A. Alan Blind
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February 14, 2000

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555-0001

Subject: Proposed Technical Specification Amendment Consisting of Administrative Changes

Transmitted herewith is an "Application for Amendment to the Operating License," sworn on February 14, 2000. This application requests an amendment to the Consolidated Edison Company of New York, Inc. (Con Edison), Indian Point Unit No. 2 Technical Specifications. In accordance with 10 CFR 50.91, a copy of this application and the associated attachments are being submitted to the designated New York State official.

The proposed administrative changes consist of the following:

a) Changes to Tables 3.6-1 and 4.4-1 to correct listing and editorial errors,

b) Change to Section 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv),

c) Deletion of Figures 3.10-2 through 3.10-6,

d) Change to Table 4.1-1 to reflect change in level indication components,

e) Change to Sections 4.19.B and 6.14.1.1 to correct editorial errors,

f) Change to Section 6.12.1 to reference the current sections of 10 CFR 20,

g) Change to Section 6.12.1 to reflect an organizational title change, and

h) Change to Section 6.13.2 to correct a typographical error.

Attachment I to this letter provides the proposed changed pages, Attachment II provides the proposed changes as markups on the existing Technical Specification pages, and Attachment III provides the Safety Assessments. It has been determined that the administrative changes set forth herein do not represent a significant hazards consideration as defined by 10 CFR 50.92(c).

In addition, Attachment IV provides a revised page to Indian Point 2 Technical Specification Basis 3.3. The description of the change and justification as to why it is not an unreviewed safety question pursuant to 10 CFR 50.59 is provided in Attachment V.

A001

Should you or your staff have any questions regarding this submittal, please contact Mr. John F. McCann, Manager, Nuclear Safety and Licensing.

Very truly yours,

Robert E. Mann
for A Blind

Attachments

cc: Mr. Hubert J. Miller
Regional Administrator-Region I
US Nuclear Regulatory Commission
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
CONSOLIDATED EDISON COMPANY	)	Docket No. 50-247
OF NEW YORK, INC.	)	
(Indian Point Station, Unit No. 2)	)	

APPLICATION FOR AMENDMENT
TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission ("NRC"), Consolidated Edison Company of New York, Inc. ("Con Edison"), as holder of Facility Operating License No. DPR-26, hereby applies for amendment of the Technical Specifications contained in Appendix A of this license.

This Application for amendment to the Indian Point 2 Technical Specifications seeks to propose administrative changes to the following:

a) Changes to Tables 3.6-1 and 4.4-1 to correct listing and editorial errors,
b) Change to Section 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv),
c) Deletion of Figures 3.10-2 through 3.10-6,
d) Change to Table 4.1-1 to reflect change in level indication components,
e) Change to Sections 4.19.B and 6.14.1.1 to correct editorial errors,
f) Change to Section 6.12.1 to reference the current sections of 10 CFR 20,
g) Change to Section 6.12.1 to reflect an organizational title change, and
h) Change to Section 6.13.2 to correct a typographical error.

The specific proposed Technical Specification Revisions are set forth in Attachment I to this Application. A mark-up of the existing Technical Specifications are provided in Attachment II. Safety Assessments of the proposed changes are set forth in Attachment III to this Application. These assessments demonstrate that the proposed changes do not represent a significant hazards consideration as defined in 10 CFR 50.92(c).

As required by 10 CFR 50.91(b)(1), a copy of this Application and our analysis concluding that the proposed changes do not constitute a significant hazards consideration have been provided to the appropriate New York State official designated to receive such amendments.

BY: Robert E. Moran
for A. Alan Blind
Vice President - Nuclear Power

Subscribed and sworn to
before me this 14th day
February, 2000.

ELIZABETH A. MELANSON
Notary Public, State of New York
No. 01ME4878094
Qualified in Orange County
Commission Expires Feb. 9, 20 01

Elizabeth A. Melanson
Notary Public

ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY 2000

Table 3.6-1

Non-Automatic Containment Isolation Valves Open Continuously Or Intermittently for Plant Operation

3418	851A	SWN-44-5-A or B(1)	1814B
3419	850A	SWN-51-5(1)	1814C
		SWN-44-1-A or B(1)	
4136	851B	SWN-51-1(1)	5018
			5019
744	850B	SWN-44-2-A or B(1)	5020
		SWN-51-2(1)	
888A	859A	SWN-44-3-A or B(1)	5021
888B	859C	SWN-51-3(1)	5022
958			5023
959	3416	SWN-44-4-A or B(1)	5024
990D	3417	SWN-51-4(1)	5025
1870	5459	SWN-71-5-A or B(1)	E-2
743	753H	SWN-71-1-A or B(1)	E-1
732	753G	SWN-71-2-A or B(1)	E-3
885A	SWN-41-5-A or B(1)	SWN-71-3-A or B(1)	E-5
885B	SWN-42-5	SWN-71-4-A or B(1)	MW-17
			MW-17-1
205	SWN-43-5	SA-24	85C
226	SWN-41-1-A or B(1)	SA-24-1	85D
227	SWN-42-1	PCV-1111-1	95C
250A	SWN-43-1	PCV-1111-2	95D
4925	SWN-41-2-A or B(1)	580A	IIP-500
250B	SWN-42-2	580B	IIP-501
4926	SWN-43-2	UH-43	IIP-502
250C	SWN-41-3-A or B(1)	UH-44	IIP-503
4927	SWN-42-3		IIP-504
250D	SWN-43-3		IIP-505
4928	SWN-41-4-A or B(1)	1814A	IIP-506
869A	SWN-42-4		IIP-507
878A	SWN-43-4		
869B			

(1) Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valve(s)in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).

refueling crane for this event must be equal or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A through visual inspection of the refueling crane shall be made after the dead-load test and prior to fuel handling.

6. The fuel storage building charcoal filtration system must be operating whenever spent fuel movement is taking place within the spent fuel storage areas unless the spent fuel has had a continuous 35-day decay period.

7. Radiation levels in the spent fuel storage area shall be monitored continuously whenever spent fuel movement is taking place in that area.

8. The equipment door, or a closure plate that restricts direct air flow from the containment, and at least one personnel door in the equipment door or closure plate and in the personnel air lock shall be properly closed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

9. Radiation levels in containment shall be monitored continuously.

10. During alteration of the core (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling shall be present to directly supervise the activity and, during this time, this person shall not be assigned other duties.

11. The minimum water level above the top of the reactor pressure vessel flange shall be at least 23 feet (El. 92'0") whenever movement of spent fuel is taking place inside containment.

12. If any of the conditions specified above cannot be met, suspend all operations under this specification (3.8.B). Suspension of operations shall not preclude completion of movement of the above components to a safe conservative position.

C. The following conditions are applicable to the spent fuel pit any time it contains irradiated fuel:

1. The spent fuel cask shall not be moved over <u>any</u> region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable. Furthermore, any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall

Technical Specification Figures 3.10-2, 3.10-3, 3.10-4, 3.10-5 and 3.10-6 are being deleted.

There are no replacement pages.

Note: Figure 3.10-4 was deleted in Amendment 152. However, a placeholder page was inserted between Figures 3.10-3 and 3.10-5. This placeholder page is being deleted.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
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	Calibration of transmitters extended on a one time basis to 37 months.
12.	Charging Flow	N.A.	R#	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R##	N.A.	Calibration of transmitters extended on a one time basis to 37 months.
14.	Boric Acid Tank Level	W	R#	N.A.	
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

Table 4.4-1

Containment Isolation Valves

Valve No.	System[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
95B	Equipment Airlock	Gas	47
95C	" "	Gas[7]	47
95D	" "	Gas[7]	47
4399	Sample Return to Cont. Sump	Water[4]	52
5132	" " "	Water[4]	52
IIP-500	22 S.G. Level	Gas	47
IIP-501	" " "	Gas	47
IIP-502	21 S.G. Level	Gas	47
IIP-503	" " "	Gas	47
IIP-504	Pressurizer Level	Gas	47
IIP-505	" "	Gas	47
IIP-506	Pressurizer Pressure	Gas	47
IIP-507	" "	Gas	47

Notes:

1. System in which valve was located.
2. Gas test fluid indicates either nitrogen or air as test medium.
3. Testable only when at cold shutdown.
4. Isolation Valve Seal Water System.
5. Sealed by Residual Heat Removal System fluid.
6. Sealed by Service Water System. Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).
7. Sealed by Weld Channel and Pressurization System.

4.19 METEOROLOGICAL MONITORING SYSTEM

Applicability

This specification applies to the surveillance requirements for the meteorological monitoring system.

Objective

To verify operability of the meteorological monitoring system such that adequate measurement and documentation of meteorological conditions at the site can be effected.

Specifications

A. Each meteorological monitoring instrumentation channel shall be demonstrated operable by performance of the surveillance testing required by Table 4.19-1.

B. Meteorological data shall be summarized and reported as required for inclusion in the Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.6.

Basis

This specification assures the operability of the meteorological monitoring instrumentation and the collection of meteorological data at the plant site. This data is used for estimating potential radiation doses to the public resulting from routine or accidental releases of radioactive materials to the atmosphere. A meteorological data collection program, as described in this specification, is necessary to meet the requirements of 10 CFR 50.36a (a) (2), Appendix E to 10 CFR 50 and 10 CFR 51.

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 analysis 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 20.1601(a) and 10 CFR 20.1601(b):

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 Shift Manager 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 1E 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 of 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 used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or 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 Annual 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.

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION MARKED-UP PAGES

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY 2000

On these marked-up pages from the current Tech Specs:

Additions are shown by ***bold italic***,

and

Deletions are shown by ~~double strikethrough~~.

Table 3.6-1

Non-Automatic Containment Isolation Valves Open Continuously
Or Intermittently for Plant Operation

3418	851A	SWN-44-5-A or B[1]	1814B
3419	850A	SWN-51-5[1]	1814C
		SWN-44-1-A or B[1]	
4136	851B	SWN-51-1[1]	5018
			5019
744	850B	SWN-44-2-A or B[1]	5020
		SWN-51-2[1]	
888A	859A	SWN-44-3-A or B[1]	5021
888B	859C	SWN-51-3[1]	5022
958			5023
959	3416	SWN-44-4-A or B[1]	5024
990D	3417	SWN-51-4[1]	5025
1870	5459	SWN-71-5-A or B[1]	E-2
743	753H	SWN-71-1-A or B[1]	E-1
732	753G	SWN-71-2-A or B[1]	E-3
885A	SWN-41-5-A or B[1]	SWN-71-3-A or B[1]	E-5
885B	SWN-42-5	SWN-71-4-A or B[1]	MW-17
			MW-17-1
205	SWN-43-5	SA-24	85C
226	SWN-41-1-A or B[1]	SA-24-1	85D
227	SWN-42-1	PCV-1111-1	95C
250A	SWN-43-1	PCV-1111-2	95D
4925	SWN-41-2-A or B[1]	580A	***IIP-500***
250B	SWN-42-2	580B	***IIP-501***
4926	SWN-43-2	UH-43	***IIP-502***
250C	SWN-41-3-A or B[1]	UH-44	***IIP-503***
4927	SWN-42-3	~~990A~~	***IIP-504***
250D	SWN-43-3	~~990B~~	***IIP-505***
4928	SWN-41-4-A or B[1]	1814A	***IIP-506***
869A	SWN-42-4		***IIP-507***
878A	SWN-43-4		
869B			

(1) Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-~~77~~***71*** series. Designation of the B valve(s)in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).

refueling crane for this event must be equal or greater than the maximum load to be assumed by the refueling crane during the refueling operation. A through visual inspection of the refueling crane shall be made after the dead-load test and prior to fuel handling.

6. The fuel storage building charcoal filtration system must be operating whenever spent fuel movement is taking place within the spent fuel storage areas unless the spent fuel has had a continuous 35-day decay period.

7. Radiation levels in the spent fuel storage area shall be monitored continuously whenever spent fuel movement is taking place in that area.

8. The equipment door, or a closure plate that restricts direct air flow from the containment, and at least one personnel door in the equipment door or closure plate and in the personnel air lock shall be properly closed. In addition, at least one isolation valve shall be operable or locked closed in each line penetrating the containment and which provides a direct path from containment atmosphere to the outside.

9. Radiation levels in containment shall be monitored continuously.

10. ~~A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place.~~ ***During alteration of the core (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling shall be present to directly supervise the activity and, during this time, this person shall not be assigned other duties.***

11. The minimum water level above the top of the reactor pressure vessel flange shall be at least 23 feet (El. 92'0") whenever movement of spent fuel is taking place inside containment.

12. If any of the conditions specified above cannot be met, suspend all operations under this specification (3.8.B). Suspension of operations shall not preclude completion of movement of the above components to a safe conservative position.

C. The following conditions are applicable to the spent fuel pit any time it contains irradiated fuel:

1. The spent fuel cask shall not be moved over <u>any</u> region of the spent fuel pit until the cask handling system has been reviewed by the Nuclear Regulatory Commission and found to be acceptable. Furthermore, any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall

Technical Specification Figures 3.10-2, 3.10-3, 3.10-4, 3.10-5 and 3.10-6 are being deleted.

There are no replacement pages.

Note: Figure 3.10-4 was deleted in Amendment 152. However, a placeholder page was inserted between Figures 3.10-3 and 3.10-5. This placeholder page is being deleted.

Table 4.1-1

Minimum Frequencies for Checks, Calibrations and Tests of Instrument Channels

	Channel Description	Check	Calibrate	Test	Remarks
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	Calibration of transmitters extended on a one time basis to 37 months.
12.	Charging Flow	N.A.	R#	N.A.	
13.	Residual Heat Removal Pump Flow	N.A.	R##	N.A.	Calibration of transmitters extended on a one time basis to 37 months.
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

Table 4.4-1

Containment Isolation Valves

Valve No.	System[1]	Test Fluid[2]	Minimum Test Pressure (PSIG)
95B	Equipment Airlock	Gas	47
95C	" "	Gas[7]	47
95D	" "	Gas[7]	47
4399	Sample Return to Cont. Sump	Water[4]	52
5132	" " "	Water[4]	52
IIP-500	***22 S.G. Level***	***Gas***	***47***
IIP-501	***" " "***	***Gas***	***47***
IIP-502	***21 S.G. Level***	***Gas***	***47***
IIP-503	***" " "***	***Gas***	***47***
IIP-504	***Pressurizer Level***	***Gas***	***47***
IIP-505	***" "***	***Gas***	***47***
IIP-506	***Pressurizer Pressure***	***Gas***	***47***
IIP-507	***" "***	***Gas***	***47***

Notes:

1. System in which valve was located.
2. Gas test fluid indicates either nitrogen or air as test medium.
3. Testable only when at cold shutdown.
4. Isolation Valve Seal Water System.
5. Sealed by Residual Heat Removal System fluid.
6. Sealed by Service Water System. Either A or B valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the B valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valve(s) associated with the penetration(s) as an additional required containment isolation valve(s).
7. Sealed by Weld Channel and Pressurization System.

4.19 METEOROLOGICAL MONITORING SYSTEM

Applicability

This specification applies to the surveillance requirements for the meteorological monitoring system.

Objective

To verify operability of the meteorological monitoring system such that adequate measurement and documentation of meteorological conditions at the site can be effected.

Specifications

A. Each meteorological monitoring instrumentation channel shall be demonstrated operable by performance of the surveillance testing required by Table 4.19-1.

B. Meteorological data shall be summarized and reported as required for inclusion in the ~~Semiannual~~ ***Annual*** Radioactive Effluent Release Report pursuant to Specification 6.9.1.6.

Basis

This specification assures the operability of the meteorological monitoring instrumentation and the collection of meteorological data at the plant site. This data is used for estimating potential radiation doses to the public resulting from routine or accidental releases of radioactive materials to the atmosphere. A meteorological data collection program, as described in this specification, is necessary to meet the requirements of 10 CFR 50.36a (a) (2), Appendix E to 10 CFR 50 and 10 CFR 51.

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 analysis 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):~~ ***10 CFR 20.1601(a) and 10 CFR 20.1601(b):***

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~~ ***Shift Manager*** 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 1E 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 of 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 used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines ~~of~~ ***or*** 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~~ ***Annual*** 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.

ATTACHMENT III

SAFETY ASSESSMENTS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY 2000

Introduction

In this Attachment, separate safety assessments are provided, one for each of the following proposed administrative changes:

a) Changes to Tables 3.6-1 and 4.4-1 to correct listing and editorial errors,

b) Change to Section 3.8.B.10 to reflect the wording in 10 CFR 50.54(m)(2)(iv),

c) Deletion of Figures 3.10-2 through 3.10-6,

d) Change to Table 4.1-1 to reflect change in level indication components,

e) Change to Sections 4.19.B and 6.14.1.1 to correct editorial errors,

f) Change to Section 6.12.1 to reference the current sections of 10 CFR 20,

g) Change to Section 6.12.1 to reflect an organizational title change, and

h) Change to Section 6.13.2 to correct a typographical error.

a) Changes To Tables 3.6-1 And 4.4-1 To Correct Listing And Editorial Errors

SECTION I - Description of Change

In Table 3.6-1 delete valves "990A" and "990B" and in Note 1, change "SWN-77 series" to "SWN-71 series." In Tables 3.6-1 and 4.4-1 add valves IIP-500 through IIP-507.

SECTION II - Evaluation of Change

Table 3.6-1 title is "Non-Automatic Isolation Valves Open Continuously Or Intermittently For Plant Operation" valves 990A and 990B do not fit this category. UFSAR Table 5.2-1 ("Containment Piping Penetrations And Valving"), Item 51; UFSAR Figure 5.2-20 ("Containment Isolation System Penetration Schematics"); and UFSAR Figure 5.2-29 ("Containment Isolation System Penetration Schematics"); show that these valves are the isolation valves for the recirculation pump discharge sample line, that the valves are normally closed and are opened intermittently after an accident to sample recirculation fluid. Finally these are motor operated valves that receive a close signal from the Phase A Containment Isolation Signal. Therefore, valves 990A and 990B should not be listed on Table 3.6-1.

Table 3.6-1 has several groupings of SWN valves, specifically the SWN-41 series, the SWN-42 series, the SWN-43 series, the SWN-44 series, the SWN-51 series and the SWN-71 series. There is no SWN-77 series listed in Table 3.6-1. Therefore, the statement "SWN-77 series" should be changed to "SWN-71 series."

UFSAR Table 5.2-1 and the In-Service Testing (IST) Program list valves IIP-500 through IIP-507. Therefore, Technical Specification Tables 3.6-1 and 4.4-1 are being updated to show these valves.

These changes are considered administrative since there is no change in the function, operation or physical configuration of the plant.

a) **Changes To Tables 3.6-1 And 4.4-1 To Correct Listing And Editorial Errors**

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed changes are administrative in nature. The changes involve correcting errors in Table 3.6-1 and additions to Tables 3.6-1 and 4.4-1 to reflect UFSAR Table 5.2-1 and the IST Program. These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed changes. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

a) Changes To Tables 3.6-1 And 4.4-1 To Correct Listing And Editorial Errors

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The changes involve correcting errors in Table 3.6-1 and additions to Tables 3.6-1 and 4.4-1 to reflect UFSAR Table 5.2-1 and the IST Program. This level of detail is either not listed or implied in the UFSAR; or already delineated in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed changes to the Technical Specifications do not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed changes do not represent a significant hazards consideration.

b) **Change To Section 3.8.B.10 To Reflect The Wording In 10 CFR 50.54(m)(2)(iv)**

SECTION I - Description of Change

In Section 3.8.B.10 change "A licensed senior reactor operator shall be at the site and designated in charge of the operation whenever changes in core geometry are taking place." to "During alteration of the core (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling shall be present to directly supervise the activity and, during this time, this person shall not be assigned other duties."

SECTION II - Evaluation of Change

In 10 CFR 50.54(m)(2)(iv) it states:

> "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person."

By law, Indian Point 2 is required to and does comply with this statement. This change is to remove any ambiguity that may have existed between the old statement in Section 3.8.B.10 and 10 CFR 50.54(m)(2)(iv).

This change is considered administrative since there is no change in the function, operation or physical configuration of the plant.

b) Change To Section 3.8.B.10 To Reflect The Wording In 10 CFR 50.54(m)(2)(iv)

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed change is administrative in nature. The change involves updating Section 3.8.B.10 to reflect 10 CFR 50.54(m)(2)(iv). This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

b) Change To Section 3.8.B.10 To Reflect The Wording In 10 CFR 50.54(m)(2)(iv)

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The change involves updating Section 3.8.B.10 to reflect 10 CFR 50.54(m)(2)(iv). This level of detail is not listed or implied in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed change to the Technical Specifications does not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed change does not represent a significant hazards consideration.

c) Deletion Of Figures 3.10-2 Through 3.10-6

SECTION I - Description of Change

Delete Figures 3.10-2, 3.10-3, 3.10-4, 3.10-5 and 3.10-6.

SECTION II - Evaluation of Change

In Amendment 194 Technical Specification Section 3.10 ("Control Rod And Power Distribution Limits") was revised and references to the Figures 3.10-2 through 3.10-6 were eliminated. In lieu of these figures the amended section referenced the Core Operating Limits Report (COLR). At the time of this amendment, Figures 3.10-2 through 3.10-6 should have been deleted, but were inadvertently left in.

Since Section 3.10 now only references Figure 3.10-1 and since there are no figures after Figure 3.10-6, no new pages will be inserted. Finally, Figure 3.10-4 was deleted in Amendment 152 and replaced with a placeholder page. This placeholder page will be removed and no new page will be inserted.

This change is considered administrative since there is no change in the function, operation or physical configuration of the plant.

c) Deletion Of Figures 3.10-2 Through 3.10-6

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed change is administrative in nature. The change involves the deletion of Figures 3.10-2, 3.10-3, 3.10-4, 3.10-5 and 3.10-6. This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

c) **Deletion Of Figures 3.10-2 Through 3.10-6**

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The change involves the deletion of Figures 3.10-2, 3.10-3, 3.10-4, 3.10-5 and 3.10-6. This level of detail is not listed or implied in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed change to the Technical Specifications does not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed change does not represent a significant hazards consideration.

d) Change To Table 4.1-1 To Reflect Change In Level Indication Components

SECTION I - Description of Change

In Table 4.1-1, Item 14 ("Boric Acid Tank Level"), under the "Remarks" column is the statement: "Bubbler tube rodded during calibration." This statement is being deleted.

SECTION II - Evaluation of Change

Boric Acid Storage Tanks 21BAT and 22BAT are normally covered tanks and the level in each tank was measured by a nitrogen bubbler system designed to determine the pressure difference between the top and bottom of the tank. A D/P cell was connected to two instrument tubes, one just penetrating the top of the tank in the nitrogen blanket above the solution level, and the second extending to the bottom of the tank immersed in the boric acid solution. The pressure difference measured equaled the level of boric acid solution above the open ended lower immersed instrument tube. A nitrogen supply was used on both sides of the D/P cell to displace the liquid from the submersed instrument tube and maintain a blanket on top of the borated water. As the level in the tank decreased, flow resistance in the bottom bubbler caused by fluid back pressure also decreased. The pressure in this line correspondingly decreased and the resulting pressure differential between the two lines would correlate with the tank's borated water level.

The tanks experienced frequent problems with this level instrumentation. The nitrogen bubbler system tubing that was immersed in the highly concentrated boric acid solution became plugged due to the crystallization of boric acid in the tube. Plugging of the level measurement tube would prevent the nitrogen purge supply from bubbling out of the tube. This would cause a higher back pressure on the high side of the D/P cell, which would cause the D/P cell to indicate a false level which was higher than actual. This higher than actual level indication would continue to increase as the pressure in the blocked tube increased up to the nitrogen supply pressure. To rectify the problem the bubbler tube had to be rodded out to clear the blockage. Under the original configuration, the statement, "Bubbler tube rodded during calibration," was included since if the bubbler tube was not rodded during the calibration of the level sensor, there could be an inaccurate calibration which could result in erroneous level indication.

Based on an evaluation of this problem, it was concluded that the best method of level measurement would be via a non-intrusive system having no contact with the process fluid. The best non-intrusive measurement system available uses microwave technology. The modification included a microwave transmitter/receiver sensor mounted above the top of each tank with a process seal to completely isolate the instrument from the tank contents. A locally mounted control panel converts the level measurement determined by the microwave sensor to a conventional instrumentation signal. The modification also removed the old system. This modification significantly improved the reliability and repeatability of Boric Acid Storage Tank level indication. Therefore, the potential for losing level indication of a required Technical Specification level has been significantly reduced.

d) Change To Table 4.1-1 To Reflect Change In Level Indication Components

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. This change does not affect possible initiating events for accidents previously evaluated or operation of the facility. While the configuration of the facility has changed with installation of the new level sensors, the safety-related function of theses sensors remains unchanged (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will annunciate in the CCR). The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The safety analysis of the facility remains complete and accurate. The plant conditions for which the design basis accidents have been evaluated are still valid. While the configuration of the facility has changed with installation of the new level sensors, the safety-related function of theses sensors remains unchanged (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will annunciate in the CCR). Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. While the configuration of the facility has changed with installation of the new level sensors, the safety-related function of theses sensors remains unchanged (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will annunciate in the CCR). Also, there are no changes to the operation of the facility. Thus the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

d) Change To Table 4.1-1 To Reflect Change In Level Indication Components

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The proposed change removes the rodding requirement for the retired bubbler tube. This level of detail is not listed or implied in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. While there has been a physical change (i.e., new level sensors), there are no functional changes (i.e., at a predetermined level of approximately 35% of instrument span, a low level alarm will still annunciate in the CCR).

SECTION V - Conclusion

Therefore, the proposed change to the Technical Specifications does not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed change does not represent a significant hazards consideration.

e) Change To Sections 4.19.B And 6.14.1.1 To Correct Editorial Errors

SECTION I - Description of Change

In Section 4.19.B and in Section 6.14.1.1, change "the Semiannual Radioactive Effluent Release Report" to "the Annual Radioactive Effluent Release Report."

SECTION II - Evaluation of Change

Amendment 172, issued July 1994, changed the submittal frequency of the Radioactive Effluent Release Report from semiannual to annual. In Amendment 198, issued in August 1998, Technical Specification Sections 3.9.A.2.c, 3.9.A.5.b, 3.9.B.2.c, 4.11.A.4, 4.11.B.3, 4.11.B.4 and Table 4.11-1 had "Semiannual Radioactive Effluent Release Report" replaced with "Annual Radioactive Effluent Release Report." At the time of this amendment, Technical Specification Sections 4.19.B and 6.14.1.1 should have been changed, but were overlooked.

This change is considered administrative since there is no change in the function, operation or physical configuration of the plant.

e) **Change To Sections 4.19.B And 6.14.1.1 To Correct Editorial Errors**

SECTION III - No Significant Hazards Evaluation

The proposed changes do not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed changes are administrative in nature. The change in Sections 4.19.B and 6.14.1.1 involve amending "the Semiannual Radioactive Effluent Release Report" to "the Annual Radioactive Effluent Release Report." These changes do not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed changes would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed changes are administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed changes would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed changes are administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

e) **Change To Sections 4.19.B And 6.14.1.1 To Correct Editorial Errors**

SECTION IV - Impact Of Changes

These change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The changes in Section 4.19.B and 6.14.1.1 involve amending "the Semiannual Radioactive Effluent Release Report" to "the Annual Radioactive Effluent Release Report." This level of detail is not listed or implied in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed changes to the Technical Specifications do not involve a significant hazards consideration. In addition, the proposed changes to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed changes do not represent a significant hazards consideration.

f) Change To Section 6.12.1 To Reference The Current Sections Of 10 CFR 20

SECTION I - Description of Change

In Section 6.12.1 change "As an acceptable alternative to the 'control device' or 'alarm signal' required by 10 CFR 20.203(c)(2)" to "As an acceptable alternative to the 'control device' or 'alarm signal' required by 10 CFR 20.1601(a) and 10 CFR 20.1601(b)."

SECTION II - Evaluation of Change

The NRC had completely revised 10 CFR 20 in 1991, and 10 CFR 20.203(c)(2) is no longer applicable. The appropriate references for Section 6.12.1 are 10 CFR 20.1601(a) and 10 CFR 20.1601(b).

In 10 CFR 20.1601 it states:

"(a) The licensee shall ensure that each entrance or access point to a high radiation area has one or more of the following features --

(1) A control device that, upon entry into the area, causes the level of radiation to be reduced below that level at which an individual might receive a deep-dose equivalent of 0.1 rem (1 mSv) in 1 hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates;

(2) A control device that energizes a conspicuous visible or audible alarm signal so that the individual entering the high radiation area and the supervisor of the activity are made aware of the entry; or

(3) Entryways that are locked, except during periods when access to the areas is required, with positive control over each individual entry.

(b) In place of the controls required by paragraph (a) of this section for a high radiation area, the licensee may substitute continuous direct or electronic surveillance that is capable of preventing unauthorized entry.

(c) A licensee may apply to the Commission for approval of alternative methods for controlling access to high radiation areas."

By law, Indian Point 2 is required to and does comply with this statement. This change is to remove any ambiguity that may have existed in Section 6.12.1 by referring to a non-applicable 10 CFR 20 Section.

This change is considered administrative since there is no change in the function, operation or physical configuration of the plant.

f) Change To Section 6.12.1 To Reference The Current Sections Of 10 CFR 20

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed change is administrative in nature. The change involves updating Section 6.12.1 to reference 10 CFR 20.1601(a) and 10 CFR 20.1601(b). This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

f) Change To Section 6.12.1 To Reference The Current Sections Of 10 CFR 20

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The change involves updating Section 6.12.1 to reference 10 CFR 20.1601(a) and 10 CFR 20.1601(b). This level of detail is not listed or implied in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed change to the Technical Specifications does not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed change does not represent a significant hazards consideration.

g) Change To Section 6.12.1 To Reflect An Organizational Title Change

SECTION I - Description of Change

In Section 6.12.1 change “Senior Watch Supervisor” to “Shift Manager.”

SECTION II - Evaluation of Change

This is a change in title only. There is no change in responsibilities or functions performed by this individual. This is an administrative change that affects only the “management” aspect of the plant. This change does not affect any equipment or physical plant attributes.

g) **Change To Section 6.12.1 To Reflect An Organizational Title Change**

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed change is administrative in nature. The change involves updating Section 6.12.1 to use the title "Shift Manager" instead of "Senior Watch Supervisor." This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

g) Change To Section 6.12.1 To Reflect An Organizational Title Change

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The change involves updating Section 6.12.1 to use the title "Shift Manager" instead of "Senior Watch Supervisor." For this change, the UFSAR is in the process of being updated. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed change to the Technical Specifications does not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed change does not represent a significant hazards consideration.

h) Change To Section 6.13.2 To Correct A Typographical Error

SECTION I - Description of Change

In Section 6.13.2 change "DOR Guidelines of NUREG-0588" to "DOR Guidelines or NUREG-0588."

SECTION II - Evaluation of Change

Technical Specification Section 6.13.1 states:

> "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 1E 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 of Modification of License No. DPR-26, dated October 24, 1980."

As can be seen from this statement, the DOR Guidelines and NUREG-0588 are two separate documents. Therefore, the statement: "DOR Guidelines of NUREG-0588" should actually be "DOR Guidelines or NUREG-0588." Since the "r" key is just above and to the left of the "f" key on a keyboard, this is evidenced to be a typographical error.

This change is considered administrative since there is no change in the function, operation or physical configuration of the plant.

h) Change To Section 6.13.2 To Correct A Typographical Error

SECTION III - No Significant Hazards Evaluation

The proposed change does not involve a significant hazards consideration because:

1) Does the proposed license amendment involve a significant increase in the probability or in the consequences of an accident previously evaluated?

   No. The proposed change is administrative in nature. The change involves updating Section 6.13.2 from "DOR Guidelines of NUREG-0588" to "DOR Guidelines or NUREG-0588." This change does not affect possible initiating events for accidents previously evaluated or alter the configuration or operation of the facility. The Limiting Safety System Settings and Safety Limits specified in the current Technical Specifications remain unchanged. Therefore, the proposed change would not involve a significant increase in the probability or in the consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed change is administrative in nature. The safety analysis of the facility remains complete and accurate. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected. Consequently no new failure modes are introduced as a result of the proposed change. Therefore, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

   No. The proposed change is administrative in nature. Since there are no changes to the operation of the facility or the physical design, the Updated Final Safety Analysis Report (UFSAR) design basis, accident assumptions, or Technical Specification Bases are not affected. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

h) Change To Section 6.13.2 To Correct A Typographical Error

SECTION IV - Impact Of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
UFSAR or SER Conclusions
Overall Plant Operations and the Environment

The change involves updating Section 6.13.2 from "DOR Guidelines of NUREG-0588" to "DOR Guidelines or NUREG-0588." This level of detail is not listed or implied in the UFSAR. Therefore, there is no UFSAR impact. There are no new failure modes introduced by this change. There are no physical changes to the facility and the plant conditions for which the design basis accidents have been evaluated are still valid. The operating procedures and emergency procedures are unaffected.

SECTION V - Conclusion

Therefore, the proposed change to the Technical Specifications does not involve a significant hazards consideration. In addition, the proposed change to the Technical Specifications has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Con Edison Nuclear Facility Safety Committee (NFSC). Both Committees concur that the proposed change does not represent a significant hazards consideration.

ATTACHMENT IV

CHANGED BASIS PAGE

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY 2000

The containment cooling function is provided by two independent systems: (1) fan-coolers plus charcoal filters and (2) containment spray. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 95ºF)[12]. In the event of a Design Basis Accident, sufficient cooling to reduce containment pressure at a rate consistent with limiting offsite doses to acceptable values is provided by three fan-cooler units and one spray pump. These constitute the minimum safeguards and are capable of being operated on emergency power with one diesel generator inoperable.

The iodine removal function is provided by two independent operating trains of the containment spray system. In the event of a Design Basis Accident, one containment spray pump provides sufficient flow to remove air borne elemental and particulate iodine at a rate consistent with limiting offsite doses to acceptable values.

Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of offsite power or operation of all emergency diesel generators.

The operability of the recirculation fluid pH control system ensures that there is sufficient trisodium phosphate (TSP) available in containment to guarantee a sump pH $\geq$ 7.0 during the recirculation phase of a postulated LOCA. This pH level is required to reduce the potential for chloride induced stress corrosion of austenitic stainless steel and assure the retention of iodine in the recirculating fluid. The specified amounts of TSP will result in a recirculation fluid pH between 7.0 and 9.5.

One of the five fan cooler units is permitted to be inoperable during power operation. This is an abnormal operating situation, in that the normal plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical.

However, because of the difficulty of gaining access to make repairs, it is important on occasion to be able to operate temporarily without at least one fan-cooler. Compensation for this mode of operation is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

The Component Cooling System is different from the system discussed above in that the pumps are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident[6]. During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards[7]. With two operable component cooling pumps, 100% redundancy will be provided. A total of three operable component cooling pumps will provide 200% redundancy. The 14 day out of service period for the third component cooling pump is allowed since this is the 200% redundant pump.

ATTACHMENT V

JUSTIFICATION FOR BASIS CHANGE

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY 2000

Justification For Basis Change

The change in the Basis in Technical Specification 3.3 is to correct a typographical error.

Prior to this correction, the third from the last sentence on page 3.3-13 read:

> "With two operable component cooling pumps, 100% redundancy will be provide."

As corrected, this sentence now reads:

> "With two operable component cooling pumps, 100% redundancy will be provided."

As can be seen, the original version was grammatically incorrect and the intent of the sentence remains unchanged upon correction of the typographical error.

Therefore, based on the criteria of 10 CFR 50.59, this change does not involve an unreviewed safety question.

Therefore, based on the criteria of 10 CFR 50.59, the change to Technical Specification Basis 3.3 does not involve an unreviewed safety question. In addition, the change to Technical Specification Basis 3.3 has been reviewed by the Station Nuclear Safety Committee (SNSC). The Committee concurs that the change does not represent an unreviewed safety question.