ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
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TECHNICAL SPECIFICATIONS

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Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
1.	Manual	2	1	1	0	Maintain hot shutdown
2.	Nuclear Flux Power Range	4	2	3	2	Maintain hot shutdown
2.a	Nuclear Flux Power Range	4	2	2	1	For zero power physics tests only
3.	Nuclear Flux Intermediate Range	2	1	1*	0	Maintain hot shutdown
4.	Nuclear Flux Source Range	2	1	1**	0	Maintain hot shutdown
5.	Overtemperature delta T	4	2	3	2	Maintain hot shutdown
6.	Overpower delta T	4	2	3	2	Maintain hot shutdown
7.	Low Pressurizer Pressure	4	2	3	2	Maintain hot shutdown

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
8.	Hi Pressurizer Pressure	3	2	2	1	Maintain hot shutdown
9.	Pressurizer Hi Water Level	3	2	2	1	Maintain hot shutdown
10.	Low Flow Loop ≥ 75% F.P.	3/loop	2/loop (any loop)	2/operable Loop	1/operable loop	Maintain hot shutdown
	Low Flow Two Loops 10-75% F.P.	3/loop	2/loop (any two loops)	2/operable loop	1/operable loop	
11.	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	1/loop	Maintain hot shutdown
12.	Undervoltage 6.9 kV Bus	1/bus	2	3	2	Maintain hot shutdown
13.	Low frequency 6.9 kV Bus	1/bus	2	3	2	Maintain hot shutdown***
14.	Quadrant power tilt monitors	2	NA	1	0	Log individual upper and lower ion chamber cur-rents once/shift and after load change > 10%

Reactor Trip Instrumentation Limiting Operating Conditions

No.	Functional Unit	1 No. of Channels	2 No. of Channels to Trip	3 Min. Operable Channels	4 Min. Degree of Redun-dancy	5 Operator Action if Conditions of Column 3 or 4 Cannot be Met
15.	DELETED					
16.	Control Rod Protection****	3	2	2	1	During RCS cooldown, manually open reactor trip breakers prior to T_{cold} decreasing below 381°F. Maintain reactor trip breakers open during RCS cool-down when T_{cold} is less than 381°F.
17.	Turbine Trip ≥ 35% F.P. A. Low Auto Stop Oil Pressure	3	2	2	1	Maintain reactor power below 35% F.P.
18.	Reactor Trip Logic	2	1	2#	1#	Be in hot shut-down within the next six hours.
18.a	Engineered Safety Features (SI) Logic	2	1	2##	1##	Be in hot shut-down within the next six hours.

Reactor Trip Instrumentation Limiting Operating Conditions

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
19.	Reactor Trip Breakers	2	1	2#	1#	With either diverse trip feature inoperable, or the breaker incapable of tripping for any other reason, restore it to operable conditions or, be in hot shutdown within the next six hours and open both reactor trip breakers. The breaker shall not be bypassed except for the time required for performing maintenance and/or testing to restore it to

operability.

Table 3.5-2

Reactor Trip Instrumentation Limiting Operating Conditions

F.P. = Rated Power

* If two of four power range channels are greater than 10% F.P., channels are not required.

** If one of two intermediate range channels is greater than 10^{-10} amps, channels are not required.

*** 2/4 trips all four reactor coolant pumps.

**** Required only when control rods are positioned in core locations containing LOPAR fuel.

A reactor trip breaker and/or associated logic channel may be bypassed for maintenance or surveillance testing for up to eight hours provided the redundant reactor trip breaker and/or associated logic channel is operable.

An Engineered Safety Feature (SI) logic channel may be bypassed for maintenance or surveillance testing for up to eight hours provided the redundant logic channel is operable.

3.9 RADIOACTIVE EFFLUENTS

Applicability

This specification applies to the control of liquid, gaseous and solid radioactive wastes from the facility.

Objective

To define limits and conditions for the controlled release of radioactive materials to the environs such that these releases are as low as reasonably achievable (ALARA) and within allowable regulatory limits.

Specifications

A. RADIOACTIVE LIQUID EFFLUENTS

1. Liquid Effluent Concentration

   a. The concentration of radioactive material released in liquid effluents to unrestricted areas (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x 10^{-4} microcuries/ml.

   b. With the concentration of radioactive material released in liquid effluents to unrestricted areas exceeding the above limits, without delay restore the concentration to within the the above limits.

2. Liquid Effluent Instrumentation

   a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.9-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.A.1.a are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

   b. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the action shown in Table 3.9-1. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. Liquid Effluent Dose

a. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from each reactor unit to unrestricted areas (see Figure 5.1-1) shall be limited:

(i) during any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

(ii) during any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

b. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the causes(s) for exceeding the limit(s) and defines both the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

4. Liquid Waste Treatment

a. The applicable portions of the liquid radwaste treatment system shall be used as needed to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent, from each reactor unit, to unrestricted areas (see Figure 5.1-1) would

would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in a 31-day period.

b. With radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that includes the following information:

(i) Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment of subsystems, and the reason for the inoperability.

(ii) Action(s) taken to restore the inoperable equipment to operable status, and

(iii) Summary description of action(s) taken to prevent a recurrence.

5. <u>Liquid Holdup Tanks</u>

a. The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Refueling Water Storage Tank
b. Primary Water Storage Tank
c. 13, 14 Waste Distillate Storage Tanks
d. Outside temporary tank

b. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, take action within 48 hours to reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report.

B. RADIOACTIVE GASEOUS EFFLUENTS

1. Gaseous Effluent Dose Rate

a. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary (see Figure 5.1-1) shall be limited to the following:

   (i) for noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

   (ii) for iodine-131, for tritium and for all radionuclides in particulate form with half lives greater than 8 days: less than or equal to 1500 mrem/yr to any organ.

b. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

2. Gaseous Effluent Instrumentation

a. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.9-2 shall be operable with their alarm/trip Setpoints set to ensure that the limits of Specification 3.9.B.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

b. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels operable, take the action shown in Table 3.9-2. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3. Noble Gases

a. The dose due to noble gases released in gaseous effluents from each reactor unit to areas at and beyond the site boundary (see Figure 5.1-1) shall be limited to the following:

(i) during any calendar quarter: less than or equal to 5 mrem to the whole body from gamma radiation and less than or equal to 10 mrem to the skin from beta radiation, and

(ii) during any calendar year: less than or equal to 10 mrem to the whole body from gamma radiation and less than or equal to 20 mrem to the skin from beta radiation.

b. With the calculated dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

4. Radioiodines, Radioactive Material in Particulate Form, and Radionuclides Other Than Noble Gases

a. The dose to a member of the public from iodine-131, tritium, and all radionuclides in particulate form with half-lives of more than 8 days, in gaseous effluents released from each reactor unit to areas at and beyond the site boundary (see Figure 5.1-1) shall be limited to the following:

(i) during any calendar quarter: less than or equal to 7.5 mrem to any organ, and

(ii) during any calendar year: less than or equal to 15 mrem to any organ.

b. With the calculated dose from the release of iodine-131, tritium, and radionuclides in particulate form with half-lives of more than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

5. Gaseous Waste Treatment System

a. The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the site boundary (see Figure 5.1-1) would exceed 0.2 mrem for gamma radiation and 0.4 mrem for beta radiation in a 31-day period. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the site boundary (see Figure 5.1-1) would exceed 0.3 mrem to any organ in a 31-day period.

b. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that includes the following information:

(i) explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

(ii) action(s) taken to restore the inoperable equipment to operable status, and

(iii) summary description of action(s) taken to prevent a recurrence.

6. Explosive Gas Mixture

a. The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

b. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

c. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

7. Waste Gas Decay Tanks

a. The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 29,761 curies of noble gases (considered as Xe-133).

b. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

C. URANIUM FUEL CYCLE DOSE COMMITMENT

1. The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

2. With the calculated doses from the releases of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.9.A.3.a(i), 3.9.A.3.a(ii), 3.9.B.3.a(i), 3.9.B.3.a(ii), 3.9.B.4.a(i), and 3.9.B.4.a(ii), calculations should be made, including direct radiation contributions from the reactor units and from outside storage tanks, to determine whether the above limits of Specification 3.9.C have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a special report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above

limits and includes the schedule for achieving conformance with the above limits. This special report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation dose to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive materials involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition, resulting in violation of 40 CFR Part 190, has not already been corrected, the special report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

D. SOLID RADIOACTIVE WASTE

1. The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

2. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

Basis

It is expected that the release of radioactive materials in liquid and gaseous effluents to unrestricted areas will not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as reasonably achievable (ALARA) in accordance with the requirement of 10 CFR 50.36a. While providing reasonable assurance that the design objectives will be met, these Specifications permit the flexibility of operation, compatible with considerations of health and safety, to ensure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that using this operational flexibility under unusual operation conditions, and exerting every effort to keep levels of radioactive materials in liquid and gaseous wastes as low as reasonably achievable, the annual release will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience, taking into account a combination of variables including defective fuel, primary system leakage, primary to secondary system leakage, steam generator blowdown and the performance of the various waste treatment systems, and are consistent with 10 CFR Part 50.36a.

The Indian Point site is a multiple-unit site. There exist shared radwaste treatment systems and shared effluent release points. Where site limits must be met, the effluents of all the units will be combined to determine site compliance. For instances where unit-specific information may be required for radwaste processed or released via a shared system, the effluents shall be proportioned among the units sharing the system(s) in accordance with the methods and agreements set forth in the ODCM.

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a member of the public and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-133 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of liquid effluents from Indian Point Units Nos. 1 and 2.

The radioactive liquid effluent instrumentation, required operable by Specification 3.9.A.2, is provided to monitor and control, as applicable, the
releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with methods set forth in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Table II of Appendix B to 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50. The purpose of tank level indicating devices is to assure the detection and control of leaks that, if not controlled, could potentially result in the transport of radioactive materials to unrestricted areas.

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as reasonably achievable". Also, for fresh water sites to unrestricted area with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentration in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", April 1977.

In addition to the limiting conditions for operation listed under Specification 3.9.A.3.a, the reporting requirements of Specification 3.9.A.3.b specify that the licensee shall identify the cause whenever the dose from the release of radioactive materials in liquid waste effluent exceeds the technical specification limits and describe the proposed program of action to reduce such releases to design objective levels on a timely basis.

Specification 3.9.A.4 requires that the licensee maintain and operate appropriate equipment installed in the liquid waste systems, when necessary, to provide assurance that the releases of radioactive materials in liquid effluents will be kept "as low as reasonably achievable". This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I to 10 CFR Part 50 for liquid effluents.

The tanks listed in Specification 3.9.A.5 include outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that, in the event of an uncontrolled release of any such tank's contents, the resulting concentration would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

Specification 3.9.B.1 is provided to ensure that the dose at any time beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area to annual average concentration exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For members of the public, who may at times be within the site boundary, the occupancy of that member of the public will usually be sufficiently low to compensate for any increase in the atmosphere diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background via the inhalation pathway to less than or equal to 1500 mrem/year.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

The radioactive gaseous effluent instrumentation, required operable by Specification 3.9.B.2, is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design criteria 60, 63 and 64 in Appendix A to 10 CFR Part 50.

Specification 3.9.B.3 is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I to 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The action statements provide the required operating flexibility and, at the same time, implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases form Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

Specification 3.9.B.4 is provided to implement the requirements of Section II.C, III.A and IV.A of Appendix I to 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The action statements provide the required operating flexibility and, at the same time, implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to unrestricted areas will be kept "as low as reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milch animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

Specification 3.9.B.5 requires that the appropriate portions of these systems be used, when specified, to provide reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of gaseous effluents from Indian Point Units Nos. 1 and 2.

Specification 3.9.B.6 is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

The tanks included in Specification 3.9.B.7 are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another technical specification to a quantity that is less than the quantity that provides assurance that, in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem in an event of 2 hours duration.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurances that, in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to a member of the public at the nearest site boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

Specification 3.9.C is provided to meet the dose limitation of 40 CFR Part 190 that has been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a special report whenever the calculated doses from plant-generated radioactive effluents and direct radiation exceed 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The special report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR Part 190 limits. For the purposes of the special report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contribution from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR Part 190, the special report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.9.A.1 and 3.9.B.1. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

Specification 3.9.D implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the process control program may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constitutents, and mixing and curing times.

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M (3)*1	Q (2)	1) Heat balance calibration 2) Signal to delta T; bistable action (permissive, rod stop, trips) 3) Upper and lower chambers for axial offset.
2.	Nuclear Intermediate Range	S (1)	N.A.	S/U (2)*2	1) Once/shift when in service Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A.	S/U (2)*2	1) Once/shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R#	Q (1)	1) Overtemperature - delta T 2) Overpower - delta T
5.	Reactor Coolant Flow	S	R#	Q	
6.	Pressurizer Water Level	S	R#	Q	
7.	Pressurizer Pressure (High & Low)	S	R#	Q	
8.	6.9 kV Voltage & Frequency	N.A.	R#	Q	Reactor Protection circuits only
9.	Analog Rod Position	S	R#	M	
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R#	Q	
12.	Charging Flow	N.A.	R#	N.A.	

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
13.	Residual Heat Removal Pump Flow	N.A.	R#	N.A.	
14.	Boric Acid Tank Level	W	R#	N.A.	Bubbler tube rodded during calibration
15.	Refueling Water Storage Tank Level	W	Q	N.A.	
16.	DELETED				
17.	Volume Control Tank Level	N.A.	R#	N.A.	
18a.	Containment Pressure	D	R#	Q	Wide Range
18b.	Containment Pressure	S	R#	Q	Narrow Range
18c.	Containment Pressure (PT-3300,PT-3301)	M	R#	N.A.	High Range
19.	Process Radiation Monitoring System	D	R#	M	
19a.	Area Radiation Monitoring System	D	R#	M	
19b.	Area Radiation Monitoring System (VC)	D	R#	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R#	N.A.	

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

Channel Description		Check	Calibrate	Test	Remarks
21a.	Containment Sump and Recirculation Sump Level (Discrete)	S	R#	R#	Discrete Level Indication Systems.
21b.	Containment Sump, Recirculation Sump and Reactor Cavity Level (Continuous)	S	R#	R#	Continuous Level Indication Systems.
21c.	Reactor Cavity Level Alarm	N.A.	R#	R#	Level Alarm System
21d.	Containment Sump Discharge Flow	S	R#	M	Flow Monitor
21e.	Containment Fan Cooler Condensate Flow	S	R#	M[*3]	
22a.	Accumulator Level	S	R#	N.A.	
22b.	Accumulator Pressure	S	R#	N.A.	
23.	Steam Line Pressure	S	R#	Q	
24.	Turbine First Stage Pressure	S	R#	Q	
25.	Reactor Trip Logic Channel Testing	N.A.	N.A.	M[*9]	
26.	Engineered Safety Features (SI) Logic Channel Testing	N.A.	N.A.	M[*9]	
27.	Turbine Trip				
	a. Low Auto Stop Oil Pressure	N.A.	R#	N.A.	

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
28.	Control Rod Protection (for use with LOPAR fuel)	N.A.	R#	*4	
29.	Loss of Power				
	a. 480v Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R#	R#	
	b. 480v Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R#	R#	
	c. 480v Emergency Bus Undervoltage (Alarm)	N.A.	R#	M	
30.	Auxiliary Feedwater				
	a. Steam Generator Water Level (Low-Low)	S	R#	R#	
	b. Low-Low Level AFWS Automatic Actuation Logic	N.A.	N.A.	M	Test one logic channel per month on an alternating basis.
	c. Station Blackout (Undervoltage)	N.A.	R#	R#	
	d. Trip of Main Feedwater Pumps	N.A.	N.A.	R#	
31.	Reactor Coolant System Subcooling Margin Monitor	M	R#	N.A.	
32.	PORV Position Indicator (Limit Switch)	M	R#	R#	

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
33.	PORV Block Valve Position Indicator (Limit Switch)	M[*5]	R#	R#	
34.	Safety Valve Position Indicator (Acoustic Monitor)	M	R#	R#	
35.	Auxiliary Feedwater Flow Rate	M	R#	R#	
36.	PORV Actuation/ Reclosure Setpoints	N.A.	R#	N.A.	
37.	Overpressure Protection System (OPS)	N.A.	R#	*6	
38.	Wide Range Plant Vent Noble Gas Effluent Monitor (R-27)	S	R#	N.A.	
39.	Main Steam Line Radiation Monitor (R-28, R-29, R-30, R-31)	S	R#	N.A.	
40.	High Range Containment Radiation Monitor (R-25, R-26)	S	R#[*7]	N.A.	
41.	Containment Hydrogen Monitor	Q	Q[*8]	N.A.	

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
42.	Manual Reactor Trip	N.A.	N.A.	R#	Includes: 1) Independent verification of reactor trip and bypass breakers undervoltage trip circuit operability up to and including matrix contacts of RT-11/RT-12 from both manual trip initiating devices, 2) independent verification of reactor trip and bypass breaker shunt trip circuit operability through trip actuating devices from both manual trip initiating devices.
43.	Reactor Trip Breaker	N.A.	N.A.	M[*9]	Includes independent verification of undervoltage and shunt trip attachment operability.
44.	Reactor Trip Bypass Breaker	N.A.	N.A.	M[*9]	Includes: 1) Automatic undervoltage trip, 2) Manual shunt trip from either the logic test panel or locally at the switchgear prior to placing breaker into service.
45.	Service Water Inlet Temperature Monitoring Instrumentation	S	R#	A	The test shall take place prior to T.S. 3.3.F.b Applicability.

Minimum Frequencies for Checks, Calibrations and
Tests of Instrument Channels

Footnotes:

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*1 By means of the movable incore detector system.

*2 Prior to each reactor startup if not done previous week.

*3 Monthly visual inspection of condensate weirs only.

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

*5 Except when block valve operator is deenergized.

*6 Within 31 days prior to entering a condition in which OPS is required to be operable and at monthly intervals thereafter when OPS is required to be operable.

*7 Acceptable criteria for calibration are provided in Table II.F-13 of NUREG-0737.

*8 Calibration will be performed using calibration span gas.

*9 Each train shall be tested at least every 62 days on a staggered test basis (i.e., one train per month).

4.2 INSERVICE INSPECTION AND TESTING

Applicability

Applies to the inservice inspection of Quality Group* A, B, and C components and the inservice testing of pumps and valves whose function is required for safety.

Objective

To provide assurance of the continued integrity and/or operability of those structures, systems, and components to which this specification is applicable.

Specifications

4.2.1 Inservice Testing

Inservice testing of pumps and valves whose function is required for safety shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50, Section 50.55a(g), except where specific written relief pursuant to 10 CFR 50, Section 50.55a has been granted.

4.2.2 Inservice Inspection

Inservice inspection of Quality Group* (* Quality Group classification is in accordance with Revision 3 of Regulatory Guide 1.26.) A, B, and C components shall be performed in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as Required by 10 CFR 50, Section 50.55a(g), except where specific written relief pursuant to 10 CFR 50, Section 50.55a has been granted.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling. At each subsequent refueling, one different flywheel shall be examined by ultrasonic methods. The examinations schedules are shown in Table 4.2-1.

4.2.4 Reactor Vessel Special Inspection

1. Interval of Inspection:

   The reactor vessel shall be examined during the second ten year interval in the area of the vessel weld located approximately 236 inches below the reactor vessel flange at 345° azimuth. This area shall be re-examined during the three successive inspection periods as defined in accordance with IWB-2410 of the 1980 ASME Boiler and Pressure Vessel Code, Section XI, as modified below.

The examination schedule may revert to the original inspection schedule per IWB-2410 if:

(i) The additional examinations reveal that the indications remain essentially unchanged over 3 successive inspections, or

(ii) Any additional examination utilizing-ultrasonic techniques per IWA-2232, or alternative techniques per IWA-2240, as supplemented by prior examination, demonstrate that the reflector meets the acceptance standards of IWB-3510. Such demonstration shall be submitted for NRC review and approval. Upon receipt of NRC concurrence, this special inspection requirement (4.2.4 in its entirety) shall become void.

2. Reporting Requirements:

   The reactor vessel inservice inspection program shall be forwarded to NRC 180 days prior to plant shutdown during which the inspection is scheduled to be accomplished. Inspection results shall be forwarded for NRC review and approval 15 days prior to plant startup.

References

(1) Letter from Robert W. Reid of NRC to William J. Cahill of Consolidated Edison dated April 22, 1976

(2) Letter from Robert W. Reid of NRC to William J. Cahill of Consolidated Edison dated November 17, 1976

(3) Letter from William J. Cahill of Consolidated Edison to Robert W. Reid of NRC dated May 27, 1976

4.11 RADIOLOGICAL ENVIRONMENTAL MONITORING

Applicability

Applies to routine testing of the radioactivity in the plant environs and is applicable to the entire Indian Point site.

Objective

The overall objectives of the radiological environmental monitoring program are:

(1) to establish a sampling schedule for the entire Indian Point site, which will recognize changes in radioactivity in the environs of the plants,

(2) to assure that the effluent releases are kept as low as reasonably achievable (ALARA) and within allowable limits in accordance with 10 CFR 20, and

(3) to verify projected and anticipated radioactivity concentrations in the environment and related exposures from releases of radioactive materials from Indian Point Units 1, 2 and 3.

Specifications

A. MONITORING PROGRAM

1. As a minimum, radiological environmental monitoring samples shall be collected pursuant to Table 4.11-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 4.11-1 and the detection capabilities required by Table 4.11-3.

2. With the radiological environmental monitoring program not being conducted as specified in Table 4.11-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.5, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

3. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 4.11-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to

Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a member of the public is less than the calendar year limits of Specifications 3.9.A.3.a, 3.9.B.3.a, and 3.9.B.4.a. When more than one of the radionuclides in Table 4.11-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 4.11-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* (* The methodology and parameters used to estimate the potential annual dose to a member of the public shall be indicated in this report.) to a member of the public is equal to or greater than the calendar year limits of Specifications 3.9.A.3.a, 3.9.B.3.a and 3.9.B.4.a. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

4. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 4.11-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.6, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

B. LAND USE CENSUS

1. A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location, in each of the 16 meteorological sectors, of the nearest milk animal, the nearest residence and the nearest garden* (* Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.11-1.4c shall be followed, including analysis of

control samples.) of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the locations, in each of the 16 meteorological sectors, of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.)

2. The land use census shall be conducted during the growing seasons at least once per calendar year using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

3. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.10.B.4, in lieu of a Licensee Event Report, identify the new location(s) in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.6.

4. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) a factor of 2 greater than at a location from which samples are currently being obtained in accordance with Specification 4.11.A, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.6, identify the new location(s) in the next Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

C. INTERLABORATORY COMPARISON PROGRAM

1. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

2. With analyses not being performed as required in Specification 4.11.C.1 above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to

Specification 6.9.1.5.

3. The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

D. ROUTINE REPORTING REQUIREMENTS

1. A summary of the results of the monitoring program in Specification 4.11.A, the results the land use census in Specification in 4.11.B, and the results of analysis performed as part of an Interlaboratory Comparison Program in Specification in 4.11.C shall all be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5.

Basis

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. Program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.11-3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Specification 4.11.B is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	40 routine monitoring stations (DR1-DR40) either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, one in each meteorological sector in the general area of the site boundary (DR1-DR16); an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site (DR17-DR32); the balance of the stations (DR33-DR40) to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations (A1-A5): 3 samples (A1-A3) from close to the 3 site boundary locations, in different sectors, of the highest calculated annual average groundlevel D/Q. 1 sample (A4) from the vicinity of a community having the highest calculated annual average ground-level D/Q. 1 sample (A5) from a control location, as for example 15-30 km distant and in the least prevalent wind direction.[c]	Continuous sampler operation with sample collection weekly, or more frequently by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radio-Activity analysis following filter change,[d] Gamma isotopic analysis[e] of composite (by location) quarterly.
3. WATERBORNE			
a. Surface[f]	1 sample upstream (Wa1) 1 sample downstream (Wa2)	Composite sample 1-month period[g]	Gamma isotopic analysis[e] monthly Composite for tritium analysis quarterly.

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Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
b. Drinking	1 sample (Wb1) of the nearest surface drinking water supply	Grab Sample - monthly	Gross beta and gamma isotopic analysis[e] monthly. Composite for tritium analysis quarterly.
c. Sediment from Shoreline	2 samples (Wc1-Wc2) 1 sample (Wc1) from downstream area with existing or potential recreational value. 1 control sample (Wc2) from an upstream area.	2 annually at least 90 days apart	Gamma isotopic analysis[e] semiannually.
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations (Ia1-Ia3) within 5 km distance having the highest dose potential. If there are none, then 1 sample from milking animals in each of 3 areas (Ia1-Ia3) between 5 to 8 km distance, if available, where doses are calculated to be greater than 1 mrem per yr[h]	Semimonthly when animals are on pasture; monthly at other times.	Gamma isotopic[e] and I-131 analysis semimonthly when animals are on pasture; monthly at other times.
	1 sample from milking animals at a control location (Ia4), 15-30 km distant and in the least prevalent wind direction.	Concurrently with indicator locations	

Radiological Environmental Monitoring Program

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
b. Fish and Invertebrates	1 sample of each of 2 commercially and/or recreationally important species in vicinity of plant discharge area (Ib1).	Sample in season, or semiannually if they are not seasonal	Gamma isotopic analysis[e] on edible portions.
	1 sample of same species, if available in areas not influenced by plant discharge (Ib2).		
c. Food Products	Samples of 3 different kinds of broad leaf vegetation (edible or inedible) grown nearest each of two different offsite locations of highest predicted annual average groundlevel D/Q if milk sampling is not performed (Ic1-Ic2).	Monthly when available	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed (Ic3).	Monthly when available	Gamma isotopic[e] and I-131 analysis.

Radiological Environmental Monitoring Program

Table Notation

a The code letters in parenthesis, e.g. DR1, A1 define generic sample locations. Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 4.11-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.5. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.6, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Annual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

b One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation.

c The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

Radiological Environmental Monitoring Program

Table Notation (continued)

d Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the previous calendar year mean of control samples, gamma isotopic analyses shall be performed on the individual samples.

e Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

f "Upstream" samples shall be taken near the intake structures as described in the ODCM. "Downstream" samples shall be taken from the mixing zone at the diffuser of the discharge canal.

g A composite sample is one in which the quantity (aliquot) of liquid sample shall be collected at time intervals that are very short (e.g. hourly) relative to the compositing period (e.g. monthly) in order to assure obtaining a representative sample.

h The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Vice President-Nuclear Power shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

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

6.2 ORGANIZATION

6.2.1 Facility Management and Technical Support

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Updated FSAR.

b. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President-Nuclear Power shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 Facility Staff

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

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 startup, scheduled reactor shutdown, and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be onsite 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.

f. DELETED

g. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel).

   The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

h. The Operations Manager shall hold a senior reactor operator license.

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, except for the Radiation Protection Manager who shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, September 1975.

6.3.2 The Plant Manager shall meet or exceed the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

6.3.3 The Watch Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

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

6.4.2 DELETED

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 Vice President-Nuclear Power on all matters related to nuclear safety.

Composition

6.5.1.2 The Station Nuclear Safety Committee shall, as a minimum, be composed of individuals, approved by the Vice President - Nuclear Power, in the following disciplines:

Chairman:	Senior Manager *
Member:	Engineering
Member:	Operations
Member:	Maintenance
Member:	Instrument and Control
Member:	Radiation Protection
Member:	Reactor Engineering

* This Senior Manager shall be appointed by and report directly to the Vice President - Nuclear Power for the SNSC function and shall be independent of the Plant Manager.

6.5.1.2.1 The committee members and alternates shall have an academic degree in engineering or a physical science, or hold a management position, and shall have a minimum of five years technical experience in one or more areas listed in 6.5.1.2. In addition, other qualified individuals meeting these requirements may be appointed by the SNSC Chairman to serve as SNSC members.

Alternates

6.5.1.3 Alternate members shall be appointed in writing by the SNSC Chairman to serve on a temporary basis, and must have qualifications similar to the member being replaced.

Meeting Frequency

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

Quorum

6.5.1.5 A quorum of the SNSC shall consist of the Chairman or his designated alternate and four members. No more than two alternate members shall be included in the quorum.

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 Chairman of SNSC 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 to the Vice President-Nuclear Power and to the Chairman of the Nuclear Facilities Safety Committee,

f. review of facility operations to detect potential nuclear safety hazards,

g. performance of special reviews and investigations and the issuance of reports thereon as required by the Chairman of the Nuclear Facilities Safety Committee,

h. review of any unplanned, radioactive release, including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Power and to the Nuclear Facility Safety Committee, and

i. review of changes to the Process Control Program and the Offsite Dose Calculation Manual,

j. review of the Fire Protection Program and implementing procedures and submission of recommended changes to the Chairman of the Nuclear Facilities Safety Committee.

Authority

6.5.1.7 The Station Nuclear Safety Committee shall:

a. recommend to the Vice President-Nuclear Power, in writing, approval or disapproval of items considered under Specifications 6.5.1.6(a) through (d) above,

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

c. provide immediate written notification to the Chairman, Nuclear Facilities Safety Committee of disagreement between the recommendations of the SNSC and the actions contemplated onsite. However, the course of action determined by the Vice President-Nuclear Power pursuant to Specification 6.1.1 above or the Plant Manager pursuant to Specification 6.1.2 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 Vice President-Nuclear Power 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:

a. reactor operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy and non-destructive testing
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices
i. radiological 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 voting membership of at least 5 persons of which a majority are independent of the Nuclear Power organization 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 Senior Vice President, Central Operations.

The Vice Chairman shall be appointed by the Senior Vice President, Central Operations. 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 interest in nuclear plant design, construction and operation or in nuclear safety shall be designated 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, subject to the Chairman's approval, 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 as voting members in NFSC 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 or at more frequent intervals at the call of the Chairman or, in his absence, the Vice Chairman.

Quorum

6.5.2.6 A majority of the permanent voting committee members, or duly appointed alternates, which shall include the Chairman or the Vice Chairman and of which a minority are from the Nuclear Power Organization shall constitute a quorum for meetings of the Committee. In the event both the Chairman and the Vice Chairman are absent, one of the permanent voting members will serve as Acting Chairman.

Review

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 10 CFR 50.59 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 10 CFR 50.59.

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

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.

g. Reportable Events, as specified by 10 CFR 50.73.

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.

j. Environmental surveillance program pertaining to radiological matters.

<u>Audits</u>

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 provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

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 6 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 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 12 months.

f. The Facility Security Plan and implementing procedures at least once per 12 months.

g. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

h. A fire protection and loss prevention inspection and audit shall be performed utilizing either qualified offsite licensee personnel or an outside fire protection firm at least once per 12 months.

i. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at least once per 36 months.

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The Offsite Dose Calculations Manual and implementing procedures at least once per 24 months.

l. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

m. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

n. Any other area of facility operation considered appropriate by the NFSC or the Senior Vice President, Central Operations.

Authority

6.5.2.9 The NFSC shall report to and advise the Senior Vice President, Central Operations on those areas of responsibility in Specifications 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 Vice President, Central Operations and to Senior Company Officers concerned with nuclear facilities within 14 days following each meeting.

b. Reports of reviews encompassed by Specifications 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the Senior Vice President, Central Operations and to Senior Company Officers concerned with nuclear facilities within 30 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the Senior Company Officers concerned with nuclear facilities and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.6 REPORTABLE EVENT ACTION

6.6.0 A Reportable Event is defined as any of the conditions specified in 10 CFR 50.73a(2).

6.6.1 The following actions shall be taken in the event of a Reportable Event:

a. A report shall be submitted to the Commission pursuant to the requirements of 10 CFR 50.73.

b. Each Licensee Event Report submitted to the Commission shall be submitted to the NFSC Chairman and the Vice President-Nuclear Power and be reviewed by the SNSC.

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 Report shall be reported to the Commission, the Vice President-Nuclear Power and to the NFSC Chairman immediately.

c. The 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 Vice President-Nuclear Power within 10 days of the violation.

6.8 PROCEDURES AND PROGRAMS

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

a. 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 (issued November 1972) except as provided in 6.8.2 and 6.8.3 below.

b. Process Control Program implementation.

c. Offsite Dose Calculation Manual implementation.

d. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 1.21, Revision 1, April 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

e. Fire Protection Program implementation.

6.8.2 Each procedure and administrative policy of Specification 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 Plant Manager, or appropriate Department Manager, with the concurrence of the Station Nuclear Safety Committee and the Vice President, Nuclear Power. 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 Updated 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 and approved by the Plant Manager, or appropriate Department Manager, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. A program which will ensure the capability to obtain and analyze samples of reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere under accident conditions. The program shall include the following:

(i) training of personnel,

(ii) procedures for sampling and analysis, and

(iii) provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

Routine Reports and Reportable Occurrences

6.9.1. In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator, Region I unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) amendments to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the appropriate tests identified in the UFSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.2 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL RADIATION EXPOSURE REPORT[1]

6.9.1.3 Routine reports of occupational radiation exposure data during the previous calendar year shall be submitted no later than April 30 of each year.

6.9.1.4 The annual radiation exposure reports shall provide a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions[2], e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter TLD or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3]

6.9.1.5 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The report shall also include the results of land use censuses required by Specification 4.11.B.

The Annual Radiological Environmental Operating Reports shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements as described in the ODCM. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps[4] covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 4.11.C; discussion and all deviations from the sampling schedule of Table 4.11-1; and discussion of all analyses in which the LLD required by Table 4.11-3 was not achievable.

RADIOACTIVE EFFLUENT REPORT[5]

6.9.1.6 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 12 months of operation shall be submitted by May 1 of each year.

The Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting

Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted by May 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability[6].

This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents releases from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Radioactive Effluent Release Report shall include the following information for each class of solid waste (in compliance with 10 CFR Part 61) shipped offsite during the report period:

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate),

c. principal radionuclides (specify whether determined by measurement or estimate),

d. source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to Unrestricted Areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Report shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 4.11.B.

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or pressurizer safety valves shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT (COLR)

6.9.1.8 Core operating limits shall be established and documented prior to each reload cycle, or prior to any remaining portion of the cycle, for the following:

a. Axial Flux Difference limits for Specifications 3.10.2.

b. Height Dependent Heat Flux Hot Channel Factor for Specification 3.10.2.

c. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.10.2.

d. Shutdown Bank Insertion Limit for Specification 3.10.4.

e. Control Bank Insertion Limits for Specification 3.10.4.

6.9.1.9 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

c. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

d. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

e. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.10.2 Height Dependent Heat Flux Hot Channel Factor.)

f. WCAP-12945-P, Westinghouse "Code Qualification Document for Best Estimate LOCA Analyses", July, 1996

6.9.1.10 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

6.9.1.11 The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

Special Reports

6.9.2 Special reports shall be submitted to the NRC Regional Administrator of the Region I Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. DELETED

b. DELETED

c. Sealed source leakage in excess of limits (Specification 4.15).

d. The complete results of the steam generator tube inservice inspection (Specification 4.13.C.).

e. Radioactive effluents (Specification 3.9).

f. Radiological environmental monitoring (Specification 4.11).

g. Meteorological monitoring instrumentation (Specification 3.15).

h. Inoperable radiation and hydrogen monitoring instrumentation (Specification 3.5) outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to operable status.

i. Operation of overpressure protection system (Specification 3.1.A.4).

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 intervals 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. Reportable Event Reports.

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

Technical Specifications.

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 on 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 Updated 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 releases to the environs.

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

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

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual except as noted in 6.10.1.

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.

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

m. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of the service lives of all snubbers addressed by Section 3.12 of the Technical Specifications, including the date at which the service life commences and associated installation and maintenance records.*

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 HIGH RADIATION AREA

6.12.1 As an acceptable alternative to the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2):

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.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of Specification 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry to such areas and the keys shall be maintained under the administrative control of the Radiation Protection Manager and/or the Senior Watch Supervisor on duty.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-26 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines of NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

    a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information,

    b. a determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes, and

    c. documentation of the fact that the change has been reviewed and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 The ODCM shall be approved by the Commission prior to implementation.

6.15.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluation justifying the change(s),

   b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations, and

   c. documentation of the fact the change has been revised and found acceptable by the SNSC.

2. Shall become effective upon review and acceptance by the SNSC.

6.16 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE SYSTEMS

6.16.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change was made. The discussion of each change shall contain:

   a. a summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59,

   b. sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information,

c. a detailed description of the equipment, components and processes involved and the interfaces with other plant systems,

d. an evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto,

e. an evaluation of the change, which shows the expected maximum exposures to individuals in the Unrestricted Area and to the general population that differ from those previously estimated in the license application and amendments thereto,

f. a comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes are to be made;

g. an estimate of the exposure to plant operating personnel as a result of the change, and

h. documentation of the fact that the change was reviewed and found acceptable by the SNSC.

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* The documentation referred to herein is required for all snubbers beginning with those replaced following the issuance of Amendment 112.

1 A single submittal may be made for a multiple-unit station. The submittal should combine those sections that are common to all units at the station.

2 This tabulation supplements the requirements of 10 CFR Part 20.407.

3 A single submittal may be made for a multiple unit station.

4 One map shall cover stations near the site boundary; a second shall include more distant stations.

5 A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

6 In lieu of submission with the Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data onsite in a file that shall be provided to the NRC upon request.

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	2***
Operator License	1	1	2
Non-Licensed	(As Required)	1	2
Watch Engineer	1	(None Required)	1

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

** Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

*** When the unit is in an operational mode other than cold shutdown or refueling, at least one Licensed Senior Reactor Operator must be in the Control Room at all times.