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

January 6, 1986

Docket No. 50-219

See Correction Letter of 1/28/86

Mr. P. B. Fiedler
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE

Re: Oyster Creek Nuclear Generating Station

The Commission has issued the enclosed Amendment No. 97 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment is in response to your application dated August 23, 1985, which superseded your request of October 22, 1984.

This amendment authorizes changes to the Oyster Creek Appendix A Technical Specifications (TS) to revise the limiting conditions for operation and to add surveillance requirements for reactor coolant system leakage. These changes add two new definitions to TS Section 1.0, Definitions; revise TS 3.3.D and add TS 4.3.H. These changes incorporate in the TS the requirements you committed to in your response dated September 8, 1983, to IE Bulletin 82-03 and in Section 4.16.2 of the Integrated Plant Safety Assessment Report for Oyster Creek, NUREG-0822 dated January 1983.

The staff finds the proposed specifications regarding primary coolant leakage and leakage detection systems to be more restrictive than the current specifications; and, with one exception, consistent with the BWR Standard Technical Specifications (STS). The exception is that the proposed specifications do not include a Limiting Condition for Operation (LCO) on pressure boundary leakage as in the STS. We recommend you submit an additional license amendment application which includes an LCO on pressure boundary leakage in TS 3.3.D.

This letter closes out the staff's actions on Section 4.16.2 of the Integrated Plant Safety Assessment Report, NUREG-0822 dated January 1983, for Oyster Creek.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on October 23, 1985 (50 FR 43027). No public comments or requests for hearing were received.

A copy of our related Safety Evaluation is also enclosed. A notice of issuance pertaining to this action will appear in the Commission's biweekly notice publication in the Federal Register.

Sincerely,

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 97 to License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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A copy of our related Safety Evaluation is also enclosed. A notice of issuance pertaining to this action will appear in the Commission's biweekly notice publication in the Federal Register.

Sincerely,

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 97 to License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 97
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation and Jersey Central Power and Light Company (the licensees) dated August 23, 1985, 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 Provisional 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. 97, 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 its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 6, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 97

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
i	i
ii	ii
1.0-6	1.0-6
-	3.3-1a
3.3-2	3.3-2
3.3-5	3.3-5
4.3-2	4.3-2

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*Issued by NRC Order dated 10-24-80

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).

1.28 FRACTION OF RATED POWER (FRP)

The fraction of rated power is the ratio of core thermal power to rated thermal power.

1.29 TOP OF ACTIVE FUEL (TAF) - 353.3 inches above vessel zero.

1.30 REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

1.31 IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE is that leakage which is collected in the primary containment equipment drain tank and eventually transferred to radwaste for processing.

1.32 UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE is all measured leakage that is other than identified leakage.

D. Reactor Coolant System Leakage

1. Reactor coolant system leakage shall be limited to:

   a. 5 gpm unidentified leakage
   b. 25 gpm total (identified and unidentified)
   c. 2 gpm increase in unidentfied leakage rate within any 4 hour period while operating at steady state power

2. With the reactor coolant system leakage greater than the limits in 3.3.D.1.a or b above, reduce the leakage rate to within the acceptable limits within 8 hours, or place the reactor in the shutdown condition within the next 12 hours and be in the cold shutdown condition within the following 24 hours.

3. With any reactor coolant leakage greater than the limit in 3.3.D.1.c above, identify the source of leakage within 4 hours, or be in the shutdown condition within the next 12 hours and be in the cold shutdown condition within the following 24 hours.

4. For determination of unidentified leakage, the primary containment sump flow monitoring system shall be operable except as specified below:

   a. With the primary containment sump flow integrator inoperable:

      1. Restore it to operable status within 7 days.
      2. Calculate the unidentified leakage rate utilizing an acceptable alternate means as specified in plant procedures.

   b. If Specification 3.3.D.4.a cannot be met, place the reactor in the shutdown condition within the next 12 hours.

5. For determination of identified leakage, the primary containment equipment drain tank monitoring system shall be operable except as specified below:

   a. With the primary containment equipment drain tank monitoring system inoperable:

      1. Restore it to operable status within 7 days.

      2. Calculate the identified leakage rate utilizing an acceptable alternate means as specified in plant procedures.

   b. If Specification 3.3.D.5.a cannot be met, place the reactor in the shutdown condition within the next 12 hours.

E. Reactor Coolant Quality

1. The reactor coolant quality during power operation with steaming rates to the turbine-condenser of less than 100,000 pounds per hour shall be limited to:

conductivity	2 uS/cm	(S = mhos at $25^{o}C(77^{o}F)$)
chloride ion	0.1 ppm	

2. When the conductivity and chloride concentration limits given in 3.3.E.1 are exceeded, an orderly shutdown shall be initiated immediately, and the reactor coolant temperature shall be reduced to less than 212°F within 24 hours.

3. The reactor coolant quality during power operation with steaming rates to the turbine-condenser of greater than or equal to 100,000 pounds per hour shall be limited to:

conductivity	10 uS/cm	(S = mhos at $25^{o}C(77^{o}F)$)
chloride ion	0.5 ppm	

4. When the maximum conductivity or chloride concentration limits given in 3.3.E.3 are exceeded, an orderly shutdown shall be initiated immediately, and the reactor coolant temperature shall be reduced to less than $212^{o}F$ within 24 hours.

5. During power operation with steaming rates to the turbine-condenser of greater than or equal to 100,000 pounds per hour, the time limit above 1.0 uS/cm at 25°C (77°F) and 0.2 ppm chloride shall not exceed 72 hours for any single incident.

6. When the time limits for 3.3.E.5 are exceeded, an orderly shutdown shall be initiated within 4 hours.

F. Recirculation Loop Operability

1. The reactor shall not be operated with one or more recirculation loops out of service except as specified in Specification 3.3.F.2.

2. Reactor Operation with one idle recirculation loop is permitted provided that the idle loop is not isolated from the reactor vessel.

3. If Specifications 3.3.F.1 and 3.3.F.2 are not met, the reactor shall be placed in the cold shutdown condition within 24 hours.

The drywell floor drain sump and equipment drain tank provide the primary means of leak detection[9,10]. Identified leakage is that from valves and pumps in the reactor system and from the reactor vessel head flange gasket. Leakage through the seals of this equipment is piped to the drywell equipment drain tank. Leakage from other sources is classified as unidentified leakage and is collected in the drywell floor drain sump. Leakage which does not flash in a vapor will drain in the sump. The vapor will be condensed in the drywell ventilation system and routed to the sump.

Condensate cannot leave the sump or the drywell equipment drain tank unless the respective pumps are running. The sump and the drain tank are provided with two pumps each. Alarms are provided for the sump that will actuate on a predetermined pumpout rate [10] and will be set to actuate at a leakage that is less than the unidentified leakage limit of 5 gpm.

Additional qualitative information [10] is available to the operator via the monitored drywell atmospheric condition. However, this information is not quantitative since fluctuation in atmospheric conditions are normally expected, and quantitative measurements are not possible. The temperature of the closed cooling water which serves as coolant for the drywell ventilation system is monitored and also provides information which can be related to reactor coolant system leakage [9]. Additional protection is provided by the drywell high pressure scram which would be expected to be reached within 30 minutes of a steam leak of about 12 gpm[10].

During a loss of offsite AC power, the control rod drive hydraulic pumps, which are powered by the diesels, each can supply 110 gpm water makeup to the reactor vessel. A 25 gpm limit for total leakage, identified and unidentified, was established to be less than the 110 gpm makeup of a single rod drive hydraulic pump to avoid the use of the emergency core cooling system in the event of a loss of normal AC power.

Materials in the primary system are primarily 304 stainless steel and zircaloy fuel cladding. The reactor water chemistry limits are placed upon conductivity and chloride concentration since conductivity is measured continuously and gives an indication of abnormal conditions or the presence of unusual materials in the coolant, while chloride limits are specified to prevent stress corrosion cracking of stainless steel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 97 TO PROVISIONAL 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 August 23, 1985, which superseded its letter dated October 22, 1984, GPU Nuclear (the licensee) requested an amendment to the Appendix A Technical Specifications (TS) to Provisional Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station (Oyster Creek). This amendment would revise the limiting conditions for operation (LCO) and add surveillance requirements for reactor coolant system leakage. These changes would add two new definitions to TS Section 1.0, Definitions; revise TS 3.3.D and add TS 4.3.H. These changes would incorporate in the TS the requirements the licensee committed to in its response dated September 8, 1983, to IE Bulletin 82-03 and in Section 4.16.2 of the Integrated Plant Safety Assessment Report (IPSAR) for Oyster Creek, NUREG-0822, dated January 1983.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on October 23, 1985 (50 FR 43027). No public comments or requests for hearing were received.

2.0 DISCUSSION AND EVALUATION

In a letter dated October 22, 1984, the licensee requested an amendment to the TS to incorporate requirements regarding Reactor Coolant System leakage limits and leakage detection system operability which it committed to in the IPSAR (NUREG-0822) for Oyster Creek and in its response to IE Bulletin 82-03.

Currently, the TS do not contain either LCO or surveillance requirements regarding the leakage detection systems, as recommended by Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973, and included in the BWR Standard Technical Specifications (BWR - STS) (NUREG-0123). In Section 4.16.2 of the IPSAR the licensee committed to provide the appropriate action requirements in the TS for inoperable leakage detection systems (i.e., an inability to measure leakage)

and any necessary procedural changes to provide surveillance and testing commensurate with the required sensitivity. To satisfy this commitment, the licensee has added TS 3.3.D.3 and 3.3.D.4 which require that the primary containment sump flow monitoring system and the primary containment equipment drain tank monitoring system be operable and identify necessary actions should either system become inoperable. The former system is used to detect unidentified leakage, whereas the latter will detect identifiable leakage. In addition to these new requirements, the licensee has proposed additions and improvements to existing TS on leakage from the primary coolant system. Specifically, a limit on the allowable increase in rate of unidentified leakage has been proposed and time limits have been incorporated in action statements which previously had none. As discussed below, the proposed changes serve to make the TS better and more consistent with the current BWR-STS (ref. 1).

The licensee for Oyster Creek was sent IE Bulletin 82-03, Stress-Corrosion Cracking In Thick-Wall, Large-Diameter, Stainless-Steel, Recirculation System Piping at BWR Plants, on October 14, 1982. This bulletin was to notify certain BWR licensees and construction permit (CP) holders about the degradation in the recirculation system piping in the reactor coolant pressure boundary that was found at Nine Mile Point Unit 1 Nuclear Generating Station and described in detail in Information Notice 82-39, dated September 21, 1982. Action was required by the affected licensees including the licensee for Oyster Creek to (1) provide a reasonable level of assurance that inspections which are currently being performed or scheduled are sufficient to detect cracking in Boiling Water Reactor (BWR) thick-wall recirculation piping welds and (2) assist the NRC in determining the generic significance of the piping degradation at Nine Mile Point.

In addition to IE Bulletin 82-03, the NRC has issued the following documents on stress corrosion cracking in the BWR reactor coolant pressure boundary: NUREG-0133, Revision 1, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, dated July 1980; IE Bulletin No. 82-03, Revision 1, dated October 28, 1982; IE Bulletin No. 83-02, Stress Corrosion Cracking In Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants, dated March 4, 1983; and Generic Letter 84-11, Inspections of BWR Stainless Steel Piping, dated April 19, 1984. NUREG-0313, Revision 1, and Generic Letter 84-11, provided acceptable TS on the rate of increase of unidentified leakage from the reactor coolant system which would be sufficiently restrictive to ensure timely investigation of unidentified leakage for pipe cracking.

3.0 EVALUATION

3.1 Proposed Specifications on Leakage Detection Systems

The rates of identified and unidentified primary coolant system leakage at Oyster Creek are determined with the primary containment equipment drain tank and primary containment sump flow monitoring systems, respectively. These systems monitor the frequency at which liquid accumulated in the containment sump and drain tank is pumped to the waste collector tanks. Indications of high

and low water level in the sump and tank are used to signal initiation and termination of pumping operations. A timer is used to determine if the pumping frequency is indicative of a fill rate approaching TS leakage limits. Alarms are actuated should this be the case. The systems are also equipped with integrating flow meters, which keep track of liquid pumped from the sump and drain tank on a daily basis. These meters are located between the sump and drain tank pumps and the respective waste collector tanks. The proposed TS permit operation for 7 days with the flow integrators inoperable. During this period, a manual procedure is used to determine leak rates. This procedure involves utilization of a dedicated operator to transfer liquid from the sump and drain tank to the appropriate waste hold-up tanks and record the times transfers are made. A calculation prescription is then used to determine the average leak rate.

Operability requirements for primary coolant leakage detection systems are included in the BWR-STS. The inclusion of such requirements in the Oyster Creek TS represents an increase in plant safety margins. The staff finds the specification proposed by the licensee to be consistent with the BWR-STS and, therefore, acceptable.

3.2 Proposed Specifications on Primary Coolant Leakage

The TS requirements on primary leakage proposed by the licensee are compared with the existing Oyster Creek leakage TS and the current BWR-STS in Tables 1 and 2. The proposed specifications are consistent with corresponding BWR-STS and conservative with respect to the present Oyster Creek TS; and are therefore acceptable.

The LCO on primary coolant leakage are in terms of identified and unidentified leakage. The definitions proposed by the licensee are consistent with the BWR-STS and correct for the Oyster Creek leakage detection system. Therefore, the staff concludes that these definitions are acceptable.

The licensee's proposed TS on the rate of increase of unidentified leakage in its application dated August 23, 1985, are also consistent with the TS in NUREG-0133, Revision 1, and GL 84-11. The TS in NUREG-0133, Revision 1, and GL 84-11 are applicable to Oyster Creek for a sump level monitoring system which the plant has. Therefore, the staff concludes this TS is acceptable because it is consistent with TS in NUREG-0313 Revision 1 and GL 84-11 which would ensure timely investigation of unidentified leakage for pipe cracking.

The licensee proposed changes to revise the Bases for the TS on reactor coolant system leakage. This revision is to correct the Bases to have it reflect the existing plant configuration. The staff has reviewed this change and concluded it is correct and, therefore, acceptable.

TABLE 1

COMPARISON OF OYSTER CREEK PRESENT AND PROPOSED TECHNICAL SPECIFICATIONS

ON RCS LEAKAGE WITH BWR STANDARD TECHNICAL SPECIFICATIONS

OYSTER CREEK (PRESENT)		OYSTER CREEK (PROPOSED)		BWR STS	
LCO	ACTION	LCO	ACTION	LCO	ACTION
None.	None.	None.	None.	No pressure boundary leakage.	Hot shutdown in 12 hours, cold shutdown in next 24 hours.
Unidentified leakage rate of less then 5 gpm.	Reduce below limit or be in cold shutdown.	Unidentified leakage rate of less than 5 gpm.	Reduce below limit in 8 hours or be in hot shutdown in 12 hours and cold shutdown in next 24 hours.	Unidentified leakage rate of less than 5 gpm.	Reduce below limit in 4 hours or be in hot shutdown in 12 hours and cold shutdown in next 24 hours.
Total leakage rate (identified and unidentified) of less than 25 gpm.	Reduce below limit or be in cold shutdown.	Total leakage rate (identified and unidentified) of less than 25 gpm.	Reduce below limit in 8 hours or be in hot shutdown in 12 hours and cold shutdown in next 24 hours.	Total leakage rate (identified and unidentified) of less than 25 gpm.	Reduce below limit in 4 hours or be in hot shutdown in 12 hours and cold shutdown in next 24 hours.
None.	None.	Increase in unidentified leakage rate less than 2 gpm within any 4 hour period.	Reduce below limit in 8 hours or be in hot shutdown in 12 hours and cold shutdown in next 24 hours.	Increase in unidentified leakage rate less than 2 gpm within any 4 hour period.	Identify source of leak in 4 hours or be in hot shutdown in 12 hours and cold shutdown in next 24 hours.

TABLE 2

COMPARISON OF OYSTER CREEK PRESENT AND PROPOSED SURVEILLANCE REQUIREMENTS

ON RCS LEAKAGE WITH BWR STANDARD TECHNICAL SPECIFICATION REQUIREMENTS

OYSTER CREEK (PRESENT)	OYSTER CREEK (PROPOSED)	BWR STS
A visual examination for leaks shall be made with the Reactor Coolant system at pressure during each scheduled refueling outage or after major repairs have been made to the reactor Coolant system.	1. Determine unidentified leakage once every 4 hours. 2. Determine total leakage once every 8 hours. 3. Perform channel calibration of primary containment sump and primary containment equipment drain tank flow integrators once every 18 months.	Demonstrate RCS leakage is within limits by monitoring: 1. Drywell atmospheric radioactivity at least once per 12 hours. 2. Drywell sump flowrate at least once per 12 hours. 3. Drywell air coolers condensate flow rate at least once per 12 hours. 4. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

3.3 Conclusion

We have reviewed the licensee's proposed TS regarding primary coolant leakage limits and leakage detection systems and compared them with the current Oyster Creek TS and BWR-STS. The staff finds the proposed specifications to be more restrictive than the current TS and consistent with BWR-STS and, therefore, concludes that the TS which have been proposed are acceptable. With this action, the staff has resolved the issue raised in Section 4.16.2 of the IPSAR.

The staff finds, however, there is one exception to the proposed TS being consistent with BWR-STS. The exception is that the proposed specifications do not include an LCO on pressure boundary leakage as in the BWR-STS. The staff recommends that the licensee submit an additional license amendment application which includes an LCO on pressure boundary leakage in TS 3.3.D.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

5.0 CONCLUSION

The staff 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; and (2) 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.

6.0 REFERENCES

1. NUREG-0123, Standard Technical Specifications for General Electric Boiling Water Reactors, Revision 4.

Principal Contributors: M. Caruso and J. Donohew.

Dated: January 6, 1986