12/30/76

Docket No.: 50-219

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Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Ave. at Punch Bowl Rd.
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 18 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications and is in response to your application dated September 12, 1975, as supplemented by letter dated June 16, 1976.

The amendment consists of changes in the Technical Specifications that will add Sections 3.5.A.7 and 4.5.Q by describing the Limiting Conditions for Operation and Surveillance Requirements for Safety-related snubbers.

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

Sincerely,

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

Enclosures:
1. Amendment No. 18 to License DPR-16
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

Const/

Contingent on word "soon" in 3.5.A.7. b being changed to "sooner".

OFFICE	ORB#3	ORB#3	OELD	ORB#3		
SURNAME	CParrish	JShea:acr	[signature]	GLear		
DATE	12/ /76	12/28/76	12/30/76	12/30/76		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No.18
License No. DPR-16

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

   A. The application for amendment by Jersey Central Power & Light Company (the licensee) dated September 12, 1975, as supplemented by letter dated June 16, 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 a change 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

James J. Shea

for George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: December 30, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 18

TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
3.5-3	3.5-3
3.5-6	3.5-3a
3.5-7	3.5-6
4.5-6a	3.5-7
4.5.9b	3.5-8
	3.5-9
	3.5-10
	3.5-11
	3.5-12
	3.5-13
	4.5-6a
	4.5-6a-1
	4.5.9b

b. Two of the fourteen suppression chamber - drywell vacuum breakers may be inoperable provided that they are secured in the closed position.

c. One position alarm circuit for each operable vacuum breaker may be inoperable for up to 15 days provided that each operable suppression chamber - drywell vacuum breaker with one defective alarm circuit is physically verified to be closed immediately and daily during this period.

5. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5.0% O_2 with nitrogen gas within 24 hours after the reactor mode selector switch is placed in the run mode. Primary containment deinerting may commence 24 hours prior to a scheduled shutdown.

6. If specifications 3.5.A.1.a, b, c(1) and 3.5.A.2 through 3.5.A.5 cannot be met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

7. Shock Suppressors (Snubbers)

   a. During all modes of operation except cold shutdown and refuel, all safety related snubbers listed in Table 3.5.1 shall be operable except as noted 3.5.A.7.b, c and d below.

   b. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

   c. If the requirements of 3.5.A.7.a and 3.5.A.7.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

   d. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

   e. Snubbers may be added to safety related systems without prior License Amendment to Table 3.6.1 provided that a revision to Table 3.6.1 is included with the next License Amendment request.

B. Secondary Containment

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.2.A is met.

   b. The reactor is in the cold shutdown condition.

   c. The reactor vessel head or the drywell head are in place.

d. No work is being performed on the reactor or its connected systems in the reactor building.

e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials.

2. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.3.

not reduce the margin of safety below that considered adequate and is judged prudent in terms of the added plant safety offered by the opportunity for leak inspection. The 24-hour time to provide inerting is judged to be a reasonable time to perform the operation and establish the required O_2 limit.

Snubbers are designed 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 as a result of a seismic or other event initiating 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.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 11) has shown that millable gum polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to define precisely an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

Because **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 shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification **3.5.A.7.d prohibits startup with inoperable snubbers.**

Secondary containment[5] is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary **containment** is an integral part of the overall **containment** system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations and whenever work is being performed on the reactor or its connected systems in the reactor building since their operation could result in inadvertent release of radioactive material.

The standby gas treatment system(6) filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs.

Two separate filter trains are provided each having 100% capacity.(6) If one filter train becomes inoperable, there is no immediate threat to secondary containment and reactor operation may continue while repairs are being made. Since the test interval for this system is one month (Specification 4.5), the time out-of-service allowance of 7 days is based on considerations presented in the Bases in Specification 3.2 for a one-out-of-two system.

References:
(1) FDSAR, Volume I, Section V-1
(2) FDSAR, Volume I, Section V-1.4.1
(3) FDSAR, Volume I, Section V-1.7
(4) Licensing Application, Amendment 11, Question III-25
(5) FDSAR, Volume I, Section V-2
(6) FDSAR, Volume I, Section V-2.4
(7) Licensing Application, Amendment 42
(8) Licensing Application, Amendment 32, Question 3
(9) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.
(10) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.
(11) Report H. R. Erickson, Bergen-Paterson to K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Arrestors.

Table 3.5.1

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Snubbers In High Radiation Area During Shut Down	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
N-1-1	North Main Steam	23'	X		X	
N-1-2	North Main Steam	23'	X		X	
N-1-3	North Main Steam	51'	X		X	
N-1-4	North Main Steam	51'	X		X	
N-1-5	North Main Steam	51'	X		X	
N-1-6	North Main Steam	51'	X		X	
N-1-7	North Main Steam	60'	X		X	
N-2-1	North Feedwater	23'	X		X	
N-2-2	North Feedwater	23'	X		X	
N-2-3	North Feedwater	51'	X		X	
N-2-4	North Feedwater	51'	X		X	
N-2-5	North Feedwater	51'	X		X	
N-2-6	North Feedwater	51'	X		X	
N-2-7	North Feedwater	51'	X		X	
N-2-8	North Feedwater	51'	X		X	
S-1-1	South Main Steam	23'	X		X	
S-1-2	South Main Steam	23'	X		X	
S-1-3	South Main Steam	51'	X		X	
S-1-4	South Main Steam	51'	X		X	
S-1-5	South Main Steam	51'	X		X	
S-1-6	South Main Steam	51'	X		X	
S-1-7	South Main Steam	60'	X		X	
S-2-1	South Feedwater	23'	X		X	
S-2-2	South Feedwater	23'	X		X	
S-2-3	South Feedwater	51'	X		X	
S-2-4	South Feedwater	51'	X		X	
S-2-5	South Feedwater	51'	X		X	
S-2-6	South Feedwater	51'	X		X	
S-2-7	South Feedwater	51'	X		X	
S-2-8	South Feedwater	51'	X		X	

Table 3.5.1

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Snubber In High Radiation Area During Shut Down	Snubber Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
N-14-1	Emergency Condenser Condensate Return	75'	X		X	
N-14-2	Emergency Condenser Condensate Return	75'	X		X	
N-14-3	Emergency Condenser Condensate Return	95'	X		X	
N-14-4	Emergency Condenser Condensate Return	95'	X		X	
N-14-5	Emergency Condenser Condensate Return	95'	X		X	
N-14-6	Emergency Condenser Condensate Return	95'	X		X	
S-14-1	Emergency Condenser Condensate Return	60'	X		X	
S-14-2	Emergency Condenser Condensate Return	60'	X		X	
S-14-3	Emergency Condenser Condensate Return	95'	X		X	
S-14-4	Emergency Condenser Condensate Return	95'	X		X	
S-14-5	Emergency Condenser Condensate Return	95'	X		X	
S-14-6	Emergency Condenser Condensate Return	95'	X		X	
16-1	Cleanup	60'	X		X	
16-2	Cleanup	51'	X		X	
16-3	Cleanup	55'	X		X	
16-4	Cleanup	65'	X		X	

Amendment No. 18

Table 3.5.1

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Snubber In High Radiation Area During Shut Down	Snubber Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
N-20-1	North Core Spray	51'	X		X	
N-20-2	North Core Spray	51'	X		X	
N-20-3	North Core Spray	75'	X		X	
N-20-4	North Core Spray	75'	X		X	
S-20-1	South Core Spray	90'		X	X	
S-20-2	South Core Spray	95'	X		X	
S-20-3	South Core Spray	95'		X	X	
N-E-1	North Electromatic Relief	51'	X		X	
N-E-2	North Electromatic Relief	51'	X		X	
S-E-1	South Electromatic Relief	51'	X		X	
S-E-2	South Electromatic Relief	51'	X		X	
S-E-3	South Electromatic Relief	51'	X		X	
21-1	Containment Spray	60'		X	X	
1	Containment Spray	-19'				X
2	Containment Spray	-19'				X
3	Containment Spray	-19'				X
4	Containment Spray	-19'				X
5	Containment Spray	-19'				X
6	Outside torus Cont. Spray	-19'				X
7	Outside torus Cont. Spray	-19'				X
8	Outside torus Cont. Spray	-19'				X
9	Outside torus Cont. Spray	-19'				X
10	Outside torus Cont. Spray	-19'				X
11	Outside torus Cont. Spray	-19'				X
12	Outside torus Cont. Spray	-19'				X
13	Outside torus Cont. Spray	-19'				X
14	Outside torus Cont. Spray	-19'				X
15	Outside torus Cont. Spray	-19'				X
16	Outside torus Cont. Spray	-19'				X
17	Outside torus Cont. Spray	-19'				X

Table 3.5.1

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Snubber In High Radiation Area During Shut Down	Snubber Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
18	Core Spray	-19'		X		X
19	Core Spray	-19'				X
1	Core Spray	20'				X
2	Core Spray	20'				X
3	Cont.. Spray	20'		X		X
4	Cont. Spray	20'		X		X
5	Core Spray	20'		X		X
6	Core Spray	20'		X		X
1	Core Spray	23'				X
2	Core Spray Pump	23'				X
3	Cont. Spray	23'				X
4	Cont. Spray	23'				X
5	Cont. Spray	23'				X
6	Cont. Spray	23'				X
7	Cont. Spray	23'		X		X
1	Cont. Spray	51'				X
2	Cont. Spray	51'				X
3	Cont. Spray	51'				X
4	Cont. Spray	51'				X
5	Core Spray	51'				X
6	Core Spray	51'				X
7	Core Spray	51'				X
8	Core Spray	51'				X
9	Core Spray	51'				X
10	Core Spray	51'				X
21	Core Spray	51'				X
22	Core Spray	51'				X
23	Core Spray	51'				X
24	Core Spray	51'				X

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Snubber In High Radiation Area During Shut Down	Snubber Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
1	Core Spray	75'				X
2	Core Spray	75'				X
3	Core Spray	75'				X
4	Core Spray	75'				X
5	Core Spray	75'				X
6	B. Emer. Cond.	75'				X
7	A. Emer. Cond.	75'				X
8	A. Emer. Cond.	75'				X
9	B. Emer. Cond.	75'				X
10	A. Emer. Cond.	75'				X
11	A. Emer. Cond.	75'				X
12	A. Emer. Cond.	75'				X
13	B. Emer. Cond.	75'				X
14	A. Emer. Cond.	75'				X
15	B. Emer. Cond.	75'				X
16	A. Emer. Cond.	75'				X
17	A. Emer. Cond.	75'				X
18	A. Emer. Cond.	75'				X
19	A. Emer. Cond.	75'				X
20	A. Emer. Cond.	75'				X
21	B. Emer. Cond.	75'				X
22	A. Emer. Cond.	75'				X
23	A. Emer. Cond.	75'				X
24	A. Emer. Cond.	75'				X
25	B. Emer. Cond.	75'				X
1	A. Emer. Cond.	95'				X
2	A. Emer. Cond.	95'		X		X
3	A. Emer. Cond.	95'		X		X
4	A. Emer. Cond.	95'				X
5	B. Emer. Cond.	95'				X
6	B. Emer. Cond.	95'				X
7	B. Emer. Cond.	95'				X
8	B. Emer. Cond.	95'				X
9	B. Emer. Cond.	95'				X

Table 3.5.1

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Snubbers In High Radiation Area During Shut Down	Sunbbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
17-1	Shutdown Cooling	48'	X		X	
17-2	Shutdown Cooling	51'	X		X	
17-3	Shutdown Cooling	51'	X		X	
17-4	Shutdown Cooling	51'	X		X	
17-5	Shutdown Cooling	51'	X		X	
17-6	Shutdown Cooling	51'	X		X	
51-7	Shutdown Cooling	51'				X
51-8	Shutdown Cooling	51'				X
51-9	Shutdown Cooling	51'				X
51-10	Shutdown Cooling	51'				X
51-11	Shutdown Cooling	51'				X
51-12	Shutdown Cooling	51'				X
51-13	Shutdown Cooling	51'				X
51-14	Shutdown Cooling	51'				X
51-15	Shutdown Cooling	51'				X
51-16	Shutdown Cooling	51'				X
51-17	Shutdown Cooling	51'				X
51-18	Shutdown Cooling	51'				X
51-19	Shutdown Cooling	51'				X
51-20	Shutdown Cooling	51'				X

If radiation levels in the vicinity of snubbers change, appropriate modifications to this Table should be submitted to NRC as an attachment to any subsequent license amendment.

P. Suppression Chamber Surveillance

1. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

2. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed until the heat addition is terminated.

4. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160F or above while the reactor primary coolant system pressure is greater than 180 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.

Q. Shock Suppressors (Snubbers)

1. All hydraulic snubbers listed in Table 3.5.1 whose seal material has been demonstrated by operating experience, lab 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 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 may be categorized in two groups, "Accessible" or "Inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. The initial inspection shall be performed within 12 months from the date of issuance of these specifications. For the purpose of entering the schedule into specification 4.5.Q.1, it shall be assumed that the facility had been on a 12 month inspection schedule.

3. 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 except when in the shutdown or refuel mode.

4. Once each refueling cycle, a representative sample of 10 hydraulic snubbers or approximately 10% of the hydraulic 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 10 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.

of the sys⌣m. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the test also gives an indication of the relative efficiency of the installed system. The test procedure is an adaptation of test procedures developed at the Savannah River Laboratory which were described in the Ninth AEC Air Cleaning Conference.*

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP at testing medium.

If laboratory tests for the adsorber material in one circuit of the Standby Gas Treatment System are unacceptable, all adsorber material in that circuit shall be replaced with adsorbent qualified according to Regulatory Guide 1.52. Any HEPA filters found defective shall be replaced with those qualified with Regulatory Position C.3.d of Regulatory Guide 1.52.

The snubber 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 provided that the seal materials are compatible with the operating environment.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests 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 should be performed once each refueling cycle. These tests will include 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 of these samples should require testing of additional units. Snubbers in high radiation areas or those especially difficult to remove (see Table 3.5.1) need not be selected for functional tests provided operability was previously verified. **Snubbers of rated capacity greater than 50,000 lb. are exempt from the functional testing requirements because of the inpracticability of testing such large units.**

*D. R. Muhbaier, "In Place Nondestructive Leak Test for Iodine Absorbers," Proceedings of the Ninth AEC Air Cleaning Confernece, USAEC Report CONF-660904, 1966.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.18 TO LICENSE NO. DPR-16

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

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.

Since a long term solution to eliminate recurring failures was not immediately available, the Division of Reactor Licensing sent a letter dated October 1, 1973, specifying continuing surveillance requirements for snubbers at Oyster Creek Unit No. 1 and requested that Jersey Central Power & Light Company submit proposed Technical Specifications for a snubber surveillance program. We provided model technical specifications for snuber surveillance by letters dated July 8, 1975, and December 24, 1975. Jersey Central Power & Light Company submitted proposed technical specifications by letter dated September 12, 1975, as supplemented by letter dated June 16, 1976. During our review, we found that certain modifications were necessary. These modifications were discussed with the licencee on *November 15, 1976 and have been included in the proposed Technical Specifications. The proposed change adds new Sections 3.5.A.7 and 4.5.Q by describing the Limiting Conditions for Operation and Surveillance Requirements for safety-related snubbers.

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 structral damage in 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 degradation 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 installation.

An extensive seal replacement program has been carried out at many reactor facilities, including Oyster Creek Unit No. 1. 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 PWRs and BWRs 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.

We have developed the attached Technical Specifications and Bases to provide additional assurance of satisfactory snubbers 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 license will be expected to commmence 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 CFR 50.36(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 is 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.

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

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

At the Oyster Creek Station, an extensive surveillane program on safety related snubbers has been in effect since July 1974.. The results and analysis for a 15 month period were submitted on October 20, 1976 and indicated no failed snubbers. Thus, we have concluded that the proposed Technical Specifications, as modified, increase the probability of successful snubber performance, increase reator safety and we therefore find them acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the 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, negative declaration, or environmental 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 changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do 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.

Date: December 30, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT

TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 18 to Provisional Operating License No. DPR-16 issued to Jersey Central Power & Light Company which revised Technical Specifications for operation of the Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey. The amendment is effective as of its date of issuance.

The amendment consist of changes to the Technical Specifications that will add Sections 3.5.A.7 and 4.5.Q by describing the Limiting Conditions for Operation and Surveillance Requirements for safety-related snubbers.

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 issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated September 12, 1975, as supplemented by letter dated June 16, 1976, (2) Amendment No. 18 to License No. DPR-16, 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 Ocean County Library, Brick Township Branch, 401 Chambers Bridge Road, Birck Town, New Jersey 08723.

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 30 day of December 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

James J. Shea

James J. Shea, Acting Chief
Operating Reactors Branch #3
Division of Operating REactors