November 26, 19

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: OYSTER CREEK - ISSUANCE OF AMENDMENT RE: 24-MONTH SURVEILLANCE EXTENSIONS TO SUPPORT 24-MONTH FUEL CYCLE (TAC NO. M96906)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated October 10, 1996, as supplemented March 25, June 6, and August 29, 1997 (TSCR 203).

The amendment extends the instrumentation surveillances for the condenser low vacuum, high temperature main steamline tunnel, recirculation flow, and reactor coolant leakage. Additionally, the change extends the equipment test/operability checks for containment vent and purge isolation, electromagnetic relief valve operability, and drywell to torus leakage test.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 193 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION: See attached page

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DATED: November 26, 1997

AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. DPR-16-OYSTER CREEK

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee), dated October 10, 1996, as supplemented March 25, June 6, and August 29, 1997, 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, and paragraph 2.C.(2) of Facility Operating License No. DPR-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 193 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

John A. Zwolinski, Deputy Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 26, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 193

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
4.1-4	4.1-4
4.1-5	4.1-5
4.1-7	4.1-7
4.3-2	4.3-2
4.4-1	4.4-1
4.5-4	4.5-4

TABLE 4.1.1 (Cont'd.)

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test and Calibration)
13. DELETED				
14. High Radiation in Reactor Building Operating Floor	1/s	1/3 mo	1/3 mo	Using gamma source for calibration
Ventilation Exhaust	1/s	1/3 mo	1/3 mo	
15. High Radiation on Air Ejector Off-Gas		1/3 mo	1/3 mo	Using built-in calibration equipment
	1/s			Channel Check
	1/mo			Source check
		1/24 mo		Calibration according to established station calibration procedures
			1/24 mo	Note a
16. IRM Level	N/A	Each startup	N/A	
IRM Scram	*	*	*	Using built-in calibration equipment
17. IRM Blocks	N/A	Prior to startup and shutdown	Prior to startup and shutdown	Upscale and downscale
18. Condenser Low Vacuum	N/A	**1/24 mo**	**1/24 mo**	
19. Manual Scram Buttons	N/A	N/A	1/3 mo	
20. High Temperature Main Steamline Tunnel	N/A	**1/24 mo**	Each refueling outage	Using heat source box

TABLE 4.1.1 (cont'd)

Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test and Calibration)
21. SRM	*	*	*	Using built-in calibration equipment
22. Isolation Condenser High Flow ΔP (Steam and Water)	N/A	1/3 mo	1/3 mo	By application of test pressure
23. Turbine Trip Scram	N/A	N/A	1/3 mo	
24. Generator Load Rejection Scram	N/A	1/3 mo	1/3 mo	
25. Recirculation Loop Flow	N/A	**1/24 mo**	N/A	By application of test pressure
26. Low Reactor Pressure Core Spray Valve Permissive	N/A	1/3 mo	1/3 mo	By application of test pressure
27. Scram Discharge Volume (Rod Block)				
a) Water level high	N/A	Each refueling outage	1/3 mo	Calibrate by varying level in sensor column
b) Scram Trip bypass	N/A	N/A	Each refueling outage	
28. Loss of Power				
a) 4.16 KV Emergency Bus Undervoltage (Loss of voltage)	1/d	1/24 mo	1/mo	
b) 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	1/d	1/24 mo	1/mo	

TABLE 4.1.2

MINIMUM TEST FREQUENCIES FOR TRIP SYSTEMS

Trip System	Minimum Test Frequency
1) Dual Channel (Scram)	Same as for respective instrumentation in Table 4.1.1
2) Rod Block	Same as for respective instrumentation in Table 4.1.1
3) DELETED	DELETED
4) Automatic Depressurization each trip system, one at a time	Each refueling outage
5) MSIV Closure, each closure logic circuit independently (1 valve at a time)	Each refueling outage
6) Core Spray, each trip system, one at a time	1/3 mo and each refueling outage
7) Primary Containment Isolation, each closure circuit independently (1 valve at a time)	Each refueling outage
8) Refueling Interlocks	Prior to each refueling operation
9) Isolation Condenser Actuation and Isolation, each trip circuit independently (1 valve at a time)	Each refueling outage
10) Reactor Building Isolation and SGTS Initiation	Same as for respective instrumentation in Table 4.1.1
11) Condenser Vacuum Pump Isolation	Prior to each startup
12) Air Ejector Offgas Line Isolation	Each refueling outage
13) Containment Vent and Purge Isolation	**1/24 mo**

4.4 EMERGENCY COOLING

Applicability: Applies to surveillance requirements for the emergency cooling systems.

Objective: To verify the operability of the emergency cooling systems.

Specification: Surveillance of the emergency cooling systems shall be performed as follows:

Item	Frequency
A. Core Spray System	
1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.
2. Motor operated valve operability	Once/month
3. Automatic actuation test	Every three months
4. Pump compartment water-tight doors closed	Once/week and after each entry
5. Core spray header ΔP instrumentation	
check	Once/day
calibrate	Once/3 months
test	Once/3 months
B. Automatic Depressurization	
1. Valve operability	Once every **24 months***
2. Automatic actuation test	Every refueling outage
C. Containment Cooling System	
1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.

*Valve operability shall be demonstrated at system operating pressure prior to exceeding 5 percent power, **following a refueling outage.**

* G. Primary Coolant System Pressure Isolation Valves Specification:

1. Periodic leakage testing [a] on each valve listed in Table 4.3.1 shall be accomplished prior to exceeding 600 psig reactor pressure every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, whenever the valve is moved whether by manual actuation or due to flow conditions, and after returning the valve to service after maintenance, repair or replacement work is performed.

H. Reactor Coolant System Leakage

1. Unidentified leakage rate shall be calculated at least once every 4 hours.

2. Total leakage rate (identified and unidentified) shall be calculated at least once every 8 hours.

3. A channel calibration of the primary containment sump flow integrator and the primary containment equipment drain tank flow integrator shall be conducted at least once per 24 months.

I. An inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC staff positions on schedule, methods, personnel, and sample expansion included in the generic letter or in accordance with alternate measures approved by the NRC staff.

Bases:

Data is available relating neutron fluence (E>1.0MeV) and the change in the Reference Nil-Ductility Transition Temperature (RT_{NDT}). The pressure-temperature (P-T) operating curves A, B, and C in Figures 3.3.1, 3.3.2, and 3.3.3 were developed based on the results of testing and evaluation of specimens removed from the vessel after 8.38 EFPY of operation. Similar testing and analysis will be performed throughout vessel life to monitor the effects of neutron irradiation on the reactor vessel shell materials.

The inspection program will reveal problem areas should they occur, before a leak develops. In addition, extensive visual inspection for leaks will be made on critical systems. Oyster Creek was designed and constructed prior to

[a] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

* NRC Order dated April 20, 1981.

(3) At least four of the suppression chamber - drywell vacuum breakers shall be inspected. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected such that Specification 3.5.A.5.a can be met.

(4) A drywell to suppression chamber leak rate test **shall be performed once every 24** months to demonstrate, that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of air flow through a 2-inch orifice.

G. Reactor Building

1. Secondary containment capability tests shall be conducted after isolating the reactor building and placing either Standby Gas Treatment System filter train in operation.

2. The tests shall be performed at least once per operating cycle (interval not to exceed 20 months) and shall demonstrate the capability to maintain a ¼ inch of water vacuum under calm wind conditions with a Standby Gas Treatment System Filter train flow rate of not more than 4000 cfm.

3. A secondary containment capability test shall be conducted at each refueling outage prior to refueling.

4. The results of the secondary containment capability tests shall be in the subject of a summary technical report which can be included in the reports specified in Section 6.

H. Standby Gas Treatment System

1. The capability of each Standby Gas Treatment System circuit shall be demonstrated by:

   a. At least once per 18 months, after every 720 hours of operation, and following significant painting, fire, or chemical release in the reactor building during operation of the Standby Gas Treatment System by verifying that:

      (1) The charcoal absorbers remove $\geq$99% of a halogenated hydrocarbon refrigerant test gas and the HEPA filters remove $\geq$99% of the DOP in a cold DOP test when tested in accordance with ANSI N510-1975.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated October 10, 1996, as supplemented March 25, June 6, and August 29, 1997, the GPU Nuclear Corporation (GPUN, the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (OCNGS). The requested changes would extend the instrumentation surveillances for the condenser low vacuum, high temperature main steamline tunnel, recirculation flow, and reactor coolant leakage. Additionally, the change would extend the equipment test/operability checks for containment vent and purge isolation, electromagnetic relief valve operability, and drywell to torus leakage test. The supplemental letters provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

Generic Letter (GL) 91-04 provides guidance on the type of analysis and information required to justify a change in instrument calibration intervals. The licensee evaluated the effects of an increased calibration interval on the instrument uncertainties, equipment qualification, and vendor requirements to ensure that an extended surveillance meets the seven actions delineated in Enclosure 2 of GL 91-04.

2.0 EVALUATION

The licensee performed an analysis of the affected instrumentation systems to establish the basis for a 24-month + 25% (30 months maximum as allowed by TS) calibration frequency to verify that the surveillance interval extensions have a small effect on plant safety and would not invalidate any assumptions in the plant licensing basis. The instrumentation evaluations were based on statistical analysis using surveillance data to predict setpoint deviations at 30 months. The data was applied to a linear regression and t-distribution for confidence interval. In sample calculation C1302-640-5350-009, for condenser low vacuum instruments, GPUN justified the use of the regression mean model by

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indicating the regression analysis predicated a slope on the order of 10^{-4} for drift. The predicted confidence interval model accounted for variances of the predicted estimate, mean, and regression line slope. Graphically, the confidence belt diverged with time which is consistent with expected incremental increases in uncertainty as the model approached 30 months. The analysis predicted the setpoint deviations to be within as-found acceptance criteria at 24 months + 25% (30 months maximum) with a 95% confidence level. GPUN's evaluations for the effect of 24-month surveillance intervals for equipment tests were based on operating experience and historical surveillance data. To address GL 91-04 guidance to monitor and assess the effects of increased calibration surveillance intervals on instrument drift and its effect on safety, licensee plant procedures require a deviation report to be generated if the setpoint is not within the as-found acceptance criteria. Deviation reports are resolved in accordance with plant procedures, which may require root-cause evaluation, trending, or corrective action as appropriate.

2.1 Condenser Low Vacuum

Proposed Change: TS Table 4.1.1, Item 18
The main condenser low vacuum instrument channels TS currently requires calibration once every 20 months. The proposed amendment would extend the test interval to 24 months + 25% (30 months maximum).

The channels consist of four limit switches that monitor condenser vacuum and initiate a scram signal to the reactor protection system on a low vacuum condition. The low condenser scram signal also serves as a backup to the scram signal generated by a turbine trip. Using statistical analysis consistent with GL 91-04 guidance, the licensee determined the predicted instrument performance at 30 months has a 95% confidence interval that the setpoint will be within the existing as-found acceptance criteria. The staff finds the proposed TS change consistent with the guidance provided in Enclosure 2 of GL 91-04 and, therefore, acceptable.

2.2 Main Steamline Tunnel High Temperature

Proposed Change: TS Table 4.1.1, Item 20
The main steamline tunnel instrumentation channels TS currently requires calibration once every 20 months. The proposed amendment would extend the test interval to 24 months + 25% (30 months maximum).

A steam tunnel high temperature condition is indicative of a main steamline break. The instrument channels provide monitoring in the main steamline tunnel and initiate a signal to close the MSIVs [main steam isolation valves] upon a high temperature condition. The temperature switches in the instrument channels were replaced during the second quarter of 1991 and therefore, insufficient plant data was available for a statistically valid drift calculation. For that reason, the drift analysis was performed using vendor data for intervals greater than 30 months. The use of vendor data is consistent with the guidance provided in GL 91-04. The licensee determined in accordance with GL 91-04 guidance that deviation of instrument setpoints due to drift over a 30-month period is very small and the instruments will perform their required safety function at 95% confidence for the maximum setpoint drift. The staff finds the proposed TS change consistent with the guidance provided in Enclosure 2 of GL 91-04 and, therefore, acceptable.

2.3 Recirculation Loop Flow

Proposed Change: TS Table 4.1.1, Item 25
The recirculation flow monitoring system TS currently requires calibration by application of a test pressure once every 20 months. The proposed amendment would extend the test interval to 24 months + 25% (30 months maximum).

The recirculation flow monitoring instrumentation provides a rod block signal if the total recirculation flowrate exceeds 100% rated flow. In addition, a total flow signal is provided to the average power range monitors (APRMs). The APRMs provide a rod block signal based on core neutron flux levels. The rod block setpoint is varied with recirculation flow. The APRM flow-biased rod block is designed to terminate rod withdrawal errors prior to reaching power levels that could cause cladding damage. As part of a modification to replace the flow transmitters and the electronics, the licensee performed a calculation to determine inaccuracies associated with the trip functions. The calculation was based on manufacturer specifications and assumed a 30-month calibration interval. In accordance with GL 91-04 guidance, manufacturer error data was converted to two sigma values which resulted in a 95% confidence with 95% probability that the instruments will perform within the as-found acceptance criteria. The staff finds the proposed TS change consistent with the guidance provided in Enclosure 2 of GL 91-04 and, therefore, acceptable.

2.4 Reactor Coolant System Leakage

Proposed Change: TS 4.3.H., Item 3
The reactor coolant system leakage instrumentation TS currently requires calibration of the primary containment drywell floor sump and equipment drain tank flow integrators once every 18 months. The proposed amendment would extend the test interval to 24 months + 25% (30 months maximum).

The drywell floor sump is used to determine unidentified reactor coolant leak rates, and the equipment drain tank flow provides the identified primary reactor coolant leak rates. The two equipment drain tank flow integrator channels consist of transmitters, signal convertors, square root integrator, totalizer, and indicator. Surveillance data from January 1988 to July 1993 for the drywell floor sump instrument loop indicated no setpoints exceeded the as-found acceptance criteria. Similarly, surveillance data from January 1988 to July 1992 for the equipment drain tank flow instrument loop indicated no setpoints exceeded the as-found acceptance criteria. The licensee statistical analysis of the equipment drain tank transmitter and flow integrator predicted deviation for a 30-month surveillance interval was significantly lower than the acceptance criteria. In accordance with GL 91-04 guidance, the licensee determined that the instrument performance at 30 months has a 95% confidence with 95% probability that the instruments will perform within the existing as-found acceptance criteria. The staff finds the proposed TS change consistent with the guidance provided in Enclosure 2 of GL 91-04 and, therefore, acceptable.

2.5 Containment Vent and Purge Isolation

Proposed Change: TS Table 4.1.2, Item 13
Currently, the OGNGS TSs specify that a test of the containment vent and purge isolation trip system shall be performed once every 20 months. The change proposed by the licensee would extend the test interval to 24 months + 25% (30 months maximum).

The sensors that signal closure of the containment purge and isolation system valves are calibrated in accordance with TS Table 4.1.1, Item 29. The system logic consists of test switches and solenoids. The licensee reviewed surveillance test results for a 5-year period and found no failures of the system or subject valves to initiate and perform their intended functions. The staff also did not identify any containment vent and purge valve operability concerns. Based upon past satisfactory performance history of required surveillance testing of the subject containment vent and purge isolation system valves, the staff finds the proposed TS change acceptable.

2.6 Electromatic Relief Valve Operability

Proposed Change: TS 4.4.B., Item 1
Currently, the TS requires a demonstration of automatic depressurization system (ADS) valve operability, at system operating pressure, prior to exceeding 5% power, following a refueling outage and on an interval not to exceed 20 months. The change proposed by the licensee would extend the demonstration to 24 months + 25% (30 months maximum).

The licensee reviewed surveillance test results for a 5-year period and found no failures of the subject valves to operate when called upon using the 20-month interval testing. The licensee also reviewed the operating history of functional tests of the ADS valves. The ADS valve functional test is done in two parts: (1) a manual operability test is performed during the refueling outage, and (2) an automatic actuation test is performed for the valves' operators during the refueling outage. The ADS valve operability test is performed manually by the operator using a switch in the control room. The automatic actuation test verifies the valve's operator functional capability separately. In addition, prior to restart, valve operability is verified again with steam at low power. The staff did not identify any valve operability concerns associated with this request. Based on past performance history of required surveillance testing of the subject valves, the staff finds the proposed change acceptable.

2.7 Drywell-to-Torus Leak Rate Test

Proposed change: TS 4.5.5, Item 5.B.4
TSs require periodic suppression pool bypass leakage rate tests to be conducted at intervals not to exceed 20 months. The proposed amendment would extend the test interval to 24 months + 25% (30 months maximum).

A review by the licensee of test results from 1977 to present indicates that there has been no degradation of leak tightness over the period. In addition, quarterly leak rate tests at power have been performed since March 1990 and the acceptance criteria have been consistently met. The vacuum breakers are also stroke tested monthly. Based on the performance history and the operability assurance provided by the stroke tests, it is the licensee's position that the requested test interval extension is acceptable.

The Oyster Creek facility has a Mark I pressure-suppression containment consisting of a drywell which houses the NSSS, and a suppression chamber which contains a pool of cool water for pressure suppression. In this type of containment, the steam released by a recirculation line break or steam line break into the drywell is directed by a vent system to the suppression pool where it is discharged under water and condensed. The noncondensible gases that were initially in the drywell pass through the pool and are compressed into the suppression chamber airspace above the pool. The condensation of the steam limits the mass and energy addition to the containment atmosphere and thereby prevents the containment from exceeding its design pressure. The pressure increase that does occur will be due to the effects of the noncondensible gases that were initially present being forced into the suppression chamber airspace. The noncondensible gases are later returned to the drywell when the drywell pressure drops and the drywell vacuum breakers open.

The pressure suppression concept thus relies on the need for the steam to be condensed in the suppression pool. If a sufficient portion of the steam that is discharged into the drywell by a loss-of-coolant accident (LOCA) or MSLB [main steamline break] bypasses the pool and goes directly to the suppression chamber airspace without being in contact with the pool water, the pressure-suppression function is threatened and the containment could be overpressurized and fail. The containment, because of its relatively small size, can accommodate only a limited amount of such bypass. The purpose of the drywell-to-torus (or "suppression pool bypass") leakage test is to assure that such leakage would be less than the amount that can be accommodated. The amount of bypass leakage that can be accommodated at Oyster Creek is that amount corresponding to a 10.5 square inch leakage area (which is equivalent to one vacuum breaker disk being off its seat by 0.371 inch). As a conservative test acceptance criterion, the leakage rate is limited to that of a 2-inch orifice (information from TS Bases). It is noted that the suppression chamber spray system would enable bypass steam to be condensed. However, the spray system is not an automatic, safety-grade system and is therefore not credited in containment analysis for such purpose.

Potential bypass leakage paths that will be detected by the bypass leakage test include: (1) the torus-to-drywell vacuum breaker valve seats, (2) cracks in the portion of the steam discharge vent piping in the suppression chamber that is above the water level, and (3) isolation valve leakage in piping that is external to the containment that connects the drywell and suppression chamber (e.g., vent/purge/nitrogen supply piping). Because the steam vent piping is not likely to develop significant cracking in a short period of time (although long-term cracking has been experienced as described in Information Notice 85-99, "Cracking in Boiling-Water-Reactor Mark I and Mark II Containments Caused by Failure of the Inerting System"), and because isolation valves are individually tested on a periodic basis, the vacuum breaker valve seats are considered the most significant leakage path with respect to the proposed test interval extension. The vacuum breakers, of which there are 14 installed, provide a means for noncondensible gases that are compressed into the suppression chamber airspace during the blowdown phase of a LOCA to return to the drywell and thereby equalize the drywell and suppression chamber pressures. During blowdown, the vacuum breaker disks are seated. Any steam leakage past the seat during blowdown will bypass the suppression pool, going directly to the airspace. However, as stated above, considerable testing has been successfully performed that indicates that vacuum breakers are not subject to rapid deterioration in leak tightness. Also, the operability (i.e., monthly "stroke") tests provide a high degree of assurance that vacuum breaker valve disks are properly seated, since the vacuum breakers are provided with sensitive disk position indicators. Based on the relatively long history of satisfactory leakage test results, and the additional assurance of proper disk position that is provided by the relatively frequent operability tests, the extension of the test interval to 24 months (+ 25%) is acceptable. A staff review of Oyster Creek records indicates that problems were experienced with the vacuum breakers very early in plant life [Ref: letter from JCP&LC to R. Schemel dated October 8, 1973], due to "growth" of teflon bushings, however, based on lack of subsequent failure reports, this problem appears to have been corrected.

Based on the above, the staff finds that the licensee has performed analyses of system operating performance and setpoint drift effects in accordance with the guidance of GL 91-04 which confirm satisfactory operation is predicted for the main condenser low vacuum, main steam line tunnel high temperature, reactor recirculation flow, reactor coolant system leakage, containment vent and purge isolation, electromatic relief valve instrumentation systems, and the drywell-to-torus leak rate test over the proposed 24-month (+ 25%) surveillance interval. The staff, therefore, concludes that the associated proposed TS changes for an extension of the surveillance interval to 24 months (+ 25%)are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 57485). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR

51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Bryant

Date: November 26, 1997