September 3, 1996

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: IMPLEMENTATION OF 10 CFR PART 50, APPENDIX J, OPTION B (TAC NO. M94855)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 186 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated July 17, 1996, as supplemented August 28, 1996 (TSCR 242, Rev. 2). The July 17, 1996, application supersedes your previous applications of February 23 and June 19, 1996.

The amendment changes the Technical Specifications to allow the implementation of 10 CFR Part 50, Appendix J, Option B.

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-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 186 to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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

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cc w/encls: See next page

M. Roche
GPU Nuclear Corporation

Oyster Creek Nuclear
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cc:

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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. 186
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 July 17, 1996, as supplemented August 28, 1996, 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. 186, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 186

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
i	i
ii	ii
iii	iii
1.0-5	1.0-5
3.5-3	3.5-3
4.5-1	4.5-1
4.5-2	4.5-2
4.5-3	4.5-3
4.5-4	4.5-4
4.5-5	4.5-5
4.5-6	4.5-6
4.5-7	4.5-7
4.5-8	4.5-8
4.5-9	4.5-9
4.5-10	4.5-10
4.5-11	4.5-11
4.5-12	4.5-12
4.5-12a	--
4.5-13	4.5-13
4.5-14	4.5-14
4.5-15	4.5-15
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4.5-19	--
6-16	6-16

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*Issued by NRC Order dated 10-24-80

B. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

1.24 SURVEILLANCE REQUIREMENTS

Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met. Each surveillance requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. [1]

Surveillance requirements for systems and components are applicable only during the modes of operation for which the system or components are required to be operable, unless otherwise stated in the specification.

This definition establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance, e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a fuel cycle length surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for the surveillance that are not performed during refueling outages. The limitation of this definition is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

1.25 APPENDIX J TEST PRESSURE

For the purpose of conducting leak rate tests to meet 10 CFR 50 Appendix J, $P_a = 35$ psig.

1.26 FRACTION OF LIMITING POWER DENSITY (FLPD)

The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

1.27 MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

[1] For the 10 CFR 50 Appendix J Type A test, the 25% shall not exceed 15 months.

(1) Maintain at least one isolation valve operable in each affected penetration that is open and within 4 hours (48 hours for the traversing in-core probe system) either;

   (a) Restore the inoperable valve(s) to operable status OR

   (b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, OR

   (c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.

(2) An inoperable containment isolation valve of the shutdown cooling system may be opened with a reactor water temperature equal to or less than 350°F in order to place the reactor in the cold shutdown condition. The inoperable valve shall be returned to the operable status prior to placing the reactor in a condition where primary containment integrity is required.

b. If the primary containment air lock is inoperable, per specification 4.5.C.2, restore the inoperable air lock to operable status within the 24 hours or be in at least a shutdown condition within the next 12 hours and in cold shutdown within the following 24 hours.

4. Reactor Building to Suppression Chamber Vacuum Breaker System

a. Except as specified in Specification 3.5.A.4.b below, two reactor building to suppression chamber vacuum breakers in each line shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the air-operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall move from closed to fully open when subjected to a force equivalent of not greater than 0.5 psid acting on the vacuum breaker disc.

b. From the time that one of the reactor building to suppression chamber vacuum breaker is made or found to be inoperable, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is made operable sooner, provided that the procedure does not violate primary containment integrity.

4.5 CONTAINMENT SYSTEM

Applicability: Applies to containment system leakage rate, continuous leak rate monitor, functional testing of valves, standby gas treatment system operability, inerting surveillance, drywell coating surveillance, instrument line flow check valve surveillance, suppression chamber surveillance, and snubber surveillance.

Objectives: To verify operability of containment systems, and that leakage from the containment system is maintained within specified values, as outlined in Appendix J of 10 CFR 50.

Specification:

A. Primary Containment Leakage Testing

A Primary Containment Leakage Rate Testing Program shall be established to implement 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance Based Containment Leak Test Program", dated September 1995, as modified by the following exception:

1. The first Type A test required by this program will be performed during refueling outage 17R

B. Type A Primary Containment Integrated Leak Rate Test (PCILRT).

PCILRT shall be performed in accordance with the Primary Containment Leakage Rate Testing Program.

C. Type B and Type C Local Leak Rate Tests (LLRT)

1. LLRT shall be performed in accordance with the Primary Containment Leakage Rate Testing Program.
2. The Drywell Airlock, Drywell Airlock electrical penetration, and Drywell Airlock barrel seal shall be local leak rate tested in accordance with the Primary Containment Leakage Rate Testing Program.
   a. When containment integrity is required, the airlock must be tested at 10 psig within 7 days after each containment access. If the airlock is opened more frequently than once every 7 days, it may be tested at 10 psig once per 30 days during this time period.

b. If the airlock is opened during a period when Primary Containment is not required, it need not be tested while Primary Containment is not required, but must be tested at P_a prior to returning the reactor to an operating mode requiring Primary Containment Integrity.

D. Primary Containment Leakage Rates shall be limited to:

1. The maximum allowable Primary Containment leakage rate is 1.0 L_a. The maximum allowable Primary Containment leakage rate to allow for plant startup following a type A test is 0.75 L_a. The leakage rate acceptance criteria for the Primary Containment Leakage Rate Testing Program for Type B and Type C tests is ≤ 0.60 L_a at P_a.
2. The leakage rate acceptance criteria for an MSIV shall be 0.05(0.75) L_a at P_a.
3. The leakage rate acceptance criteria for the drywell airlock shall be ≤ 0.05 L_a when measured or adjusted to P_a.

E. Continuous Leak Rate Monitor

1. When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements.

2. This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

F. Functional Test of Valves

1. All containment isolation valves specified in Table 3.5.2 shall be tested for automatic closure by an isolation signal during each refueling outage. The following valves are required to close in the time specified below:

Main steam line isolation valves	$\geq$ 3 sec & $\leq$ 10 sec
Isolation condenser isolation valves	$\leq$ 60 sec
Cleanup system isolation valves	$\leq$ 60 sec
Cleanup auxiliary pumps system isolation valves	$\leq$ 60 sec
Shutdown system isolation valves	$\leq$ 60 sec

2. Each containment isolation valve shown in Table 3.5.2 shall be demonstrated operable prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel and verifying the specified isolating time. Following maintenance, repair or replacement work on the control or power circuit for the valves shown in Table 3.5.2, the affected component shall be tested to assure it will perform its intended function in the circuit.

3. Quarterly, during periods of sustained power operation, each main steam isolation valve shall be closed (one at a time) and its closure time verified to be within the limits of specification 4.5.F.1 above. Such testing shall be conducted with reactor power not greater than 50% of rated power.

4. Reactor Building to Suppression Chamber Vacuum Breakers

   a. The reactor building to suppression chamber vacuum breakers and associated instrumentation, including setpoint, shall be checked for proper operation every three months.

   b. During each refueling outage, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker from closed to fully open does not exceed the force specified in Specification 3.5.A.4.a. The air-operated vacuum breaker instrumentation shall be calibrated during each refueling outage.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic Operability Tests

      Once each month and following any release of energy which would tend to increase pressure to the suppression chamber, each operable suppression chamber - drywell vacuum breaker shall be exercised. Operation of position switches, indicators and alarms shall be verified monthly by operation of each operable vacuum breaker.

   b. Refueling Outage Tests
      (1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.
      (2) The suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

(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 (interval not to exceed 20 months) shall 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.

(2) Results of laboratory carbon sample analysis show $\geq$90% radioactive methyl iodine removal efficiency when tested in accordance with ASTM D 3803-79 (30° C, 95% relative humidity).

b. At least once per 18 months by demonstrating:

(1) That the pressure drop across a HEPA filter is equal to or less than the maximum allowable pressure drop indicated in Figure 4.5.1.

(2) The inlet heater is capable of at least 10.9 KW input.

(3) Operation with a total flow within 10% of design flow.

c. At least once per 30 days on a STAGGERED TEST BASIS by operating each circuit for a minimum of 10 hours.

d. Anytime the HEPA filter bank or the charcoal absorbers have been partially or completely replaced, the test per 4.5.H.1.a (as applicable) will be performed prior to returning the system to OPERABLE STATUS.

e. Automatic initiation of each circuit every 18 months.

I. Inerting Surveillance

When an inert atmosphere is required in the primary containment, the oxygen concentration in the primary containment shall be checked at least weekly.

J. Drywell Coating Surveillance

Carbon steel test panels coated with Firebar D shall be placed inside the drywell near the reactor core midplane level. They shall be removed for visual observation and weight loss measurements during the first, second, fourth and eighth refueling outages.

K. Instrument Line Flow Check Valves Surveillance

The capability of each instrument line flow check valve to isolate shall be tested at least once in every period between refueling outages. Each time an instrument line is returned to service after any condition which could have produced a pressure flow disturbance in that line, the open position of the flow check valve in that line shall be verified. Such conditions include:

Leakage at instrument fittings and valves
Venting an unisolated instrument or instrument line
Flushing or draining an instrument
Installation of a new instrument or instrument line

L. 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 160° F 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.

M. Shock Suppressors (Snubbers)

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

1. Each snubber shall be demonstrated OPERABLE by performance of the following inspection program:

a. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of the categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.5-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.5-1.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) that there are no visible indications of damage or impaired OPERABILITY; (2) attachments to the foundation or supporting structure are functional; and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional.

Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified as acceptable for the purpose of establishing the next visual inspection interval, providing that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.5.M.d or 4.5.M.e. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

c. Functional Tests

At least once every 24 months, a representative sample (10% of the total of each type of snubber in use in the plant) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.5.M.d or 4.5.M.e, an additional 10% of that type of snubber shall be functionally tested. As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, mechanical or hydraulic.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle.

2. Snubbers within 5 feet of heavy equipment (valve, pump, motor, etc.).

3. Snubbers within 10 feet of the discharge from a safety relief valve.

In addition to the regular sample, snubbers which failed a previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed (if it is repaired and installed in another position) and the replacement snubber shall be retested. The results from testing of these snubbers are not included for determining additional sampling requirements.

For any snubber that fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated. If caused by manufacturer or design deficiency, actions shall be taken to ensure that all snubbers of the same design are not subject to the same defect.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubbers to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiated free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 24 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records. Service life shall not at any time affect reactor operations.

N. Secondary Containment Isolation Valves

1. Each secondary containment isolation valve shall be demonstrated operable prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel. Following maintenance, repair or replacement work on the control or power circuit for the valves, the affected component shall be tested to assure it will perform its intended function in the circuit.

2. At least once per refueling outage, all valves shall be tested for automatic closure by an isolation signal.

Basis: In the event of a loss-of-coolant accident, the peak drywell pressure would be 38 psig which would rapidly reduce to 20 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 35 psig is calculated to be about 7 seconds. Following the pipe break, absorption chamber pressure rises to 20 psig within 8 seconds, equalizes with drywell pressure at 25 psig within 60 seconds and thereafter rapidly decays with the drywell pressure decay.(1)

The design pressures of the drywell and absorption chamber are 62 psig and 35 psig, respectively.[2] The original calculated 38 psig peak drywell pressure was subsequently reconfirmed.[3] A 15% margin was applied to revise the drywell design pressure to 44 psig. The design leak rate is 0.5%/day at a pressure of 35 psig. As pointed out above, the pressure response of the drywell and absorption chamber following an accident would be the same after about 60 seconds. Based on the calculated primary containment pressure response discussed above and the absorption chamber design pressure, primary containment pre-operational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and absorption chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.0%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 10 rem and the maximum total thyroid dose is about 139 rem at the site boundary considering fumigation conditions over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 2 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission product from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected offsite doses and 10 CFR 100 guideline limits.

Although the dose calculations suggest that the allowable test leak rate could be allowed to increase to about 2.0%/day before the guideline thyroid dose limit given in 10 CFR 100 would be exceeded, establishing the limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as-built" condition is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163,"Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J", Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements".

The Primary Containment Leakage Rate Testing Program conforms with this guidance.

The maximum allowable leakage rate for the primary containment (L_a) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a). As discussed below, P_a for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

The containment integrity isolation valves are provided to maintain containment integrity following the design basis loss-of-coolant accident. The closure times of the isolation valves on the containment are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each refueling outage is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). The specified closure times are adequate to restrict the coolant loss from the circumferential rupture of any of these lines outside the containment to less than that for a main steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

Since the main steam line isolation valves are normally in the open position, more frequent testing is specified. Per ASME Boiler and Pressure Vessel Code, Section XI, the quarterly full closure test will ensure operability and provide assurance that the valves maintain the required closing time. The minimum time of 3 seconds is based on the transient analysis of the isolation valve closure that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time of 10 seconds is based on the value assumed for the main steam line break dose calculations.

Surveillance of the suppression chamber-reactor building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. As a result, a testing frequency of three months for operability is considered justified for this equipment. Inspections and calibrations are performed during the refueling outages, this frequency being based on equipment quality, experience, and engineering judgement.

The 14 suppression chamber-drywell vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage, each valve is tested to assure that it will open fully in response to a force less than that specified. Also, it is inspected to assure that it closes freely and operates properly.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 10.5 $in.^2$ (expressed as vacuum breaker open area). This is equivalent to one vacuum breaker disk off its seat 0.371 inch; this length corresponds to an angular displacement of 1.25°. A conservative allowance of 0.10 inch has been selected as the maximum permissible valve opening. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a non-seated valve.

At the end of each refueling cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure. The pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event, the leakage source will be identified and eliminated before power operation is resumed. If the drywell pressure can be increased by 1 psi over the suppression chamber, the rate of change of the suppression chamber pressure must not exceed a rate equivalent to the rate of air flow from the drywell to the suppression chamber through a 2-inch orifice. In the event the rate of change of pressure exceeds this value, then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised monthly and immediately following termination of discharge of steam into the suppression chamber. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening- closing cycle, the position indicating lights are designed to function as follows:

Full Closed	2 Green - On
(Closed to 0.10" open)	2 Red - Off
Open 0.10"	2 Green - Off
(0.10" open to full open)	2 Red - Off

During each refueling outage, four suppression chamber-drywell vacuum breakers will be inspected to assure components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in about 1/10th of the design lifetime is extremely conservative. The alarm systems for the vacuum breakers will be calibrated during each refueling outage. This frequency is based on experience and engineering judgement.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain a 1/4 inch of water vacuum, tests the operation of the reactor building isolation valves, leakage tightness of the reactor building and performance of the standby gas treatment system. Checking the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing the reactor building in leakage test prior to refueling demonstrates secondary containment capability prior to extensive fuel handling operations associated with the outage. Verifying the efficiency and operation of charcoal filters once per 18 months gives sufficient confidence of standby gas treatment system performance capability. A charcoal filter efficiency of 99% for halogen removal is adequate.

The in-place testing of charcoal filters is performed using Freon-112* which is injected into the system upstream of the charcoal filters. Measurement of the Freon concentration upstream and downstream of the charcoal filters is made using a gas chromatograph. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. 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 Cleaning Conference.**

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential releases 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.

* Trade name of E. I. DuPont de Nemours & Company

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

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 the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and used as the basis upon which to determine the next inspection interval for that category.

If continued operation cannot be justified with an unacceptable snubber, the snubber shall be declared inoperable and the applicable action requirements met. To determine the next surveillance interval, the snubber may be reclassified as acceptable if it can be demonstrated that the snubber is operable in its as-found condition by the performance of a functional test and if it satisfies the acceptance criteria for functional testing.

The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval. This interval depends on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. Table 4.5-1 establishes the length of the next visual inspection interval.

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 represents an adequate sample for such tests. Observed failures of these samples require testing of additional units.

After the containment oxygen concentration has been reduced to meet the specification initially, the containment atmosphere is maintained above atmospheric pressure by the primary containment inerting system. This system supplies nitrogen makeup to the containment so that the very slight leakage from the containment is replaced by nitrogen, further reducing the oxygen concentration. In addition, the oxygen concentration is continuously recorded and high oxygen concentration is annunciated. Therefore, a weekly check of oxygen concentration is adequate. This system also provides the capability for determining if there is gross leakage from the containment.

The drywell exterior was coated with Firebar D prior to concrete pouring during construction. The Firebar D separated the drywell steel plate from the concrete. After installation, the drywell liner was heated and expanded to compress the Firebar D to supply a gap between the steel drywell and the concrete. The gap prevents contact of the drywell wall with the concrete which might cause excessive local stresses during drywell expansion in a loss-of-coolant accident.

The surveillance program is being conducted to demonstrate that the Firebar D will maintain its integrity and not deteriorate throughout plant life. The surveillance frequency is adequate to detect any deterioration tendency of the material.[8]

The operability of the instrument line flow check valves are demonstrated to assure isolation capability for excess flow and to assure the operability of the instrument sensor when required.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and also observed during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

References

(1) Licensing Application, Amendment 32, Question 3
(2) FDSAR, Volume I, Section V-1.1
(3) GE-NE 770-07-1090, "Oyster Creek LOCA Drywell Pressure Response," February 1991
(4) Deleted
(5) FDSAR, Volume I, Sections V-1.5 and V-1.6
(6) FDSAR, Volume I, Sections V-1.6 and XIII-3.4
(7) FDSAR, Volume I, Section XIII-2
(8) Licensing Application, Amendment 11, Question III-18

FIGURE 4.5.1

MAXIMUM ALLOWABLE PRESSURE DROP
FOR HEPA FILTERS

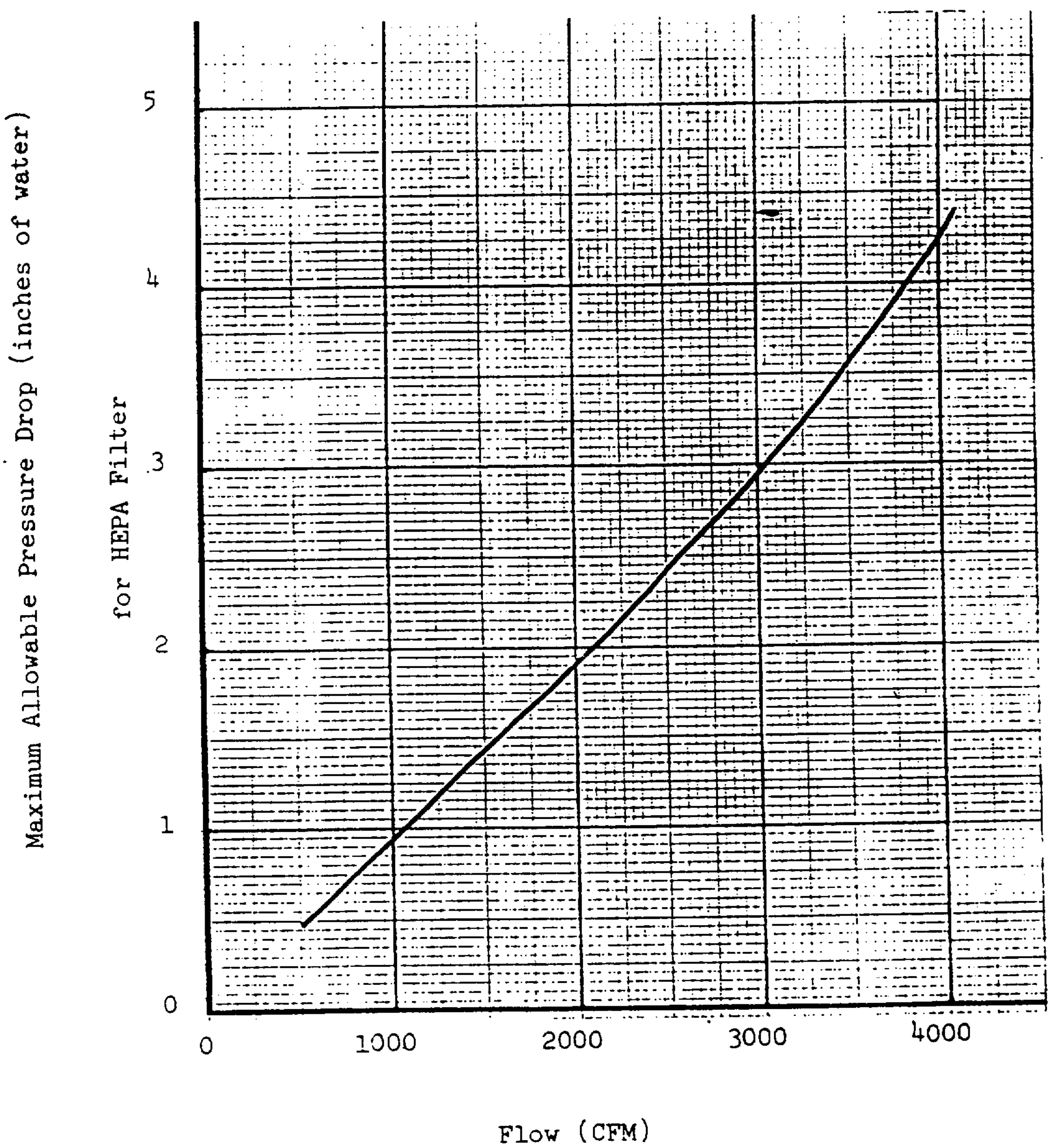

TABLE 4.5-1
SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF UNACCEPTABLE SNUBBERS

Population or Category (Notes 1,2)	Column A Extend Interval (Notes 3,6)	Column B Repeat Interval (Notes 4,6)	Column C Reduce Interval (Notes 5,6)
1	0	0	1
80	0	0	2
100	0	1	4

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the decision on how to categorize the snubbers must be made and documented before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissable. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number. in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

TABLE 4.5-1
SNUBBER VISUAL INSPECTION INTERVAL
Page 2 of 2

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: Each inspection interval shall be subject to the limitations of Technical Specification 1.24.

6.9.3 UNIQUE REPORTING REQUIREMENTS

Special reports shall be submitted to the Director of Regulatory Operations Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Materials Radiation Surveillance Specimen Reports (4.3A)

b. (Deleted)

c. Results of required leak tests performed on sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

d. Core Spray Sparger Inservice Inspection (Table 4.3.1-9)

   Prior to startup of each cycle, a special report presenting the results of the inservice inspection of the Core Spray Spargers during each refueling outage shall be submitted to the Commission for review.

e-j. Pursuant to the ODCM.

k. Records of results of analyses required by the Radiological Environmental Monitoring Program.

l. Failures and challenges to Relief and Safety Valves which do not constitute an LER will be the subject of a special report submitted to the Commission within 60 days of the occurrence. A challenge is defined as any automatic actuation (other than during surveillance or testing) of Safety or Relief Valves.

m. Plans for compliance with standby liquid control Specifications 3.2.C.3(b) and 3.2.C.3(e)(1) or plans to obtain enrichment test results per Specification 4.2.E.5.

n. Inoperable high range radioactive noble gas effluent monitor (3.13.H)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 186

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

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated February 23, 1996 (TSCR 242), GPU Nuclear Corporation (the licensee) requested changes to the Technical Specifications (TSs) for Oyster Creek Nuclear Generating Station to allow implementation of Appendix J, Option B. In a letter to the staff dated June 19, 1996, (TSCR 242, Rev. 1), the licensee docketed a revised submittal which replaced the February 23, 1996, submittal in its entirety. After further discussion with the NRC staff on July 16, 1996, the licensee provided another submittal, dated July 17, 1996 (TSCR 242, Rev. 2), which completely replaced the June 19, 1996, submittal. The licensee has established a "Containment Leakage Rate Testing Program" and proposed revising the TS to be compatible with Appendix J, Option B. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 which specifies a method acceptable to the NRC for complying with Option B.

On August 28, 1996, the licensee provided updated and corrected TS pages. These revisions were within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination; therefore renoticing was not warranted.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Containment Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995 (60 FR 49495), and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage-rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Oyster Creek Nuclear Generating Station TS.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's July 17, 1996, application to the NRC proposes to establish a "Primary Containment Leakage Rate Testing Program" and revises the TS to be compatible with this program. The program references RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires a change to the existing TS. In particular, a footnote to Definition 1.24, "Surveillance Requirements," and Definition 1.25, "Appendix J Test Pressure," TS Sections 4.5, "Containment System," and 6.9.3, "Unique Reporting Requirements." The corresponding bases were also modified, and page and section numbers were revised as necessary.

Option B permits a licensee to choose Type A; or Type B, and C; or Type A, B, and C testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The licensee has proposed an exception to RG 1.163 to permit a one-time schedular extension of the Type A test interval. This exception states that

> The first Type A test required by this program will be performed during refueling outage 17R.

The next Option A Type A test is currently scheduled for upcoming outage 16R. A Type A test also would be required during this outage under Option B of Appendix J because RG 1.163 specifies that the Type A test must be performed at P_a (35 psig for Oyster Creek). RG 1.163 also specifies that in order to extend the Type A test interval to 10 years, at least one of the previous two consecutive successful Type A tests must be performed at P_a. The licensee has previously performed Type A tests at a pressure less than P_a (20 psig).

Paragraph V.B.3 of 10 CFR Part 50, Appendix J, Option B provides that the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a RG. The licensee requested the one-cycle extension to the Type A test interval to allow an orderly transition from the existing reduced pressure Type A leakage rate testing prior to performing a full pressure Type A test and provided supporting analyses. The staff finds the licensee's request acceptable for the following reasons.

Over the last 10 years, three Type A tests have been performed and all three were within the specified limits of Appendix J. In addition, Oyster Creek's performance is consistent with that of the industry as a whole in that the major contributor to total identified leakage is found by Type B and C tests. Only a small fraction of the total leakage is detectable only through Type A testing. Type B and C testing will continue to be performed on the required schedules.

The Oyster Creek containment is inerted by replacing air with nitrogen. Nitrogen makeup to the containment is monitored daily and would serve as an indication of gross containment leakage.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and are, therefore, acceptable to the staff.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. This also 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 40019). 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.

6.0 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: R. Lobel

Date: September 3, 1996