UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated December 2, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-26 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes there to through Change No. 10."

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3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Change No. 10
Technical Specifications

Date of Issuance: MAY 9 1975

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1.6.1 Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include comparison of the channel with other independent channels measuring the same variable.

1.6.2 Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.6.3 Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.7 Containment Integrity

Containment integrity is defined to exist when:

a. The required non-automatic containment isolation valves are closed and blind flanges are properly installed.

b. The equipment door is properly closed and sealed by the Weld Channel and Penetration Pressurization System.

c. At least one door in each personnel air lock is properly closed.

d. All automatic containment isolation valves are operable or closed.

e. The containment leakage satisfies Specification 4.4.

1.8 Abnormal Occurrence

An Abnormal Occurrence shall be any of those conditions specified in Revision 3 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications" Sections C.2.a and C.2.b. 10

1.9 Quadrant Power Tilt

The quadrant power tilt is defined as the ratio of maximum to average of the upper excore detector currents or the lower excore detector currents whichever is greater. If one excore detector is out of service, the three in-service units are used in computing the average.

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 Annual Operating Report in accordance with Specification 6.9.1. 10

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 or low steam line pressure.

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 month period, the licensee shall report such as per Section 6.9.2.d.1 of the Technical Specifications. 10

B. Gaseous Effluents

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

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

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

$\overline{E}_{\gamma 1}$ 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).

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

$\overline{E}_{\beta 2}$ 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 16,500 Ci 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 equipment 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 6.9.2.d.2 of the Technical Specifications. 10

Basis

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

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 6.10.2.e. Estimates of the error associated with these analyses should be included. 10

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 6.10.2.e. Estimates of the error associated with these analyses should be included. 10

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 the 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 10 CFR Part 50, 50.36(a)(2) following the format of Appendix A of USAEC Safety Guide 21 of January, 1972. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may time to time require the licensee to take 10

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 10 CFR Part 50, 50.35(a)(2) following the format of Appendix A of USAEC Safety Guide 21 of January, 1972. 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. 10

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

FACILITY MANAGEMENT AND TECHNICAL SUPPORT

6.2.1 The organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1. 10

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling. This individual shall have no other concurrent responsibilities during this operation.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions.

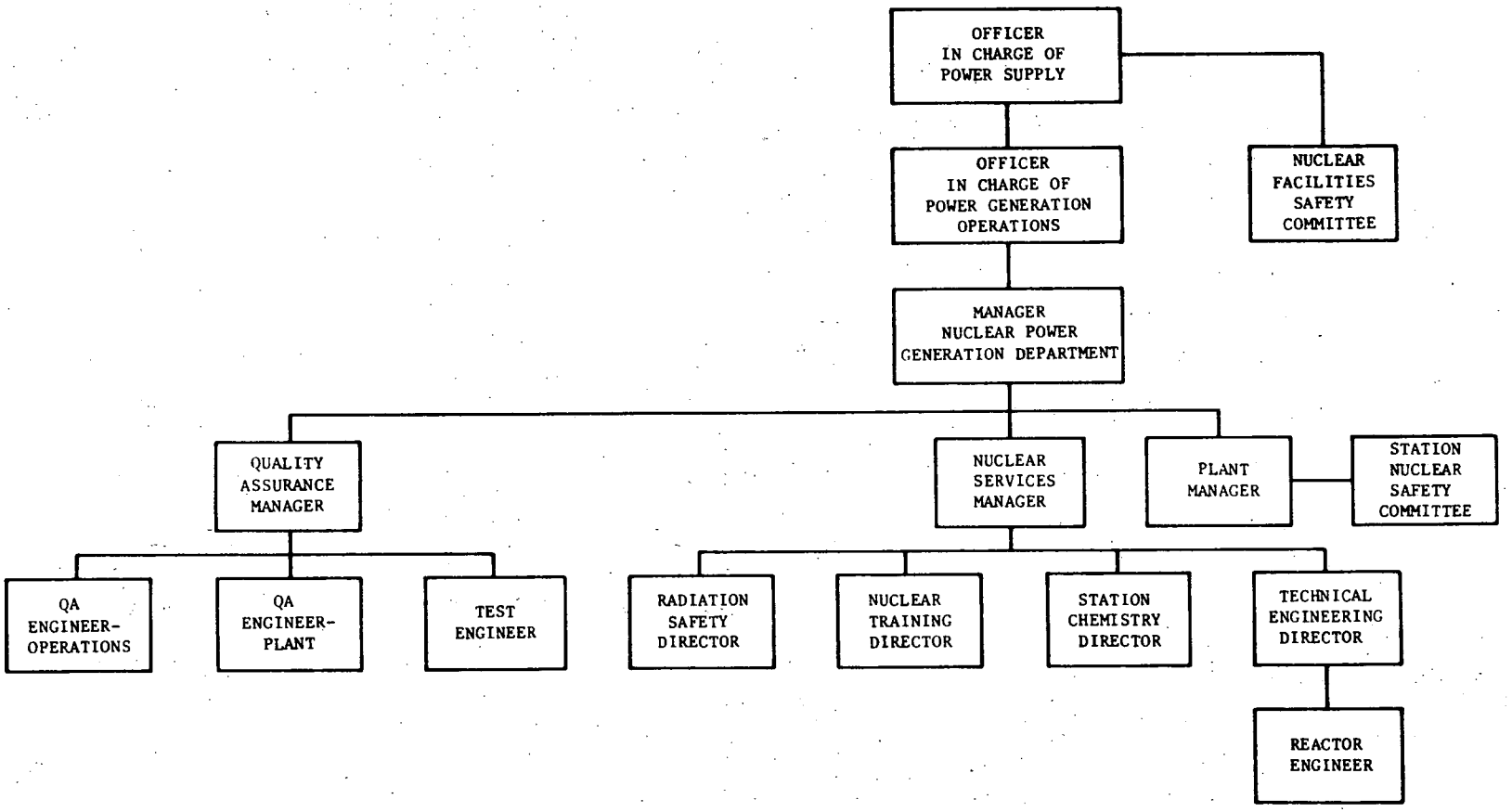

Figure 6.2-1 Facility Management and Technical Support Organization

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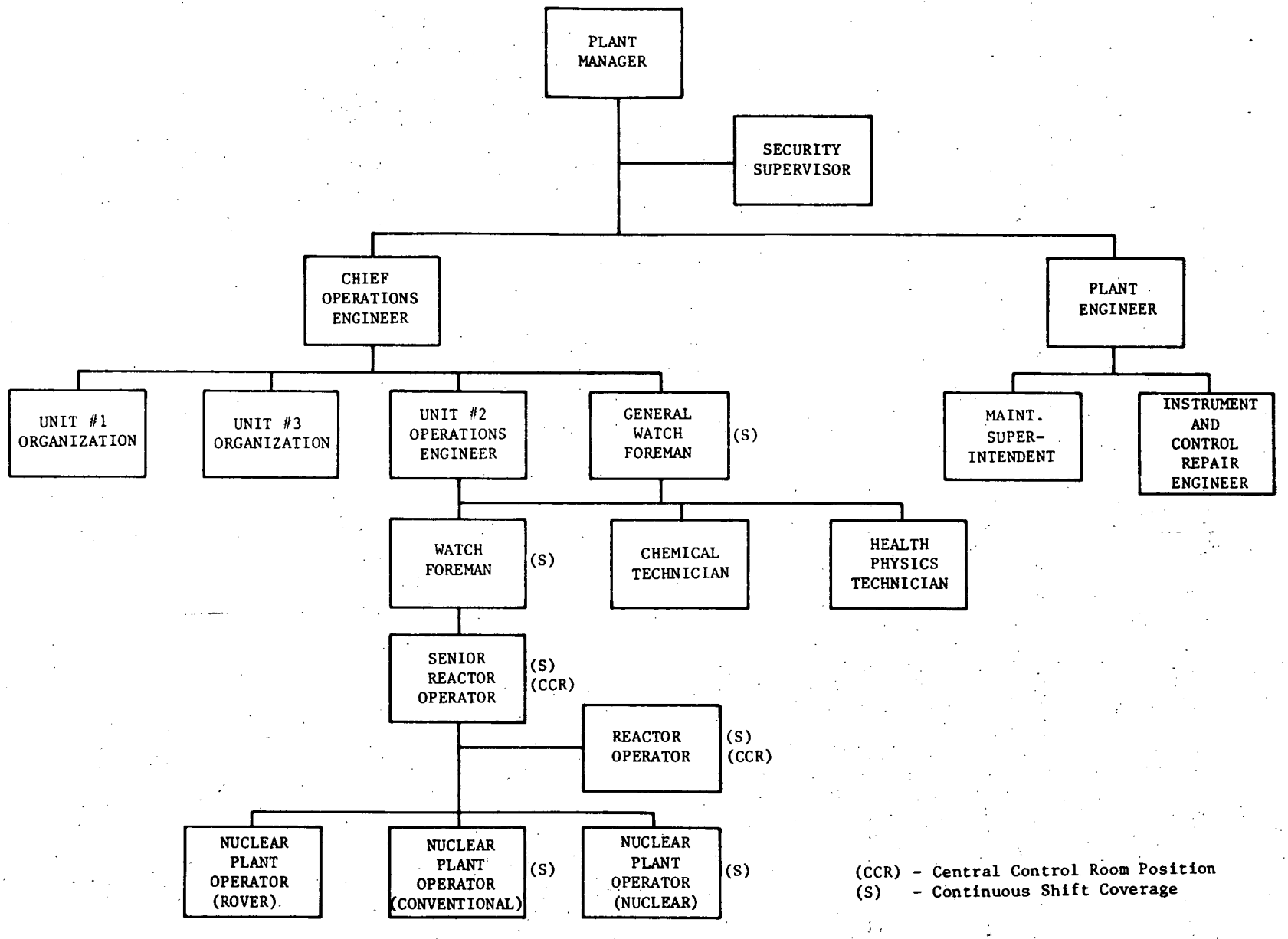

Figure 6.2-2 Facility Organization

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Table 6.2-1

Minimum Shift Crew Composition

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

*Includes individual with SRO license supervising fuel movement as per Section 6.2.2(e).

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6.4 TRAINING

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

6.5 REVIEW AND AUDIT

6.5.1 STATION NUCLEAR SAFETY COMMITTEE (SNSC)

FUNCTION

6.5.1.1 The Station Nuclear Safety Committee shall function to advise the Plant Manager on all matters related to nuclear safety.

6.5.1.2 The Station Nuclear Safety Committee shall be composed of the:

Chairman:	Technical Engineering Director
Member:	Radiation Safety Director
Member:	Operations Engineers
Member:	Reactor Engineers
Member:	Station Chemistry Director
Member:	Maintenance Superintendent
Member:	I & C Repair Engineer

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ALTERNATES

6.5.1.3 Alternate members shall be appointed in writing by the SNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate in SNSC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SNSC shall meet at least once per calendar month and as convened by the SNSC Chairman.

QUORUM

6.5.1.5 A quorum of the SNSC shall consist of the Chairman or Vice Chairman and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Station Nuclear Safety Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, and 2) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to the Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications and preparation and forwarding of a report covering evaluation and recommendations to prevent recurrence via the Plant Manager to the Manager, Nuclear Power Generation Department and to the Chairman of the Nuclear Facilities Safety Committee. 10

f. Review of facility operations to detect potential safety hazards.

g. Performance of special reviews and investigations and the issuance of reports thereon as requested by the Plant Manager or the Chairman of the Nuclear Facilities Safety Committee.

h. Review of the Plant Security Plan and implementing procedures and submission of recommended changes via the Plant Manager to the Chairman of the Nuclear Facilities Safety Committee.

i. Review of the Emergency Plan and implementing procedures and submission of recommended changes via the Plant Manager to the Chairman of the Nuclear Facilities Safety Committee.

AUTHORITY

6.5.1.7 The Station Nuclear Safety Committee shall:

a. Recommend to the Plant Manager, in writing, approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

AUTHORITY (Continued)

c. Provide immediate written notification to the Chairman, Nuclear Facilities Safety Committee and the Manager, Nuclear Power Generation Department of disagreement between the recommendations of the SNSC and the actions contemplated by the Plant Manager. However, the course of action determined by the Plant Manager pursuant to 6.1.1 above shall be followed.

RECORDS

6.5.1.8 The Station Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to, as a minimum, the Plant Manager, the Manager, Nuclear Power Generation Department and the Chairman, Nuclear Facilities Safety Committee.

6.5.2 NUCLEAR FACILITIES SAFETY COMMITTEE (NFSC)

FUNCTION

6.5.2.1 The Nuclear Facilities Safety Committee shall function to provide independent review and audit of designated activities in the areas of: 10

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 practices

i. environmental effects

j. other appropriate fields associated with the unique characteristics of the nuclear power plant

COMPOSITION

6.5.2.2 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 or the President of the Company. In the absence of the Chairman, he will serve as Chairman.

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

Committee members from departments having a direct 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

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

No more than two alternates shall participate in activities at any one time.

Alternate members shall have voting rights.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NFSC Chairman.

MEETING FREQUENCY

6.5.2.5 The NFSC shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of NFSC shall consist of the Chairman or his designated alternate and a majority of the NFSC members including alternates. In the event both the Chairman and the Vice Chairman are absent, one of the permanent voting members will serve as Acting Chairman. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW 10

6.5.2.7 The following subjects shall be reported to and reviewed by the Committee insofar as they relate to matters of nuclear safety:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or expeiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

REVIEW (Continued)

g. ABNORMAL OCCURRENCES, as defined in Section 1.0 of these Technical Specifications.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

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

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the NFSC. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.

b. The performance, training and qualifications of the entire facility staff at least once per year. 10

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two years.

e. The Facility Emergency Plan and implementing procedures at least once per two years.

f. The Facility Security Plan and implementing procedures at least once per two years.

g. Any other area of facility operation considered appropriate by the NFSC or the Senior Company Officer in charge of Power Supply.

AUTHORITY

6.5.2.9 The NFSC shall report to and advise the Senior Company Officer in charge of Power Supply on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NFSC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NFSC meeting shall be prepared, approved and forwarded to the Senior Company Officer in charge of Power Supply within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the Senior Company Officer in charge of Power Supply within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Senior Company Officer in charge of Power Supply and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 ABNORMAL OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of an ABNORMAL OCCURRENCE: 10

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each Abnormal Occurrence Report submitted to the Commission shall be reviewed by the SNSC and submitted to the NFSC Chairman, the Plant Manager and the Manager, Nuclear Power Generation Department.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Commission, the Manager, Nuclear Power Generation Department and to the NFSC Chairman immediately.

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

d. The Safety Limit Violation Report shall be submitted to the Commission, the NFSC Chairman and the Manager, Nuclear Power Generation Department within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.1 and 5.3 of ANSI N18.7-1972 and Appendix "A" of USAEC Regulatory Guide 1.33 except as provided in 6.8.2 and 6.8.3 below.

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6.8.2 Each procedure and administrative policy of 6.8.1 above, and any changes to them shall be reviewed and approved for implementation in accordance with a written administrative control procedure 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, require that:

a. Each proposed procedure/procedure change involving safety related components and/or operation of same receives a pre-implementation review by the SNSC except in case of an emergency.

b. Each proposed procedure/procedure change which renders or may render the Final Safety Analysis Report or subsequent safety analysis reports inaccurate and those which involve or may involve potential unreviewed safety questions are approved by the SNSC prior to implementation.

c. The approval of the Nuclear Facilities Safety Committee shall be sought if, following its review, the Station Nuclear Safety Committee finds that the proposed procedure/procedure change either involves an unreviewed safety question or if it is in doubt as to whether or not an unreviewed safety question is involved.

6.8.3 A mechanism shall exist for making temporary changes and they shall only be made by approved management personnel in accordance with the requirements of ANSI 18.7-1972. The change shall be documented, and reviewed by the SNSC within 7 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE AND ABNORMAL OCCURRENCE REPORTS

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 3 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications".

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of Region 1, Office of Inspection and Inforcement within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. 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-destructive fuel inspections will be conducted. 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 Nuclear Regulatory Commission. The results of these inspection will be reported to the Nuclear Regulatory Commission. 10

d. A written report shall be forwarded within 30 days to the Division of Reactor Licensing and to the Director of the Region 1, Office of Inspection and Enforcement, in the event of:

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

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

6.10 RECORD RETENTION

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

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

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

c. ABNORMAL OCCURRENCE REPORTS.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications. 10

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

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

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

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

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

c. Records of facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles. 10.

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

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

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

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

k. Records of meetings of the SNSC and the NFSC.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 RESPIRATORY PROTECTION PROGRAM

ALLOWANCE

6.12.1 Pursuant to 10 CFR 20.102(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 facility 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:

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

b. 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. 10

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table 1, 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.

PROTECTION PROGRAM

6.12.2 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:

a. The limits specified in 6.12.1 above, are not exceeded.

b. 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 determined by dividing the ambient airborne concentration by the protection factor

specified in Table 6.12-1 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.

c. 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.

d. 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:

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

2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment. 10

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

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

5. 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.

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

e. The licensee shall use equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. 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 6.12-1.

PROTECTION PROGRAM (Continued)

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

REVOCATION

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6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

TABLE 6.12-1

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)

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TABLE 6.12-1 (Continued)

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 circuit)

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:

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

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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 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.

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

TABLE 6.12-1 (Continued)

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. 10

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.

6.13 HIGH RADIATION AREA

6.13.1 As an acceptable alternate to the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. Each High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area. 10

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Watch Foreman on duty.