January 4, 1977

Docket No.: 50-286

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Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

The Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated November 23, 1976. As discussed with your staff, modifications have been made to your proposed changes to meet regulatory requirements.

This amendment revises the Technical Specifications to establish requirements for inspection and testing of shock suppressors (snubbers) at Indian Point Unit No. 3.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 6
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

Const 1

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 6
License No. DPR-64

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. and the Power Authority of the State of New York (the licensees) sworn to November 23, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 4, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 6

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
i & ii	i & ii
iii & iv	iii & iv
--	3.13-1 & 3.13-2
--	Table 3.13-1 (8 pgs.)
--	4.11-1 - 4.11-3

Changed areas on the revised and new pages are shown by marginal lines.

Pages i & iii are unchanged and are included for convenience only.

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LIST OF TABLES

3.13 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the operability of snubbers required for protection of safety-related components.

Objective

To define the time during which reactor operation is permitted after detection of inoperable snubbers.

Specification

1. During all modes of operation except cold shutdown and refueling all snubbers listed in Table 3.13-1 shall be operable except as noted in 3.13.2 through 3.13.4.
2. From and after the time that any snubber is determined to be inoperable, continued reactor operation or hot shutdown is permissible only during the succeeding 72 hours unless the snubber is made operable sooner or replaced.
3. If the requirements of 3.13.2 cannot be met, the reactor shall be in a cold shutdown condition within an additional 36 hours.
4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown mode, the snubber shall be made operable or replaced prior to bringing the reactor above cold shutdown.
5. Snubbers may be added to safety related systems without prior License Amendment to Table 3.13-1. A revision to Table 3.13-1 will be included with the next License Amendment request.

Basis

Snubbers are required to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping in the event of dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation. Because the snubber protection is required only during low-probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shut-down condition will permit an orderly shutdown consistent with standard operating procedures, Specification 3.13.4 prohibits startup if snubbers are known to be inoperable.

TABLE 3.13-1 (SHEET 1 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION (1)		CATEGORY (2)
1	MS-R-1-2-H	AFB	63'	3
1	MS-R-1-3-H	AFB	63'	3
1	MS-R-200-H	VC	101'	3
2	MS-R-2-1-H	AFB	75'	3
2	MS-R-2-2-H	AFB	75'	3
2	MS-R-2-3-H	AFB	75'	3
3	MS-R-100-H	VC	101'	3
3	MS-R-3-1-H	AFB	75'	3
3	MS-R-3-2-H	AFB	75'	3
3	SR-M53	AFB	75'	3
4	MS-R-4-1-H	AFB	63'	3
4	MS-R-4-2-H	AFB	63'	3
4	SR-M55	AFB	63'	3
5	BFD-R-5-1-H	AFB	40'	3
6	BFD-R-6-1-H	AFB	41'	3
6	BF - R - 300-H	AFB	59'	3
7	BFD-R-7-1-H	AFB	41'	3
8	BFD-R-8-1-H	AFB	39'	3
9	AC-R-215-H	PAB	17'	4
9	AC-R-221-H	PAB	26'	4
9	AC-R-216-H	PAB	17'	4
9	AC-R-9-2-H	PAB	30'	4
9	AC-R-9-11-H	PAB	51'	4
9	AC-R-227-H	PAB	53'	4
9	AC-R-222-H	PAB	44'	4
10	AC-R-40-H	VC	59'	3
10	AC-R-41-H	VC	59'	3
10	AC-R-10-7-H	VC	59'	3

TABLE 3.13-1 (SHEET 2 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION		CATEGORY
10	AC-R-10-12-H	PAB	52'	4
10	AC-R-10-13-H	PAB	52'	4
10	AC-R-10-18A-H	PAB	26'	4
10	AC-R-10-18B-H	PAB	26'	4
12A	SW-R-12a-7-H	VC	91'	3
12C	M/S - 12d - 39 SW - R - 12c - H	VC	92'	3
12C	SW - R - 12C - 13 -H	VC	92'	3
12E	SW - R - 12E - 3A -H	VC	87'	3
16	SI - R - 16 - 9 - H	PAB	61	4
16	SI - R - 16 - 28 - H	VC	60'	3
16A	SI - R - 16A - 11 - H	VC	59'	3
17	CH - R - 781 - H	VC	58'	3
17B	CH - R - 17B - 16A - H	VC	83'	3
17B	CH - R - 17B - 23 - H	VC	49'	3
17C	CH - R - 17C - 5A - H	VC	80'	3
17C	CH - R - 17C - 14 - H	VC	48'	3
31	SI - R - 537 - H	PAB	42'	4
38A	RC - R - 38A - 3 - H	VC	60'	3
38B	RC - R - 38B - 1 - H	VC	58'	3
45	BD - R - 45 - 8 - H	VC	66'	3
45	BD - R - 45 - 16B - H	PAB	61'	4
46	BD - R - 46 - 4 - H	VC	68'	3
47	BD - R - 47 - 28 - H	PAB	62'	4
48	BD - R - 48 - 18 - H	PAB	62'	4
52A	AC - R - 52A - ? - H	VC	49'	3
53A	AC - R - 53A - 11 - H	VC	50'	3

TABLE 3.13-1 (SHEET 3 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION		CATEGORY
56	SI - R - 56 - 5 - H	PAB	44'	4
60	SI - H - 60 - 1 - V	VC	55'	3
60	SI - R - 60 - 2 - H	VC	55'	3
60	SI - R - 404 - H	PAB	55'	4
61	RC - R - 61 - 5 - H	VC	67'	3
61	RC - R - 61 - 11 - H	VC	78'	3
62	RC - R - 25 - H	VC	70'	3
70	RC - R - 70 - 6A - H	VC	103'	3
70	RC - R - 70 - 6B - H	VC	103'	3
70	RC - R - 70 - 8 - H	VC	102'	3
70	RC - R - 70 - 9 - H	VC	101'	3
70	RC - R - 70 - 11 - H	VC	76'	3
70	RC - R - 70 - 12A - H	VC	65'	3
70	RC - R - 70 - 12B - H	VC	65'	3
77	CH - R - 77 - 15 - H	VC	46'	3
79	CH - R - 13 - H	VC	56'	3
79	CH - R - 18 - H	VC	56'	3
79	CH - R - 19 - H	VC	56'	3
80	CH - R - 80 - 4 - H	VC	58'	3
81	WD - R - 81 - 2 - H	VC	51'	3
82	RC - R - 82 - 2 - H	VC	51'	3
82	RC - R - 82 - 3 - H	VC	52'	3
84	RC - R - 84 - 2 - H	VC	51'	3
84	RC - R - 84 - 3 - H	VC	51'	3
93	SI - R - 93 - 1A - H	VC	85'	3

TABLE 3.13-1 (SHEET 4 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION		CATEGORY
93	SI - R - 93 - 1B - H	VC	85'	3
94	SI - R - 94 - 1A - H	VC	85'	3
94	SI - R - 94 - 1B - H	VC	86'	3
103	PW - R - 227 - H	PAB	49'	4
104	CH - R - 730 - H	PAB	45'	4
106	WD - R - 641 - H	PAB	41'	4
107	WD - R - 107 - 16 - H	PAB	42'	4
116	CH - R - 656 - H	PAB	44'	4
117	CH - R - 801 - H	PAB	44'	4
121	CH - R - 121-14-H	PAB	45'	4
130	WD - R - 673 - H	PAB	44'	4
132	WD - R - 132 - 4 - H	PAB	74'	4
155	AC - R - 155 - 2 - H	PAB	33'	4
161	SI - R - 161-17A-H	PAB	40'	4
190	AC - R - 190 - 1 - H	PAB	33'	4
202	CH - R - 202 - 3 - H	PAB	75'	4
208	CH - R - 208 - 10 - H	PAB	75'	4
224	PW - R - 404 - H	PAB	68'	4
235	CH - R - 235 - 9B - H	PAB	80'	4
270	SI - R - 270 - 27 - H	PAB	61'	4
342	RC - R - 342 - 5A - H	VC	126'	3
342	RC - R - 342 - 5B - H	VC	126'	3
342	RC - R - 342 - 6 - H	VC	103'	3
343	RC - R - 343-4A-H	VC	123'	3
343	RC - R - 343 - 4B - H	VC	123'	3
343	RC - R - 343 - 5 - H	VC	103'	3

TABLE 3.13-1 (SHEET 5 OF)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION		CATEGORY
344	RC - R - 344 - 4A - H	VC	123	3
344	RC - R - 344 - 4B - H	VC	123	3
344	RC - R - 344 - 5 - H	VC	103'	3
351	PWR - 127	VC	64'	3
352	PWR - 152	VC	65'	3
353	PWR - 147B	VC	62'	3
353	PWR - 148	VC	49'	3
353	SI-R-353 - 4 - H	VC	59'	3
354	PW - R - 484 - H	PAB	49'	4
356	SI - R - 356 - 2 - H	VC	53'	3
356	SI - R - 356 - 3 - H	VC	55'	3
356	SI - R - 356 - 4 - H	VC	61'	3
356	SI - R - 356 - 5 - H	VC	61'	3
358	SI - R - 44 - H	VC	57'	3
361	SI - R - 361 - 1B - H	VC	53'	3
361	SI - R - 361 - 4 - H	VC	53'	3
361	SI - R - 361 - 8 - H	VC	65'	3
361	SI - R - 361 - 9 - H	VC	65'	3
418	WD - R - 693 - H	PAB	43'	4
475	SI - R - 475 - 7 - H	VC	78'	3
475	SI - R - 475 - 11 - H	VC	79'	3
475	SI - R - 475 - 14 - H	VC	79'	3
479	PW - R - 479 - 3 - H	PAB	79'	4
480	M/S - 493 CH - R - 480A - H	PAB	52'	4
552	RC - R - 552 - 3 - H	VC	61'	3
592	CH - R - 592 - 8 - H	PAB	35'	4
595	CH - R - 695 - H	PAB	50'	4

TABLE 3.13-1 (SHEET 6 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION		CATEGORY
611	SI - R - 611 - 4 - H	VC	81'	3
611	SI - R - 611 -10 - H	VC	79'	3
654	AC - R - 212 - H	PAB	17'	4
654	AC - R - 213 - H	PAB	17'	4
654	AC - R - 214 - H	PAB	18'	4
753	RC - R - 753 - 17 - H	VC	60'	3
776	RC - R - 506 - H	VC	64'	3
776	RC - R - 510 - H	VC	63'	3
778	RC - R - 778 - 2 - H	VC	64'	3
778	RC - R - 778 - 4 H	VC	63'	3
789	RC - R - 501 - H	VC	64'	3
789	RC - R - 505 - H	VC	63'	3
791	RC - R - 791 - 2 - H	VC	64'	3
791	RC - R - 791 - 4 - H	VC	63'	3
1026	MS - R - 1026 - 2 - H	AFB	66'	3
1027	MS - R - 1027 - 7 - H	AFB	56'	3
1027	MS - R - 1027 - 4 - H	AFB	68'	3
1133	MS - R - 1133 - 2A - H	AFB	56'	3
1134	M/S - 1133 2A	AFB	56'	3
	M/S - R-1134 - H			3
1143	IA - R - 1143 - 3 - H	CR	25'	4

TABLE 3.13-1 (SHEET 7 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

LINE NO.	SNUBBER NO.	LOCATION		CATEGORY
STM. GEN. # 31	S1	VC	48'	3, 2
	S2	VC	48'	3, 2
	S6	VC	90'	3, 2
	S6	VC	92'	3, 2
	S8	VC	90'	3, 2
	S8	VC	92'	3, 2
STM. GEN. # 32	S1	VC	48'	3, 2
	S3	VC	48'	3, 2
	S4	VC	90'	3, 2
	S4	VC	92'	3, 2
	S8	VC	90'	3, 2
	S8	VC	92'	3, 2
STM. GEN. # 33	S1	VC	48'	3, 2
	S3	VC	48'	3, 2
	S5	VC	90'	3, 2
	S5	VC	92'	3, 2
	S8	VC	90'	3, 2
	S8	VC	92'	3, 2
STM. GEN. # 34	S1	VC	48'	3, 2
	S2	VC	48'	3, 2
	S7	VC	90'	3, 2
	S7	VC	92'	3, 2
	S8	VC	90'	3, 2
	S8	VC	92'	3, 2

TABLE 3.13-1 (SHEET 8 OF 8)

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

NOTES:

(1) Location: AFB - Auxiliary Boiler Feed Pump Building and Pipe Bridge Area

PAB - Primary Auxiliary Building
VC - Containment Building

(2) Categories:

1. Snubber in high radiation area during shutdown.*
2. Snubber especially difficult to remove - (Because of size and location).
3. Snubber inaccessible during normal operation.* (Because of high radiation and / or temperature environment).
4. Snubber accessible during normal operation. *

* Modifications to this table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

4.11 SHOCK SUPPRESSORS (SNUBBERS)

Applicability

Applies to the inspection and testing of all hydraulic snubbers listed in Table 3.13-1.

Objective

To verify that snubbers will perform their design functions in the event of a seismic or other transient dynamic event.

Specification

1. All hydraulic snubbers whose seal material has been demonstrated by operating experience, laboratory testing, or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
≥ 8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers are categorized in Table 3.13-1 as accessible or inaccessible during reactor operation. These two groups may be

inspected independently according to the above schedule.

2. All hydraulic snubbers whose seal materials have not been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.
3. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.11.1, it shall be assumed that the facility had been on a 6-month inspection interval.
4. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lock up and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lb need not be functionally tested.

Basis

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include inspection of the hydraulic fluid reservoir, fluid connections, and linkage connections to piping and structures. The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for

the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule. Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance.

Snubbers containing seal material which has not been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed.

To further increase the assurance of snubber reliability functional tests will be performed once each refueling cycle. These tests will include the stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Those snubbers designated in Table 3.13-1 as being in high radiation areas or especially difficult to remove need not be selected for functional tests. Snubbers of rated capacity greater than 50,000 lb are exempt from the functional testing requirements because of the impracticality of testing such large units.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 6 TO LICENSE NO. DPR-64

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

INTRODUCTION

During the summer of 1973, inspections at two reactor facilities revealed a high incidence of inoperable hydraulic shock suppressors (snubbers) manufactured by Bergen Paterson Pipesupport Corporation. As a result of those findings, the Office of Inspection and Enforcement required each operating reactor licensee to immediately inspect all Bergen Paterson snubbers utilized on safety systems and to reinspect them 45 to 90 days after the initial inspection. Snubbers supplied by other manufacturers were to be inspected on a lower priority basis.

By letter dated November 8, 1973, we specified continuing surveillance requirements for snubbers at Indian Point Unit No. 2 and requested that Con Ed submit proposed Technical Specifications for a snubber surveillance program. We provided model technical specifications for snubber surveillance by letters dated July 8, 1975, and December 23, 1975. Con Ed submitted proposed technical specifications for Unit 2 by letter dated August 25, 1975, as superseded by letter dated September 2, 1976. By letter dated October 5, 1976, we requested that Con Ed submit proposed Technical Specifications for Indian Point Unit No. 3 consistent with model Technical Specifications. Con Ed and the Power Authority of the State of New York (the licensees) submitted proposed snubber Technical Specifications for Unit No. 3 by letter dated November 23, 1976.

EVALUATION

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal movement during startup and shutdown.

The consequence of an inoperable snubber is an increase in the probability of structural damage to piping resulting from a seismic or other postulated event which initiates dynamic loads. It is, therefore, necessary that snubbers installed to protect safety system piping be operable during reactor operation and be inspected at appropriate intervals to assure their operability.

Examination of defective snubbers at reactor facilities has shown that the high incidence of failures observed in the summer of 1973 was caused by severe degration of seal materials and subsequent leakage of the hydraulic fluid. The basic seal materials used in Bergen Paterson snubbers were two types of polyurethane; a millable gum polyester type containing plasticizers and an unadulterated molded type. Material tests performed at several laboratories (Reference 1) established that the millable gum polyurethane deteriorated rapidly under the temperature and moisture conditions present in many snubber locations. Although the molded polyurethane exhibited greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. The investigation indicated that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

An extensive seal replacement program has been carried out at many reactor facilities, including Indian Point Unit No. 3. Experience with ethylene propylene seals has been very good with no serious degradation reported thus far. Although the seal replacement program has significantly reduced the incidence of snubber failures, some failures continue to occur. These failures have generally been attributed to faulty snubber assembly and installation, loose fittings and connections and excessive pipe vibrations. The failures have been observed in both PWR's and BWR's and have not been limited to units manufactured by Bergen Paterson. Because of the continued incidence of snubber failures, we have concluded that snubber operability and surveillance requirements should be incorporated into the Technical Specifications. We have further concluded that these requirements should be applied to all safety related snubbers, regardless of manufacturer, in all light water cooled reactor facilities.

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(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

The proposed Technical Specifications provide additional assurance of satisfactory snubber performance and reliability. The specifications require that snubbers be operable during reactor operation and prior to startup. Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repair or replacement of defective units before the reactor must be shut down. The licensee will be expected to commence repair or replacement of a failed snubber expeditiously. However, the allowance of 72 hours is consistent with that provided for other safety-related equipment and provides for remedial action to be taken in accordance with 10 CFR50.36(c)(2). Failure of a pipe, piping system, or major component would not necessarily result from the failure of a single snubber to operate as designed, and even a snubber devoid of hydraulic fluid would provide support for the pipe or component and reduce pipe motion. The likelihood of a seismic event or other initiating event occurring during the time allowed for repair or replacement is very small. Considering the large size and difficult access of some snubber units, repair or replacement in a shorter time period is not practical. Therefore, the 72 hour period provides a reasonable and realistic period for remedial action to be taken.

An inspection program is specified to provide additional assurance that the snubbers remain operable. The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The longest inspection interval allowed in the Technical Specifications after a record of no snubber failures has been established in nominally 18 months. Experience at operating facilities has shown that the required surveillance program should provide an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. Snubbers containing seal material which has not been demonstrated to be compatible with the operating environment are required to be inspected every 31 days until the compatibility is established or an appropriate seal change is completed.

To further increase the level of snubber reliability, the proposed Technical Specifications require functional tests once each refueling cycle. The tests will verify proper piston movement, lock up and bleed.

We have concluded that the proposed Technical Specifications, as modified, increase the probability of successful snubber performance, increase reactor safety and we therefore find them acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 4, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 6 to Facility Operating License No. DPR-64, issued to Consolidated Edison Company of New York, Inc. and the Power Authority of the State of New York (the licensees), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to establish requirements for inspection and testing of shock suppressors (snubbers) at Indian Point Unit No. 3.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment transmitted by letter dated November 23, 1976, (2) Amendment No. 6 to License No. DPR-64, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Hendrick Hudson Free Library, 31 Albany Post Road, Montrose, New York.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of January 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors