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. ("Consolidated Edison"), as holder of Facility Operating License No. DPR-26, hereby applies for amendment of the Technical Specifications contained in Appendix A to that License. Specifically, Consolidated Edison requests that Indian Point Unit No. 2 Technical Specifications be revised in accordance with the requirements of Generic Letter 83-37 dated November 1, 1983 regarding NUREG-0737 Technical Specifications.

The proposed Technical Specification page revisions are set forth in Attachment B to this Application. A Safety Assessment of the proposed changes is set forth in Attachment C to this Application. This Assessment demonstrates that the proposed changes do not represent

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a significant hazards consideration as defined in 10 CFR 50.92(c) and will not cause any change in the types, or an increase in the amounts of effluents, or any change in the authorized power level of the facility.

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INC.

By: John D. O'Toole
John D. O'Toole
Vice President

Subscribed and sworn to
before me this 21 day
of December 1984.

Conrad Tromba
Notary Public

CONRAD TROMBA
Notary Public State of New York
No. 30-4022375
Qualified in Nassau County
Terms expires March 30, 1985

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CONSOLIDATED EDISON COMPANY	)	Docket No. 50-247
OF NEW YORK, INC.	)	
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STATE OF NEW YORK)
) ss:
COUNTY OF NEW YORK)

AFFIDAVIT OF SERVICE

Joseph C. Carbonara , being duly sworn, states:

That he is an Engineer employed by Consolidated Edison Company of New York, Inc., and that he has served the foregoing document, sworn to on December 21, 1984, entitled "Application for Amendment to Operating License", by mailing copies thereof, first class, postage prepaid and properly addressed to the following persons:

Jay Dunkelberger
Division of Policy Analysis and Planning
New York State Energy Office
Agency Building 2, Empire State Plaza
Albany, New York 12222

Hon. Cheryl Harding
Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York

Joseph Carbonara
Joseph C. Carbonara

Subscribed and sworn to
before me this 21 day
of December, 1984.

Conrad Tromba
Notary Public

CONRAD TROMBA
Notary Public State of New York
No. 30-4022875
Qualified in Nassau County
Terms expires March 30, 1985

ATTACHMENT A

Response to Generic Letter No. 83-37

NUREG - 0737 Technical Specifications

Consolidated Edison of New York

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

December, 1984

Generic Letter No. 83-37 addressed a number of NUREG 0737 items which were scheduled for completion after December 31,1981. The generic letter included a request and guidance for submitting technical specifications which would cover those systems and instrumentation resulting from our implementation of NUREG 0737 Items. The information provided below addresses each of the items contained in Generic Letter No. 83-37 .

(1) Reactor Coolant System Vents (II.B.1)

By letter dated February 28, 1984, Consolidated Edison of New York submitted to the NRC staff proposed revisions to the Technical Specifications for Indian Point Unit No. 2 (IP2) to incorporate requirements relative to NUREG 0737,Item II.B.1. Those proposed technical specifications follow the intent of the model technical specifications provided in Enclosure 3 of Generic Letter No.83-37 .

(2) Post Accident Sampling (II.B.3)

Consistent with the guidance contained in Generic Letter No. 83-37, proposed technical specification revisions which address the administrative controls for post accident sampling and analysis are provided in Attachment B to this submittal.

(3) Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)

Enhanced Auxiliary Feedwater System Technical Specifications were issued by the NRC as Amendments Nos. 60, 72, 74, and 79 to DPR-26 for IP2. Therefore, this requirement has already been implemented and no additional technical specifications are necessary.

(4) Noble Gas Effluent Monitors (II.F.1.1)

Attachment B to this submittal includes proposed technical specification revisions in response to this item.

(5) Sampling and Analysis of Plant Effluents(II.F.1.2)

Consistent with the guidance contained in Generic Letter No.83-37, proposed technical specification revisions which address the administrative controls for sampling and analysis for radioiodine and particulates in plant gaseous effluents during and following an accident are provided in Attachment B to this submittal.

(6) Containment High Range Radiation Monitor (II.F.1.3)

Proposed technical specification changes in response to this item are included in Attachment B to this submittal.

(7) Containment Pressure Monitor (II.F.1.4)

Proposed technical specification changes in respose to this item are included in Attachment B to this submittal.

(8) Containment Water Level Monitor (II.F.1.5)

Technical specifications which encompass the model technical specification requirements of this item are already included in the existing IP2 Technical Specifications. Operability and surveillance requirements for these monitors are found in Specification 3.1.F.1 and 4.1.a (Table 4.1-1) of the current IP2 Technical Specifications. Therefore, no additional technical specification change is necessary.

(9) Containment Hydrogen Monitor (II.F.1.6)

Attachment B to this submittal includes proposed technical specification changes in response to this item.

(10) Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

The subcooling margin monitor is already included in the IP2 Technical Specifications (Table 3-5). Our designs for the installed reactor coolant inventory tracking system and the core exit thermocouples were provided to the NRC by letter dated September 4, 1982 and in our March 23, 1983 response to Generic Letter No. 82-28 . As indicated by the NRC letter dated May 2, 1984, conformance of some of the IP2 ICC system components to this NUREG 0737 Item are still undergoing review by the Staff. Upon completion of the NRC review of the IP2 ICC design appropriate proposed technical specification revisions will be submitted.

(11) Control Room Habitability Requirements (III.D.3.4)

The toxic gas technical specification requirements of this item are addressed by the proposed technical specification changes included in this submittal (Attachment B). The remaining technical specification requirements of this item are already included in Sections 3.3.H and 4.5.E of the current IP2 Technical Specifications.

ATTACHMENT B

APPLICATION FOR AMENDMENT

TO OPERATING LICENSE

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License No. DPR-26

December, 1984

TABLE 3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM NO. OF CHANNELS OPERABLE*	APPLICABLE ACTIONS
1. Pressurizer Water Level	2	1
2. Reactor Coolant System Subcooling Margin Monitor	1**	1,2
3. PORV Position Indicator (Limit Switch)	1/valve***	1
4. PORV Block Valve Position Indicator (Limit Switch)	1/valve****	1
5. Safety Valve Position Indicator (Acoustic Monitor)	1/valve	1
6. Auxiliary Feedwater Flow Rate	1/S.G.*****	1
7. Wide Range Containment Pressure Monitor (PT-3300,PT-3301)	1	3
8. Plant Vent Noble Gas Effluent Monitor(R-27)	1	3
9. Main Steam Line Radiation Monitor (R-28,R-29,R-30,R-31)	1/steam line	3
10. High Range Containment Radiation Monitor (R-25, R-26)	1	3
11. Containment Hydrogen Concentration Monitor	2	3

TABLE 3-5 (Continued)

Footnotes:

* Encompasses the entire channel from sensor to display where either an indicator, recorder or alarm is acceptable.

** PROTEUS Subcooling margin readout can be used as substitute for the subcooling monitor.

*** Except at times when the associated block valve is closed and de-energized. Acoustic monitoring of PORV position (headered discharge) can be used as a substitute for the PORV Position Indicator-Limit Switches instrument.

**** Except at times when the block valve is closed and de-energized.

***** Steam Generator Level instrumentation can be used as a substitute for auxiliary feedwater flow rate monitoring.

TABLE 3-5 (Continued)

ACTION STATEMENTS

1. With the number of operable accident monitoring instrumentation channels less than the minimum channels operable requirement of Table 3-5, restore the inoperable instrument channel(s) to operable status within 7 days and/or recorder(s) within 14 days. If the minimum number of channels required is not restored to meet the above requirements within the time periods specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

   b. If the requirements are not satisfied within an additional 48 hours after hot shutdown, the reactor shall be cooled to below 350°F utilizing normal operating procedures. The cooldown shall start no later than the end of the 48 hour period.

2. If the subcooling margin monitor is inoperable for more than seven (7) days, plant operation may continue for an additional thirty (30) days provided that steam tables are continously maintained in the control room and the subcooling margin is determined and recorded once a shift.

3. With the number of operable channels less than required by the minimum number of channels requirement, within 7 days either restore the inoperable channel(s) to the minimum operable status or:

   a. Initiate an alternate method of monitoring the appropriate parameter(s), and

   b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2.j within the next 14 days following the event.

F. Service Water System

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

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

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

G. Hydrogen Recombiner System and Post Accident Containment Venting System

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

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

   b) The post accident containment venting system is operable.

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

2. During power operation, the requirements of 3.3.G.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.G.1 within the time specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures.

   a) One hydrogen recombiner unit or its associated flow path, or oxygen supply system or control system may be inoperable for a period not to exceed thirty days, provided the other recombiner unit and the post accident containment venting system are operable.

   b) The post accident containment venting system may be inoperable for a period not to exceed thirty days provided that both hydrogen recombiners are operable.

H. Control Room Air Filtration System

1. The control room air filtration system shall be operable at all times when containment integrity is required.

2. From the date that the control room air filtration system becomes and remains inoperable for any reason, operations requiring containment integrity are permissible only during the succeeding 3.5 days. At the end of this 3.5 days period if the conditions for the control room air filtration system cannot be met, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the conditions are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

3. Two independent toxic gas detection systems, each capable of detecting chlorine, anhydrous ammonia, and hydrogen cyanide, shall be operable at all times except as specified in 3.a, 3.b, or 3.c below. The alarm/trip setpoint for each toxic gas system shall be adjusted to actuate at a toxic gas concentration of less than or equal to 3.5 ppm.

a) With one toxic gas detection system inoperable, restore the inoperable detection system to operable status within 7 days.

b) If 3.a above cannot be satisfied within the specified time, then within the next 6 hours initiate and maintain operation of the control room ventilation system in the recirculation mode of operation.

c) With both toxic gas detection systems inoperable for any one toxic gas, within one hour initiate and maintain operation of the control room ventilation in the recirculation mode of operation.

I. Cable Tunnel Ventilation Fans

1. The reactor shall not be made critical unless the two cable tunnel ventilation fans are operable.

2. During power operation, the requirement of 3.3.I.1 may be modified to allow one cable tunnel ventilation fan to be inoperable for seven days, provided the other fan is operable.

philosophy, the selected venting initiation point of 3 percent hydrogen maximizes the time period before venting is required while at the time allows a sufficient margin of safety below the lower flammability limit of hydrogen.

The control room air filtration system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The control room system is designed to automatically start upon control room isolation. Control room isolation is initiated either by a safety injection signal or by detection of high radioactivity in the control room. If the control room air filtration system is found to be inoperable,there is no immediate threat to the control room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within 3.5 days, the reactor is placed in the hot shutdown condition. If the repairs cannot be completed within an additional 48 hours, the reactor is placed in the cold shutdown condition.

The control room ventilation sytem is equipped with a toxic gas detection system consisting of redundant monitors capable of detecting chlorine, anhydrous ammonia, and hydrogen cyanide. These toxic gas detection systems are designed to isolate the control room from outside air upon detection of toxic concentration of the monitored gases in the control room ventilation system. The operability of the toxic gas detection systems provides assurance that the control room operators will have adequate time to take protective action in the event of an accidental toxic gas release. Selection of the gases to be monitored and the setpoint established for the monitors are based on the results described in the Indian Point Unit No. 2 Control Room Habitability Study dated May, 1981.

The cable tunnel is equipped with two temperature controlled ventilation fans. Each fan has a capacity of 21,000 cfm and is connected to a 480v bus. One fan will start automatically when the temperature in the tunnel reaches 100°F. Under the worst conditions, i.e. loss of outside power and all the Engineered Safety Features in operation, one ventilation fan is capable of maintaining the tunnel temperature below 104°F. Under the same worst conditions, if no ventilation fans were operating, the natural air circulation through the tunnel would be sufficient to limit the gross tunnel temperature below tolerable value of 140°F. However, in order to provide for ample tunnel ventilation capacity, the two ventilation fans are required to be operable when the reactor is made critical. If one ventilation fan is found inoperable, the other fan will ensure that cable tunnel ventilation is available.

Valves 856A, C, D and E are maintained in the open position during plant operation to assure a flow path for high-head safety injection during the injection phase of a loss-of-coolant accident. Valves 856B and F are maintained in the closed position during plant operation to prevent hot leg injection during the injection phase of a loss-of coolant accident. As an additional assurance of preventing hot leg injection, the valve motor

TABLE 4.1-1(CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
10.	Rod Position Bank Counters	S	N.A.	N.A.	With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
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	R	N.A.	
16.	Boron Injection Tank Level	W	R	R	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	a. Containment Pressure	D	R	M	Wide Range
	b. Containment Pressure	S	R	M	Narrow Range
	c. Containment Pressure(PT-3300,PT-3301)	M	R	N.A.	High Range
19.	Process and Area Radiation Monitoring Systems	D	R	M	
20.	Boric Acid Make-up Flow Channel	N.A.	R	N.A	
21.A	Containment Sump and Recirculation Sump Level (Discrete)	S	R	R	Discrete Level Indication Systems.
21.B	Containment Sump, Recirculation Sump and Reactor Cavity Level (Continuous)	S	R	R	Continuous Level Indication Systems.
21.C	Reactor Cavity Level Alarm	N.A.	R	R	Level Alarm System
21.D	Containment Sump Discharge Flow	S	R	M	Flow Monitor
21.E	Containment Fan Cooler Condensate Flow	S	R	M*	*Monthly visual inspection of condensate weirs only.

TABLE 4.1-1(CONTINUED)

	Channel Description	Check	Calibrate	Test	Remarks
37.	Wide Range Plant Vent Noble Gas Effluent Monitor (R-27)	S	R	N.A.	
38.	Main Steam Line Radiation Monitor (R-28, R-29, R-30, R-31)	S	R	N.A.	
39.	High Range Containment Radiation Monitor (R-25, R-26)	S	R*	N.A.	*Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.
40.	Containment Hydrogen Monitor	Q	Q*	N.A.	*Calibration will be performed using calibration span gas.

B. Containment Spray System

1. System tests shall be performed at each reactor refueling interval. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

2. The spray nozzles shall be tested for proper functioning at least every five years.

3. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

C. Hydrogen Recombiner System

1. A complete recombiner system test shall be performed at each normal reactor refueling on each unit. The test shall include verification of ignition and attainment of normal operating temperature.

2. A complete control system test shall be performed at intervals not greater than six months on each unit. The test shall consist of a complete dry run startup using artificially generated signals to simulate light off.

3. The above tests will be considered satisfactory if visual observations and control panel indication indicate that all components have operated satisfactorily.

4. Each recombiner air-supply blower shall be started at least at two month intervals. Acceptable levels of performance shall be that the blowers start, deliver flow, and operate for at least 15 minutes.

c) Verifying that the system maintains the control room at a neutral or positive pressure relative to the outside atmosphere during system operation.

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 1840 cfm ± 10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at ambient conditions and at a flow rate of 1840 cfm ± 10%.

7. Each toxic gas detection system shall be demonstrated operable by performance of a channel check at least once per day, a channel test at least once per 31 days and a channel calibration at least once per 18 months.

F. Fuel Storage Building Air Filtration System

The fuel storage building air filtration system specified in Specification 3.8 shall be demonstrated operable:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

2. At each refueling shutdown prior to refueling operations or (1) after any structural maintainance on the HEPA filter or charcoal adsorber housings, or (2) at any time painting, fire or chemical releases could alter filter integrity by:

a) Verifying a system flow rate at ambient conditions of 20,000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

6.6 REPORTABLE OCCURRENCE ACTION

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

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specifications 6.9.1.9.1 and/or 6.9.1.9.2.

b. Each Reportable Occurrence submitted to the Commission shall be reviewed by the SNSC and submitted to the NFSC Chairman, and the Vice President-Nuclear Power.

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, system 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 amd 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.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and any changes to them shall be reviewed and approved for implementation in accordance with a written administrative control procedure approved by the appropriate General Manager, with the concurrence of the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. The administrative control procedure required by this specification shall, as a minimum require that:

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

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

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

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

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

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

(i) Training of personnel,

(ii) Procedures for sampling and analysis,

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

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Region I Office of Inspection and Inforcement 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 FSAR 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 March 1 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.

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[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 20.407 Of 10 CFR Part 20

Special Reports

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

a. Each containment integrated leak rate test shall be the subject of a summary technical report including analyses and interpretation of the results which demonstrate compliance in meeting the leak rate limits specified in the Technical Specifications.

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

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

d. Inoperable Fire protection and detection equipment (Specification 3.13).

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

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

g. Radioactive effluents (Specification 3.9).

h. Radiological environmental monitoring (Specification 4.11)

i. Meteorological monitoring instrumentation (Specification 3.15)

j. 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.

ATTACHMENT C

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Safety Assessment

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2
Docket No. 50-247
Facility Operating Licence No. DPR-26

December, 1984

Affected pages and tables : Table 3-5, Table 4.1-1, pages 6-22, 3.3-6, 3.3-7 and 4.5-2

Description of Changes

Tables 3.5 and 4.1-1 were revised to include the appropriate Limiting Condition for Operation and Surveillance requirements for the following NUREG-0737 accident monitoring instrumentation:

a) Noble Gas Monitor, NUREG-0737 Item II.F.1.1 ;

b) High Range Containment Radiation Monitor, NUREG-0737 Item II.F.1.3 ;

c) High Range Containment Pressure Monitor, NUREG-0737 Item II.F.1.4 ; and

d) Containment Hydrogen Monitor, NUREG-0737 Item II.F.1.6

For these monitors, Table 3-5 now includes requirements for the minimum number of channels operable, and varying action requirements for inoperable accident monitoring instrumentation. Table 4.1-1 now includes the surveillance requirements for these monitors. Since the above accident monitors do not have any functional capabilities (i.e. isolation, trip function), a channel functional test in not applicable for these monitors. Page 6-22 was revised to include applicable reporting requirements for inoperable radiation and hydrogen concentration monitors. This change is consistent with the model technical specifications provided in Enclosure 3 of Generic Letter 83-37.

The operability and surveillance requirements for the containment atmosphere sampling system (sampling pump, piping and valves) within the Hydrogen Recombiner System specifications,Sections 3.3.G.1.c and 4.5.C.3 of the existing Technical Specifications, were deleted from pages 3.3-6, 3.3-7 and 4.5-2 since these functions have been replaced by the new containment hydrogen concentration monitor system and the new specifications for that system are proposed herein.

Significant Hazards Evaluation

This revision constitutes additional limitations, restrictions and controls not presently found in the Technical Specifications and is intended to increase the reliability of the accident monitoring systems installed to meet various NUREG-0737 Items. The operation of Indian Point Unit No. 2 with these proposed changes will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated;

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

Also the Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards considerations by providing certain examples published in the Federal Register on April 6, 1983 (48 FR 14864). One of the examples of an amendment which will likely be found to involve no significant hazards considerations is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement. The proposed technical specification changes fall within the Commission's example (ii) of a change not likely to involve a significant hazards consideration.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.

Affected Pages : 3.3-7(a), 3.3-15, 4.5-6

Description of Changes

Section 3.3.H.3 was added to the Control Room Filtration System Section of the Technical Specifications to include the appropriate Limiting Condition for Operation for the control room toxic gas detection systems which were installed to satisfy NUREG-0737 Item II.D.3, "Control Room Habitability". Page 3.3-15 was revised to include the basis for these monitors and the attendant surveillance requirements were incorporated in Section 4.5.E.7 on page 4.5-6.

Significant Hazards Evaluation

This revision constitutes additional limitations, restrictions and controls not presently found in the Technical Specifications and is intended to increase the reliability of the Control Room Toxic Gas Detection systems installed to meet NUREG-0737 Item II.D.3. The operation of Indian Point Unit No. 2 with these proposed changes will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated;

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

Also the Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards considerations by providing certain examples published in the Federal Register on April 6, 1983 (48 FR 14864). One of the examples of an amendment which will likely be found to involve no significant hazards considerations is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement. The proposed technical specification changes fall within the Commission's example (ii) of a change not likely to involve a significant hazards consideration.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.

Affected pages: 6-12, 6-13, and 6-14

Description of changes

Section 6.8.4 was added to the Administrative Controls Section of the Technical Specifications requiring that a program be implemented to provide assurance that the capability to obtain and analyze reactor coolant,containment atmosphere samples, and radioactive iodine and particulates in plant gaseous effluents under accident conditions will be maintained. This change implements the requirements of NUREG-0737 Item II B.3 and II F.1.2 and is based on the model Technical Specifications provided in Enclosure 3 of Generic Letter 83-37

Significant Hazards Evaluation

This change constitutes an additional limitation, restriction and control not presently found in the Technical Specifications and is intended to increase the reliability of the post-accident sampling system and the in plant effluent monitoring capabilities. The operation of Indian Point Unit No. 2 with this proposed change will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated;

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

Also the Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards considerations by providing certain examples published in the Federal Register on April 6, 1983 (48 FR 14864). One of the examples of an amendment which will likely be found to involve no significant hazards considerations is a change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications; for example, a more stringent surveillance requirement. The proposed technical specification changes fall within the Commission's example (ii) of a change not likely to involve a significant hazards consideration.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.