environmental monitoring, nuclear training, cold and hot chemical surveillance, testing and reactor engineering. This responsibility shall be expressly delegated to a specified member of the Station management staff during any off-duty status period of the Manager, Nuclear Services.

6.1.4 The Station Quality Assurance Engineer shall have direct responsibility for overseeing and directing the Station's quality assurance program. This responsibility shall be expressly delegated to a specified member of the Station management staff during any off-duty status period of the Station Quality Assurance Engineer.

6.1.5 In all matters pertaining to operation of the nuclear facility, and to these Technical Specifications, the Chief Engineer shall report to and be directly responsible to the Station Manager who, in turn, shall report to the Manager, Nuclear Power Generation Department, as indicated on the corporate management organization chart of Figure 6.1-1.

CORPORATE ORGANIZATION

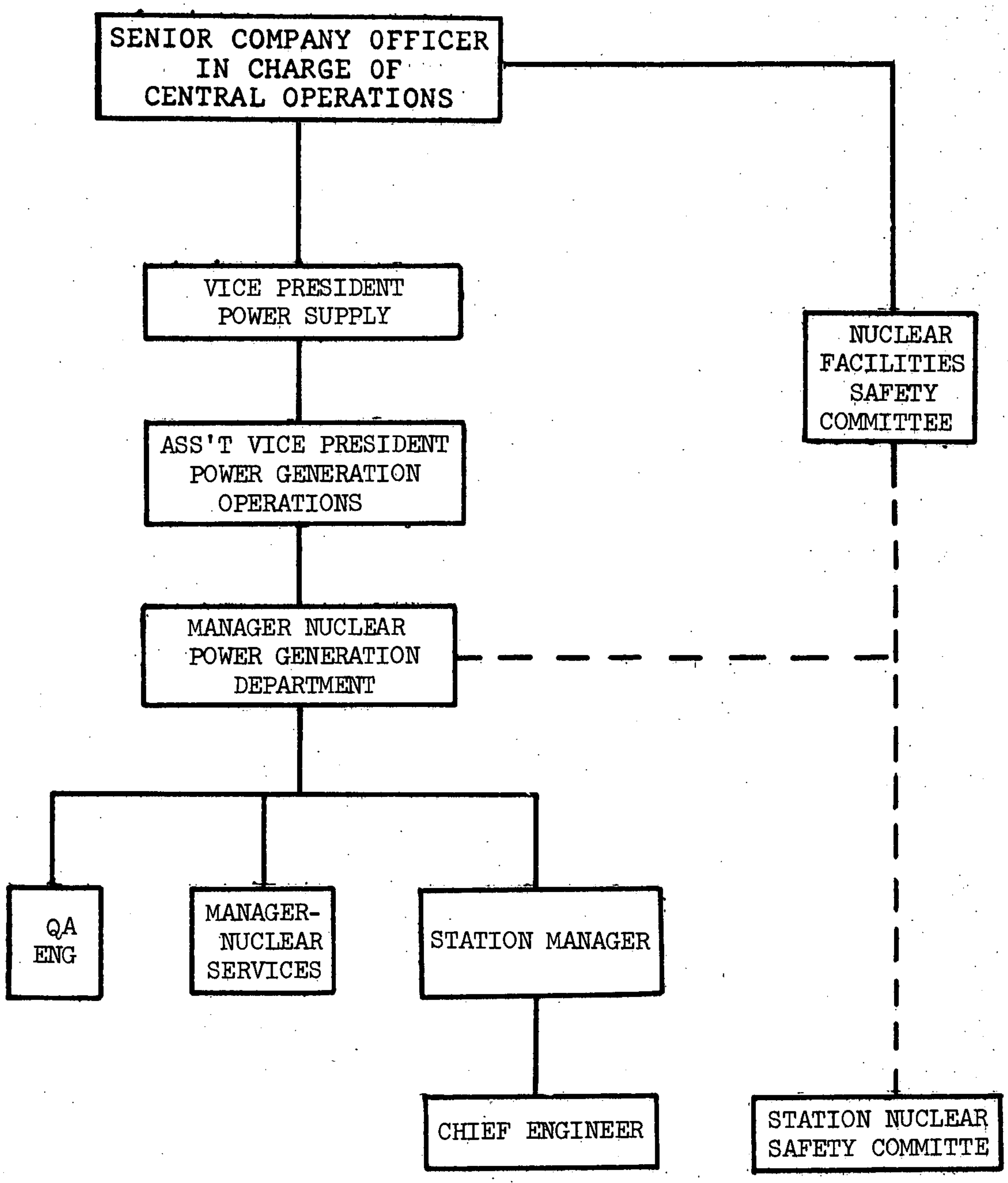


FIGURE 6.1-1

6.2 STATION STAFF ORGANIZATION

6.2.1 The Station staff organization shall be as shown in Figure 6.2-1 and shall function as follows:

a. The minimum number and type of licensed and unlicensed operating personnel required on site for each shift shall be as shown in Table 6.2.1.

b. One licensed Operator shall be in the control room at all times when there is fuel in the reactor.

c. Two licensed Operators shall be in the control room during startup, scheduled shutdown and during recovery from trips caused by transients or emergencies.

d. An individual possessing the qualifications of a Health Physics Technician shall be on site at all times nuclear fuel is located thereon.

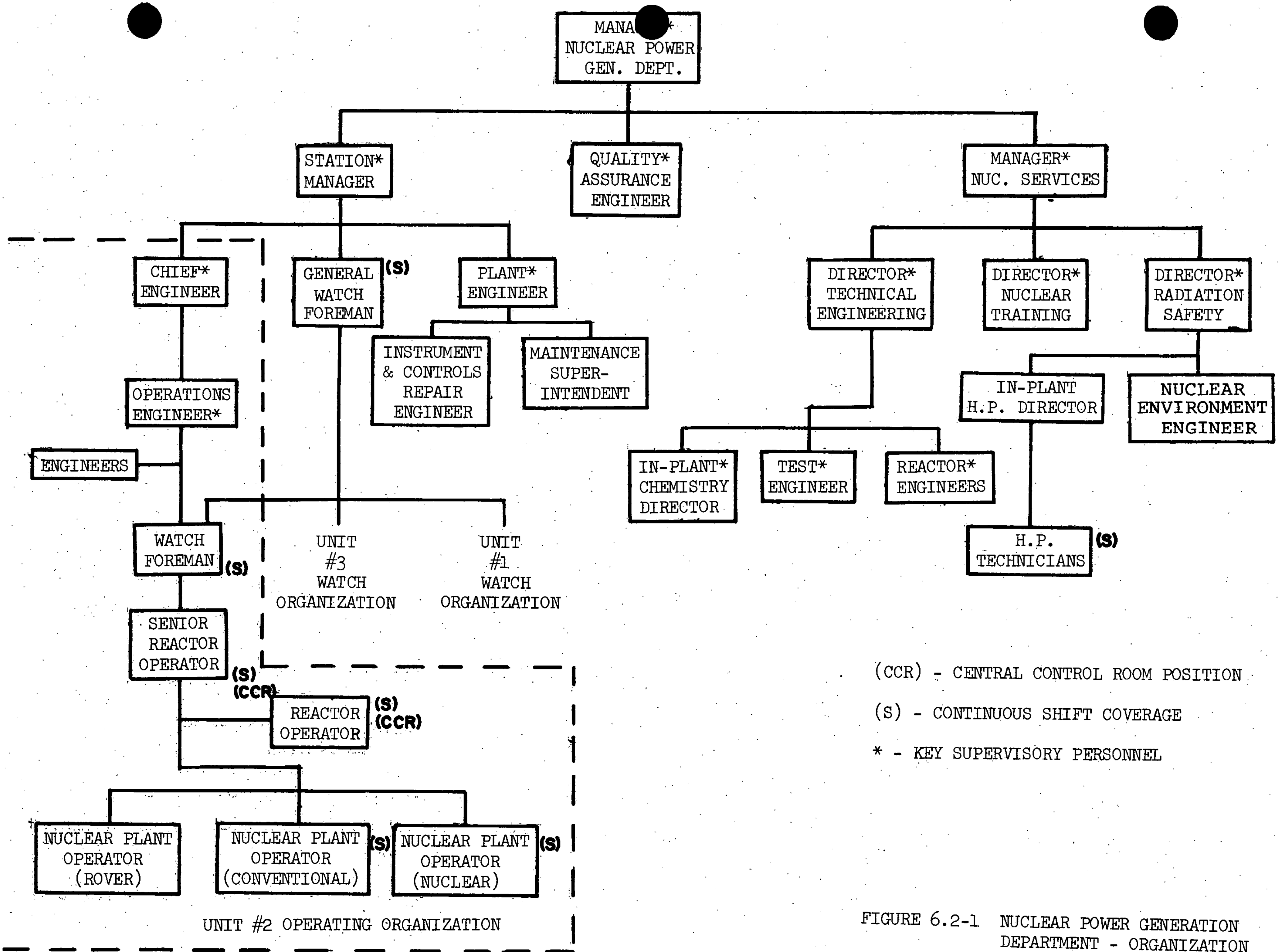
e. All operations including core alterations shall be performed under the direct supervision of and inividual holding a Senior Operating License. This individual shall have no other responsibilities during this assignment, and shall functionally report to the individual on-site exercising overall responsibility for the facility.

Table 6.2-1

Minimum Shift Crew Composition

License Category	During Operations Involving Fuel Handling	During Cold Shutdown or Refueling Periods	At All Other Times
Senior Operator License	2 *	1	1
Operator License	1	1	2
Non-Licensed	(As Required)	1	2

* Includes individual with SRO license supervising fuel movement as per Section 6.2.1(e).



(CCR) - CENTRAL CONTROL ROOM POSITION

(S) - CONTINUOUS SHIFT COVERAGE

* - KEY SUPERVISORY PERSONNEL

FIGURE 6.2-1 NUCLEAR POWER GENERATION DEPARTMENT - ORGANIZATION CHART

6.3 PERSONNEL QUALIFICATIONS

6.3.1 Except as specifically authorized by the Commission, the provisions as set forth in ANSI N18.1-1971 shall be met or exceeded by the Station staff.

6.4 RETRAINING AND REPLACEMENT TRAINING

6.4.1 A training program shall be implemented to maintain the overall proficiency of the operating organization. Except as provided below, this program shall consist of both retraining and replacement training elements and shall meet the minimum provisions outlined in ANSI N18.1-1971.

6.4.2 The training program referenced in 6.4.1 above, shall be under the overall direction of a specified member of the Station management staff.

6.4.3 Records of individual Station staff member's qualifications, including their specific training and retraining, shall be maintained.

6.5 REVIEW AND AUDIT

6.5.1 Station Nuclear Safety Committee - A Station staff committee shall be constituted and function as described below:

a. Membership:

1. Chairman: Director - Technical Engineering.

2. Vice Chairman: Plant Engineer.

3. Member: Director - Radiation Safety.

4. Member: Chief Engineer - Unit No. 1.

5. Member: Chief Engineer - Unit No. 2.

6. Member: Chief Engineer - Unit No. 3.

7. Member: Reactor Engineer (of affected facility).

8. Member: Quality Assurance Engineer.

b. Alternates: Alternate members shall be appointed by each member; however, no more than two (2) members shall serve on the committee at any one time.

c. Consultants: Additional personnel with expertise in specific areas may serve as consultants to the Station Nuclear Safety Committee.

d. Meeting frequency; Bi-Monthly, and as required, on call of the Chairman.

e. Quorum: Chairman or Vice Chairman and four members including designated alterantes.

f. Responsibilities:

1. The committee shall review proposed normal, off-normal and emergency operating procedures; proposed maintenance procedures and proposed changes as determined by the Station Manager, Manager - Nuclear Services, or the Q.A. Engineer, to be related to nuclear safety.

2. The committee shall review proposed tests and experiments.

3. The committee shall review proposed changes or modifications to plant systems or equipment.

4. The committee shall review all permanent procedure changes, plant modifications, tests or experiments which were authorized for implementation without prior review by the committee in accordance with the administrative controls required by specification 6.8.2.

5. The committee shall review all proposed changes to the Technical Specifications.

6. The committee shall investigate all violations of Technical Specifications and shall prepare and forward a report covering its evaluation

and recommendations to prevent recurrence to the Manager, Nuclear Power Generation Department, and to the Chairman of the Nuclear Facilities Safety Committee.

7. The committee shall perform special reviews and investigations and render reports thereon as requested by the Chairman of the Nuclear Facilities Safety Committee or the Manager, Nuclear Power Generation Department.

8. The committee shall monitor that periodic drills are conducted on emergency procedures, including evacuation (partial or complete) of the site and that the adequacy of communications with off-site support groups is checked.

9. The committee shall periodically review the Station Security Plan and implementing procedures and shall submit recommended changes to Manager, Nuclear Power Generation Department with a copy of the report to the Chairman of the Nuclear Facilities Safety Committee.

10. The committee shall periodically review the Emergency Plan and implementing procedures and, if appropriate, shall submit recommended changes to Manager, Nuclear Power Generation Department with a copy of the report to the Chairman of the Nuclear Facilities Safety Committee.

g. Authority:

1. The Station Nuclear Safety Committee shall be advisory to the Manager, Nuclear Power Generation Department.

2. The Station Nuclear Safety Committee shall recommend in writing to the Manager, Nuclear Power Generation Department, approval or disapproval of proposals under Items f (1) through (5) above.

3. In the event of disagreement between the recommendations of the Station Nuclear Safety Committee and the actions contemplated by the Manager, Nuclear Power Generation Department, the course of action determined by the Department Manager will be followed. Any such disagreement will be made a matter of Station record, however, and a copy of the record shall be forwarded for immediate review to the Chairman of the Nuclear Facilities Safety Committee.

4. The Station Nuclear Safety Committee shall make tentative determinations as to whether or not proposals considered by the committee involve unreviewed safety questions, as defined in 10 CFR 50.59. All such determinations shall be subject to review by the Nuclear Facilities

Safety Committee, but only those proposals possibly involving an unreviewed safety question, in the opinion of the committee, need be reviewed prior to adoption.

h. Records: Minutes shall be kept at the Station of all meetings of the Station Nuclear Safety Committee and copies shall be sent to the Manager, Nuclear Power Generation Department and to the Chairman of the Nuclear Facilities Safety Committee.

i. Procedures:

Written committee administrative procedures shall be prepared and maintained that describe:

1. The method of submission and the content of presentations to the committee;

2. provisions for the use of subcomittees;

3. the method for review and approval of written committee evaluations and recommendations;

4. the criteria governing the type of review mechanism which will be used, e.g., meeting of the members, telephone conference call, or independent review of written material; and

5. the distribution of minutes.

6.5.2 Company Nuclear Review Board - A corporate committee shall be constituted and function as described below:

a. Membership:

The Committee shall have a permanent membership of at least 5 persons of which a majority are independent of the Nuclear Power Generation Department and shall include technically competent persons from departments of Consolidated Edison having a direct interest in nuclear plant design, construction, operation or in nuclear safety. In addition, persons from departments not having a direct interest in nuclear plant design, construction, operation or nuclear safety may serve as members of the Committee if experienced in the field of nuclear energy. The Chairman and Vice Chairman will be senior Officials of the Company experienced in the field of nuclear energy.

The Chairman of the Nuclear Facilities Safety Committee, hereafter referred to as the Chairman, shall be appointed by the Chairman of the Board or the President of the Company.

The Vice Chairman shall be appointed by the Chairman of the Board of the President of the Company. In the absence of the Chairman, he will serve as Chairman.

The Secretary shall be appointed by the Chairman of the Committee.

Committee members from departments having a direct Managerial interest in nuclear plant design, construction and operation or in nuclear safety shall be designated in writing by the Vice President of the Company who is responsible for the functioning of the department subject to the approval of the Chairman. Committee members from other departments may be appointed by the Chairman with the concurrence of the Vice President of that department.

Alternates:

a. Each permanent voting member may appoint an alternate to serve in his absence. Committee records shall be maintained showing each such current designation.

b. No more than two alternates shall serve on the Committee at any one time.

c. Alternate members shall have voting rights.

b. Qualifications:

1. A minimum of four permanent voting members shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences and

possess a minimum of three years of professional level experienee in nuclear services, nuclear plant operation, or nuclear engineering, and the necessary overall nuclear background to determine when to engage the services of consultants and contractors for solving problems beyond the expertise of the Company organization.

2. Members and alternates shall collectively have the capability required to review the areas of:

   a. reactor operations

   b. nuclear engineering

   c. chemistry and radiochemistry

   d. metallurgy

   e. instrumentation and control

   f. radiological safety

   g. mechanical and electrical engineering

   h. quality assurance

   i. nuclear-related environmental effects

   j. other appropriate fields required by the unique characteristics of the nuclear units involved.

3. When the nature of a particular situation dictates, special consultants shall be utilized to provide expert advice to Committee members upon request of any two permanent members.

4. If sufficient expertise in the areas indicated in b.2 above is not available from within the Committee membership, staff specialists and/or outside consultants shall be used to advise the Committee.

c. Charter:

1. The Committee shall be constituted by a written charter stating:

a. Subjects within purview of the Committee.

b. Responsibility and authority.

c. Mechanisms for convening meetings.

d. Provisions for use of subcommittees.

e. Authority for access to unit records.

f. Reporting requirements.

d. Records:

1. Written minutes of each meeting shall be prepared, formally approved, and promptly

distributed to each Committee member, the Senior Company Officer in charge of Central Operations, and other appropriate members of management having responsibility in the areas reviewed. Permanent copies of these minutes shall be retained as specified in Section 6.13.

2. Written reports of each audit function performed including follow-up action and re-audits shall be prepared, approved, and forwarded to the Senior Company Officer in charge of Central Operations and to the other management members having responsibility in the areas audited. Copies of these reports shall be retained as specified in Section 6.13.

e. Meeting Frequency:

Meetings shall be convened no less frequently than semi-annually.

f. Quorum:

1. No less than a majority of the permanent board voting membership shall constitute a quorum.

2. Either the Chairman or Vice Chairman shall be present.

3. No more than a minority of the quorum shall have direct line responsibility for nuclear unit operation.

g. Responsibilities:

1. The Committee shall review:

   a. Proposed tests and experiments whose performance may constitute an unreviewed safety question as defined in 10 CFR 50.59.

   b. Proposed changes to equipment, systems, and procedures which are described in the FSAR and which may involve an unreviewed safety question as defined in 10 CFR 50.59, or which are referred to the Committee by the plant staff.

   c. Proposed Technical Specification changes or license amendments.

   d. Violations of applicable statues, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having safety significance on facility operation.

   e. Significant operating abnormalities or deviations from normal performance of facility equipment.

   f. Abnormal occurrences as defined in Section 1.8.

   g. The Station Emergency Plan and implementing procedures.

   h. The Station Security Plan and implementing procedures.

i. Nuclear Environmental Monitoring Program.

j. Nuclear safety matters deemed essential to the safe operation of the facility by the Nuclear Power Generation Department Manager, the Station Nuclear Safety Committee, the Station Manager of Power Plants and the Manager - Nuclear Services.

k. Reports and meeting minutes of the Station Nuclear Safety Committee.

l. Reports submitted to the Atomic Energy Commission and associated responses.

2. The Committee shall direct and evaluate the results of periodic audits performed to ensure safe facility operation. These audits shall encompass:

   a. The conformance of facility operation to all provisions contained within the Technical Specifications, applicable license requirements, and company rules and internal policy.

   b. The performance of the entire facility staff.

   c. The results of all actions taken to correct anomalies occurring in facility equipment, structures, systems or method of operation.

d. The adequacy of the Quality Assurance Program to meet the criteria specified in 10 CFR 50, Appendix "B".

e. Any other area of facility operation considered appropriate by the Committee or the Senior Company Officer in charge of Central Operations.

h. Authority:

The Committee shall report to and advise the Senior Company Officer in charge of Central Operations in writing on all matters related to nuclear safety.

6.6 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE

6.6.1 Any abnormal occurrence shall be promptly reported to the Manager, Nuclear Power Generation Department and promptly reviewed by the Station Nuclear Safety Committee.

6.6.2 The Station Nuclear Safety Committee shall prepare a report for each abnormal occurrence. This report shall describe the cause of the occurrence, the corrective action taken, and committee recommendations for appropriate action to reduce the probability of recurrence.

6.6.3 All such reports shall be submitted to the Manager, Nuclear Power Generation Department, with a copy to the Chairman of the Nuclear Facilities Safety Committee, for review and approval of any recommendations.

6.6.4 The Manager, Nuclear Power Generation Department shall report the circumstances of any abnormal occurrence to the AEC as specified in Section 6.12 "Plant Reporting Requirements".

6.7 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

6.7.1 If a safety limit is exceeded, the reactor shall be promptly shut down and reactor operation shall only be resumed in accordance with the provisions of 10 CFR 50.36 (c) (1) (i).

6.7.2 A report of each safety limit violation shall be promptly made to the Manager, Nuclear Power Generation Department and the Chairman of the Nuclear Facilities Safety Committee.

6.7.3 The Manager, Nuclear Power Generation Department shall report to the AEC as specified in Section 6.12 in the event a safety limit is exceeded.

6.7.4 The Station Nuclear Safety Committee shall prepare a complete report of each safety limit violation which includes appropriate analyses and evaluations of (1) applicable circumstances preceding the occurrence, (2) effects of the occurrence upon facility components, systems or structures, and (3) recommended corrective action to prevent recurrence. This report shall be submitted to the Manager, Nuclear Power Generation Department with a copy to the Chairman of the Nuclear Facilities Safety Committee.

6.8 PROCEDURES

6.8.1 Detailed written procedures covering the areas listed below, including applicable check-off lists and instructions, shall be prepared, approved as specified in Section 6.8.2, and adhered to:

a. Normal startup, operation, and shutdown of the reactor and of all systems and components involving nuclear safety.

b. Refueling operations.

c. Actions to be taken in response to potential malfunctions of systems, or components, including the responses to alarms as appropriate, suspected primary system leaks and abnormal reactivity changes.

d. Emergency conditions involving either potential or actual releases of radioactivity.

e. Facility modifications and preventive or corrective maintenance involving nuclear safety.

f. Surveillance and testing requirements of the nuclear facility.

g. Tests and Experiments involving nuclear safety.

h. Station Security Plant implementation.

i. Station Emergency Plan implementation.

6.8.2 All written procedures required by Specification 6.8.1, and any changes to them shall be reviewed and approved for implementation in accordance with written administrative control procedures approved by the Manager, Nuclear Power Generation Department, with the concurrence of the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. The administrative control procedure required by this specification shall, as a minimum, clearly establish the criteria which will govern when pre-implementation review by the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee is required.

6.8.3 Except with respect to changes, Specification 6.8.2 shall not be applicable to any procedure approved for implementation prior to adoption of Specification 6.8.2.

6.9 RADIATION AND RESPIRATORY PROTECTION SYSTEM

6.9.1 RADIATION PROTECTION SYSTEM

Procedures for personnel radiation protection shall be prepared and adhered to for all station operations. These procedures shall be formulated to maintain radiation exposures received during operation and maintenance as far below the limits specified in 10 CFR 20 as practicable. The procedures shall include planning, preparation and appropriate training for each operation and maintenance activity. They shall also include exposure allocation, radiation and contamination control techniques, and final debriefing.

6.9.2 RESPIRATORY PROTECTION PROGRAM

a. Pursuant to 10 CFR 20.103 (c) (1) and (3), allowance may 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, subject to the following conditions and limitations.

1. The limits provided in Section 20.103 (a) and (b) shall not be exceeded.

2. If the radioactive material is of such form that intake through the skin or other additional route

is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column 1, of 10 CFR 20.

3. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in 20.101. These materials shall be subject to applicable process and other engineering controls.

b. In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

1. The limits specified in Paragraph a. above are not exceeded.

2. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1 of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be initially estimated by dividing the ambient airborne concentration by the protection factor specified in Table 6.9-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

3. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

4. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

a. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

b. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

c. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

d. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair and storage.

e. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

f. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

5. The licensee shall use equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.9-1. Equipment not approved under U. S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table 6.9-1.

6. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.9-1 in selecting and using respiratory protective equipment.

c. These specifications with respect to the provisions of 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20, Section 20.103, which would make this specification unnecessary.

TABLE 6.9-1
PROTECTION FACTORS FOR RESPIRATORS

Description	MODES 1/	PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask 4/ 7/	NP	5	21B 30 CFR 14.4(b) (4)
Facepiece, full 7/	NP	100	21B 30 CFR 14.4(b) (5); 14F 30 CFR 13
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full	CF	1,000	19B 30 CFR 12.2(c) (2) Type C(i)
Facepiece, full 7/	D	100	19B 30 CFR 12.2(c) (2) Type C(ii)
Facepiece, full	PD	1,000	19B 30 CFR 12.2(c) (2) Type C(iii)
Hood	CF	5/	6/
Suit	CF	5/	6/
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full 7/	D	100	13E 30 CFR 11.4(b) (2) (i)
Facepiece, full	PD	1,000	13E 30 CFR 11.4(b) (2) (ii)
Facepiece, full	R	1,000	13E 30 CFR 11.4(b) (1)
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere supplying respirator		Protection factor for type and mode of operation as listed above.	19B CFR 12.2(e) or applicable schedules as listed above

1/, 2/, 3/, 4/, 5/, 6/, 7/, (These notes are on the following pages)

TABLE 6.9-1
(Cont'd)

1/ See the following symbols:

- CF: continuous flow
- D: demand
- NP: negative pressure (i.e., negative phase during inhalation)
- PD: pressure demand (i.e., always positive pressure)
- R: recirculating (closed citcuit)

2/ a. For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

b. The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency (above 99.9% removal of efficiency by U. S. Bureau of Mines type dioctyl phthalate (DOP) test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

3/ Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide.

TABLE 6.9-1
(Cont'd)

See also footnote 5/, below, concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U. S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U. S. Bureau of Mines in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of this part are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

6.10 INDUSTRIAL SECURITY PROGRAM

6.10.1 An industrial security program shall be maintained throughout the life of the facility in accordance with the provisions of the Station Security Plan. Annual review of the Station Security Plan shall be performed by the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee.

6.10.2 Any actual or attempted introduction into the Generating Station area of any dangerous weapon, explosive, or material capable of producing injury or damage to persons or property, or that in any way could seriously affect the safe operation of the facility, shall be reported immediately upon detection to the Station security force.

6.10.3 Investigations shall be conducted by the Station security force of all attempted or actual security infractions in cooperation with any Federal, State, or local agencies involved, and a report filed with the Station Manager, Manager, Nuclear Power Generation Department, and Chairman of the Station Nuclear Safety Committee.

6.11 EMERGENCY PLAN

6.11.1 A Site Emergency Plan shall be maintained throughout the life of the facility in accordance with the provisions of 10 CFR 50, Appendix "E".

6.11.2 Site evacuation exercises shall be conducted at least annually utilizing applicable provisions contained within the Emergency Plan. This exercise shall involve coordination with off-site support groups and include communication checks.

6.11.3 The Emergency Plan and implementing procedures shall be reviewed on an annual basis by the Station Nuclear Facilities Safety Committee.

6.12 PLANT REPORTING REQUIREMENTS

The following information shall be submitted to the USAEC in addition to those reports required by Title 10, Code of Federal Regulations.

6.12.1 Operations Reports:

Operations Reports shall be submitted in writing to the Director of Licensing, USAEC, Washington, D. C. 20545.

a. Startup Report

A summary report of facility startup and power escalation testing shall be submitted following receipt of an operating license, following amendments to the licenses involving a planned increase in power level, following the installation of fuel that has a different design and/or was fabricated by a different vendor, or following modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the facility. The report shall include a description of the measured and predicted values and describe any corrective action taken to obtain acceptable operation. Startup reports shall be submitted within 90 days following completion of the startup test program or the beginning or resumption of commercial power operation.

b. First Year Operation Report

A report shall be submitted after completion of the first year of commercial power operation. This report shall be incorporated into the next due Semi-Annual Operating Report and shall cover the following:

1. an evaluation of facility performance to date in comparison with design predictions and specifications;

2. an assessment of the safety analyses submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analyses;

3. an assessment of the performance of structures, systems and components important to safety;

4. a progress and status report on any items identified as requiring additional information during the operating license review or during the startup of the plant, including items discussed in the AEC's Safety Evaluation, including supplements, items on which additional information was required as conditions of the license, and items identified in the licensee's startup report;

5. a report of inplant radiation levels which are a factor of two or greater than the estimates provided in the Final Safety Analysis Report.

c. Semi-Annual Operating Reports

Semi-Annual operating reports covering operation of the previous six months shall be submitted within 60 days after January 1 and July 1 of each year. The first such report shall begin with the date of initial criticality and shall cover the period ending December 31, 1973. These reports shall include the following:

1. Operations Summary

   A summary of operating experience occurring during the reporting period that relates to the safe operation of the facility, including a summary of:

   a. performance characteristics (e.g., equipment and fuel performance);

   b. changes in procedures which were necessitated by (a) or which otherwise were required to improve the safety of facility operations;

   c. results of surveillance tests and inspections required by these Technical Specifications;

   d. the results of any containment leak rate tests performed during the reporting period;

   e. a brief summary of those change, tests and experiments requiring authorization from the

Commission pursuant to 10 CFR 50.59(a), and;

f. changes in the facility operating staff for those positions designated as key supervisory personnel on Figure 6.2-1.

2. Power Generation

A summary of power generated during the reporting period including:

a. gross thermal power generated (in MWH);

b. gross electrical power generated (in MWH);

c. net electrical power generated (in MWH);

d. number of hours the reactor was critical;

e. number of hours the generator was on-line;

f. Histogram of thermal power vs. time.

3. Shutdowns

Descriptive materials covering all outages occurring during the reporting period. For each outage, information shall be provided on:

a. the cause of the outage;

b. the method of shutting down the reactor, e.g., automatic trip or manually controlled deliberate shutdown;

c. duration of the outage;

d. facility status during the outage; e.g., cold shutdown or hot shutdown;

e. corrective action taken to prevent repetition, if appropriate.

4. Maintenance

A summary of corrective maintenance (excluding preventive maintenance) performed during the reporting period on safety-related (as defined in ANSI 18.7-1972) systems and components and on systems and components that reduce or prevent the release of radioactive material to the environs. Information shall be provided on:

a. the system or component involved;

b. the cause of the malfunction;

c. the affect of the malfunction on safe operation;

d. corrective action taken to prevent repetition;

e. precautions taken to provide for reactor safety during repair.

5. Changes, Tests and Experiments

A brief description and a summary of the safety evaluation for those changes, tests, and experiments carried out without prior Commission approval, pursuant to the requirements of Part 50.59(b) of the Commission's regulations.

6. Radioactive Effluent Releases

A statement of the quantity of radioactivity released from the plant, with data summarized on a monthly basis, following the format of Appendix A of USAEC Safety Guide 21 of January 1972:

a. Gaseous Effluents

1. Gross Radioactivity Releases

a. Total gross radioactivity (in curies), primarily noble and activation gases.

b. Maximum gross radioactivity release rate during any one-hour period.

c. Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

d. Percent of Technical Specification limit.

2. Iodine Releases

a. Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

b. Percent of Technical Specification limit for I-131 released.

3. Particulate Releases

a. Total gross radioactivity (β, γ) released (in curies) excluding background radioactivity.

b. Gross alpha radioactivity released (in curies) excluding background radioactivity.

c. Total gross radioactivity released (in curies) of nuclides with half-lives greater than eight days.

d. Percent of Technical Specification limit for particulate radioactivity with half-lives greater than eight days.

b. Liquid Effluents

1. Total gross radioactivity (β, γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

2. The maximum concentration of gross radioactivity (β, γ) released to the unrestricted

area (averaged over the period of release).

3. Total tritium and total alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

4. Total dissolved noble gas radioactivity (in curies) and average concentration released to the unrestricted area.

5. Total volume (in liters) of liquid effluent released.

6. Total volume (in liters) of dilution water used prior to release from the restricted area.

7. Total radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

8. Percent of Technical Specification limit for total activity released.

7. Solid Radioactive Waste

a. The total amount of solid radioactive waste packaged (in cubic feet).

b. The total estimated gross radioactivity (in curies) of packaged material.

c. Disposition of material including date and destination is shipped off-site.

8. Occupational Personnel Radiation Program

Tabulate the number of occupational personnel exposures for Plant operations personnel (permanent and temporary) in the following exposure increments for the reporting period:

Less than 100 mRem, 100-250 mRem, 250-500 mRem, 500-750 mRem, 750-1,000 mRem, 1-2 Rem, 2-3 Rem, 3-4 Rem, 4-5 Rem, 5-6 Rem, greater than 6 Rem, and greater than 2500 mRem. Tabulate the number of personnel receiving more than 500 mRem exposure in this reporting period according to duty function, i.e., routine plant surveillance and inspection, routine plant maintenance, special plant maintenance, routine refueling operation, special refueling operation, and other job-related exposures. Annually tabulate the number of personnel receiving more than 3000 mRem and report major cause(s).

9. Primary Coolant Radioactivity Levels

Tabulate on a monthly basis, the maximum, the average and minimum values for the following primary coolant system parameters:

Gross radioactivity in µCi/ml

Suspended Solids in parts per million

Gross tritium in μCi/ml

Iodine-131 in μCi/ml

Ratio of Iodine-131 to Iodine-133

Hydrogen in cc per kg.

Lithium in parts per million

Boron in parts per million

Oxygen in parts per million

Chloride in parts per million

pH at 25°C

6.12.2 NON-ROUTINE REPORTS

a. Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone and in writing promptly to the Director of the Region 1 Regulatory Operations Office, (cc to the Director of Licensing), followed by a written report within 10 working days to the Director of Licensing (cc to the Director of the Regional Regulatory Operations Office) in the event of an abnormal occurrence as defined in Section 1.0. The written report to the Directorate of Licensing on these abnormal occurrences, and to the extent possible, the preliminary notification to Regulatory Operations shall: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and (d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

b. Unusual Events

A written report shall be forwarded within 30 days to the Director of Licensing and to the Director of the Region 1 Regulatory Operations Office, in the event of:

1. Discovery of any substantial errors in the transient or accident analyses, or in the methods used for such analyses, as described in the Safety Analysis Report or in the bases for the Technical Specifications.

2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications or in the Safety Analysis Report.

3. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to

perform its safety function.

4. Discovery of the release of radioactive liquids excluding tritium and dissolved noble gases exceeding 5 curies from the site during a consecutive 3-month period.

5. Discovery of the release of radioactive gases exceeding 50% of the limits specified in Specification 3.9.B.3.

6.12.3 SPECIAL REPORTS

Special reports shall be submitted in writing within 90 days to the Director of Licensing, USAEC, Washington, D. C. 20545.

a. Each containment integrated leak rate test shall be the subject of a summary technical report including results of the local leak rate tests since the last report. The report shall include analyses and interpretations of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

b. A report covering the X-Y xenon stability tests within three months upon completion of the tests.

c. To provide the Commission with added verifications of the safety and reliability of the pre-pressurized Zircaloy-clad nuclear fuel, a limited program of non-destrictive fuel inspections will be conducted at Indian Point Unit No. 2. The program shall consist of a visual inspection (e.g., underwater TV, periscope, or other) of the two lead burnup assemblies in each region during the first, second, and third refueling shutdowns. Any condition observed by this inspection which would lead to unacceptable fuel performance may be the object of an expanded surveillance effort. If another domestic plant which contains pre-pressurized fuel of a similar design reaches fuel exposures equal to or greater than at Indian Point Unit No. 2, and if a limited inspection program is or has been performed there, then the program may not have to be performed at Indian Point Unit No. 2. However, such action requires approval of the Atomic Energy Commission. The results of these inspections will be reported to the Atomic Energy Commission.

6.13 RECORDS RETENTION

6.13.1 All records and logs relative to the following areas shall be retained for at least 5 years:

a. Records of normal nuclear unit operations, including power levels and periods of operation at each power level.

b. Records of principal maintenance activities, including inspection, repair substitution or replacement of principal items of equipment pertaining to nuclear safety.

c. Records of abnormal occurrences.

d. Records of periodic checks, inspections and calibrations performed to verify that surveillance requirements are being met.

e. Records of any special reactor test or experiments.

f. Records of changes made in procedures required by Section 6.8.

g. Records of radioactive shipments.

6.13.2 All records relative to the following areas shall be retained for the life of the plant:

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

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

c. Records of plant radiation and contamination surveys.

d. Records of off-site environmental monitoring surveys.

e. Records of radiation exposures of all station personnel, and others who enter the controlled areas.

f. Records of radioactivity in liquid and gaseous wastes released to the environment.

g. Records of transient or operational cycling for those plant components that have been designed to operate safety for a specific number of transients or operational cycles.

h. Records of individual plant staff members indicating qualifications, training and retraining.

i. In-service inspections of the reactor coolant system.

j. Minutes of meetings of the Station Nuclear Safety Committee and Nuclear Facilities Safety Committee.

k. Records of radioactive releases.

6.14 PLANT SURVEY FOLLOWING AN EARTHQUAKE

Applicability

Applies to the inspecting of plant systems following an earthquake at the Indian Point site.

Objective

To specify procedures that determine whether plant systems are functioning properly after the occurrence of an earthquake.

Specification

If an earthquake shock is felt or reported to occur in the vicinity of the Indian Point Station the company's seismological consultant will be contacted within 24 hours by telephone for an evaluation of the magnitude at the Indian Point Site. The operator, after observing the earthquake, will immediately check the control boards to determine the effect, if any, on instrumentation, controls, and plant operation. An inspection will then be performed of Class II structures and equipment for visual indication of damage. An emergency evaluation will be made if damage exists or if the magnitude of the earthquake at Indian Point as reported by the seismological consultant exceeds the design basis earthquake.

Basis

Even though the Indian Point site is a region of low seismic activity the Indian Point No. 2 plant is designed to withstand seismic activity. Appendix A of the Indian Point 2 FSAR defines this design. All equipment and structures are classified as either Class I, and Class II, or Class III.[1]

Reference

(1) FSAR - Appendix A

A tour of inspection of Class II structures and equipment will be conducted if an earthquake is felt at Indian Point. This inspection will be coincident with a check of control and instrumentation for Class I systems in the control room. Visual signs of damage of Class II equipment and structures and indications from the control room of anomalies in the operation of Class I equipment will be the basis for further plant inspection and, if necessary, corrective action.

6.15 PLANT EMERGENCY PROGRAM
IN THE EVENT OF A TORNADO WATCH OR TORNADO WARNING

Applicability

Applies to plant tornado emergency program.

Objective

To specify plant procedures in the event of a tornado watch[1] or a tornado warning.[2]

Specification

In the event of a tornado watch the emergency foreman will notify the station. Upon notification, personnel will be assigned to listen and look for a tornado.

Upon notification of a tornado warning the gas turbine generator will be started and fuel handling operations in the fuel handling building will be halted. Should a fuel handling cask be suspended from the crane at this time, it will be set down.

Basis

Conditions that can result in a tornado can be determined by the Weather Bureau. Con Edison is in contact with local weather officials and will be advised immediately if tornado watch conditions are in effect. If a tornado is seen or if the plant operator is informed by weather officials that a tornado has occurred in the vicinity of the Indian Point site, non-essential plant operations will be halted.

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[1] Tornado watch - means meteorological conditions are favorable for the formation of a tornado.

[2] Tornado warning - means that a tornado has been sighted in the area of the plant.