Mr. Michael B. Roche
Vice President and Direc'
GPU Nuclear, Inc.
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, NJ 08731

May 13, 1999

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION UNIT 1 - ISSUANCE OF AMENDMENT RE: LICENSING BASIS CHANGE FOR CONTAINMENT OVERPRESSURE (TAC NO. MA1719)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 206 to Facility Operating License No. DPR-16 in response to your application dated May 5, 1998, as supplemented by letters dated August 3 (2 letters), September 14, and December 22, 1998.

The amendment approves the use of a small amount of containment overpressure to ensure sufficient net positive suction head for the emergency core cooling system pumps, which constitutes a change to the licensing basis for Oyster Creek.

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

Sincerely,
ORIGINAL SIGNED BY:
Ronald B. Eaton, Sr. Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosure: 1. Amendment No. 206 to DPR-16
2. Safety Evaluation

cc w/encl: See next page

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

May 13, 1999

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

GPU NUCLEAR, INC.

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear, Inc., et al. (the licensee), dated May 5, 1998, as supplemented August 3 (2 letters), September 14, and December 22, 1998, 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, changes to the UFSAR to reflect the use of a small amount of containment overpressure to ensure sufficient net positive suction head for the emergency core cooling system pumps as set forth in the licensee's application dated May 5, 1998, as supplemented August 3 (2 letters), September 14, and December 22, 1998, are authorized.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: May 13, 1999

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR, INC.

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated May 5, 1998 (Reference 1), as supplemented by letters dated August 3 (2 letters, References 2 and 3), September 14 (Reference 4), and December 22, 1998 (Reference 5), GPU Nuclear, Inc., (GPU or the licensee), requested a change to the Oyster Creek licensing basis. The requested change involves the use of a small amount of containment overpressure to ensure sufficient net positive suction head (NPSH) for the emergency core cooling system (ECCS) pumps. GPU installed new large-capacity ECCS strainers during the 17R refueling outage to meet the requested actions under NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors," (Reference 6). The licensee determined this was an unreviewed safety question pursuant to the requirements of 10 CFR 50.59.

The supplemental letters provided additional information that was within the scope of the original application and did not change the staff's proposed no significant hazards consideration determination.

Oyster Creek is a BWR 2 with a Mark I Containment. The Oyster Creek ECCS system consists of four containment spray pumps and four core spray pumps. All eight pumps take suction from a common ring header. The core spray system contains two completely independent loops. Each core spray loop consists of two main pumps, two booster pumps, two sets of parallel isolation valves inside and outside the drywell, a spray sparger, and associated piping, instrumentation and controls. One main pump from each system along with one booster pump from either system can supply 100 percent rated flow.

This safety evaluation will discuss the two significant cases presented by the licensee; case 8, referred to as the Emergency Operating Procedure (EOP) case, and case 9, referred to as the design basis case. The staff notes that it is the EOP case which requires the requested amount of containment overpressure credit and not the design basis case. This is due to the different assumptions of the two cases. In their submittal, GPU demonstrated that 1.25 psig of containment overpressure was required to ensure adequate NPSH to the Oyster Creek core spray pumps for the first hour following a large break loss of coolant accident (LOCA).

Additionally, the staff reviewed the assumptions used by the licensee in sizing the proposed ECCS suction strainers and the licensee's calculations demonstrating that adequate containment pressure exists when required.

2 .0 CORE SPRAY NPSH CALCULATIONS

The licensee provided several cases which incorporate different operator actions in response to a large break LOCA and a single failure. The limiting single failure is the failure of the "A" core spray pump. According to the licensee, this failure minimizes the NPSH conditions while maximizing flows.

In response to Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps" (Reference 7), GPU provided the relationship which was used to calculate the available NPSH (NPSHA) for the core spray pumps.

$$NPSHA = h_a + h_{st} - h_{vapor} - h_{fs}$$

where

- h_a atmospheric pressure
- h_{st} static fluid pressure associated with the pumps
- h_{vapor} saturation pressure of the suppression pool
- h_{fs} head loss associated with the flow through the system

The staff notes that the original licensee response to Generic Letter 97-04 did not include the strainer head loss associated with the new ECCS suction strainers since the new ECCS suction strainers were not installed at that time. The new strainer head loss is included in the revised NPSH calculations which the staff reviewed as part of its evaluation. This additional head loss is represented by $h_{strainer}$ which is added to the h_{fs} term in the equation above. The current Oyster Creek NPSH analyses include the time dependent head loss for debris loading across the new ECCS suction strainers.

2.1 Short Term NPSH Requirements

The "short term" for this accident analysis is defined as the first 10 minutes after the LOCA. Generally, in design basis cases, operator action is not credited during the short term. For this analysis, core spray pumps "B" and "C" are at run out flow during the short term. The "A" core spray pump is the limiting single failure for NPSH calculations and is assumed to fail while the "D" core spray pump is the backup to the "B" core spray pump and is not required. One of the design assumptions used in the analysis is that the operators make no attempt to throttle the core spray pumps during the first 10 minutes, so the core spray pumps will run at the highest flow rate that piping friction losses and reactor pressure will physically allow. For analysis purposes, the core spray pumps are at their run out flow of 5000 gpm when a booster pump is also running. The reactor is assumed to be at 0 psig.

At Oyster Creek, the containment sprays do not have an automatic initiation based on high containment pressure. The containment spray pumps must be initiated by operator action. The staff notes that early initiation of the containment sprays will reduce the containment pressure and maximize the flow through the strainers.

In accordance with the EOPs, the operators will manually start the containment spray pumps when the wetwell pressure exceeds 12 psig. Therefore, the EOP case (case 8), assumes that the operators manually initiate the containment spray pumps at the onset, i.e., within a few seconds of the start of the large break LOCA. The "B" and "C" core spray pumps are assumed to operate at run out flow. No operator action is credited to reduce core spray or containment spray flow. The EOP case maximizes the flow through the strainers which results in higher head losses and decreases the NPSHA. Refer to Table 1 for sequencing and flows of the core spray and containment spray pumps during this time period.

The design basis case, (case 9), assumes that only the "B" and "C" core spray pumps are operating at run out flows for the first 10 minutes following the LOCA. No operator action is credited to reduce core spray flow or initiate containment spray. The design basis case maximizes the suppression pool temperature for the short and long term after the LOCA. Refer to Table 2 for sequencing and flow of the core spray pumps during this time period.

The GPU calculations state that the maximum suppression pool temperature at 10 minutes is 139.2 °F for case 8 and 139.7 °F for case 9. The licensee's calculations demonstrate that, at the run out flows, with the calculated ECCS strainer head loss and a suppression pool temperature up to 139.2 °F, containment overpressure of 1.25 psig is required for the core spray pumps during the first 10 minutes after the LOCA for the EOP case. This conclusion is based on the assumption that the containment spray pumps ran continuously during the first 10 minutes after the LOCA, which is conservative. The staff notes that GPU also took credit for 1.25 psig during the short term for case 9, the design basis case. Although this credit was not required to meet the NPSH requirements for case 9, the staff finds it reasonable that the 1.25 psig credit be applied to the design basis case. Based on these analyses, the staff finds the use of 1.25 psig containment overpressure acceptable for the first 10 minutes after the LOCA for cases 8 and 9.

2.2 Long-term NPSH Requirements

The long term of the accident analysis is defined as the time period from 10 minutes to the end of the accident. Ten minutes after the onset of the LOCA, the analyses credit operator action to throttle the "C" core spray pump to the design flow rate of 4350 gpm and trip the "C" booster pump. At 1 hour after the LOCA, the analyses credit operator action to throttle the "B" core spray pump to the design flow rate of 4350 gpm and trip the "B" booster pump.

As stated before, the operators will manually start the containment spray pumps when the wetwell pressure exceeds 12 psig in accordance with procedures. Therefore, the EOP case models two containment spray pumps operating at 4200 gpm. Although the design basis case results in the higher suppression pool temperature, the EOP case results in higher flows through the strainers during the first hour of the event. Thus, the EOP case is more limiting than the design basis case from the NPSH perspective for the first hour. Refer to Table 1 for sequencing and flows of the core spray and containment spray pumps during this time period.

The design basis case analyzes the degraded case where operators start one containment spray pump at 10 minutes after the LOCA. This containment spray pump is assumed to be operating at 3200 gpm for the remainder of the accident scenario. Refer to Table 2 for sequencing and flows of the core spray and containment spray pumps for the design basis case during this time period.

The GPU calculations state that the maximum suppression pool temperature that will occur at 60 minutes is 142.5 °F for case 8 and 151.7 °F for case 9. The staff notes that the peak suppression pool temperature, as calculated in case 9, is not reached at the 60-minute mark in the accident analysis. Rather, the peak suppression pool temperature occurs at approximately 4.5 hours after the onset of the LOCA. As noted above, at 60 minutes, the operators throttle the "B" core spray pump and trip the associated booster pump. Thus, the NPSH requirements for the core spray pumps are lower when the maximum suppression pool temperature is reached. Furthermore, the licensee's calculations demonstrate that with the calculated ECCS strainer head loss and a suppression pool temperature up to 142.5 °F, containment overpressure of 1.25 psig is required for the core spray pumps during the first hour after the LOCA for the EOP case. This conclusion is based on the assumption that the containment spray pumps run continuously during the first hour after the LOCA, which is conservative.

The staff notes that GPU also credited 1.25 psig during the long term for case 9, the design basis case. Although this credit was not required to meet the NPSH requirements for case 9, the staff finds it reasonable that the 1.25 psig credit be applied to the design basis case. Based on these analyses, the staff finds the use of 1.25 psig containment overpressure acceptable for the first hour after the LOCA for cases 8 and 9. Additionally, the staff finds that with credit for 1.25 psig of containment overpressure for the first hour following a LOCA, NPSH for the ECCS pumps will be available to meet the long-term worst case scenario.

3.0 EVALUATION OF STRAINER LOADING AND ADEQUATE CONTAINMENT PRESSURE

This evaluation consists of two parts. The first part (Section 3.1) deals with the licensee's analysis of the potential debris loading on the strainers. The second part (Section 3.2) deals with the licensee's containment pressure analysis.

3.1 Potential debris loading

Although not directly part of the licensee's unreviewed safety question, the staff has reviewed the licensee's estimated debris loading. The amount of debris which collects on the suction strainers during a postulated LOCA is significant in determining the head losses which are calculated as part of determining the available NPSH.

By letter dated November 23, 1998, the licensee informed the staff that it had installed new large passive suction strainers. These strainers are General Electric Company (GE) Stacked Disk ECCS Suction Strainers described in the General Electric Company topical report NEDC-32721 (GE proprietary) which was submitted to the NRC staff for review by letters dated April 3 and November 21, 1997. The licensee has designed the strainers in accordance with the Boiling Water Reactor Owners Group topical report NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," dated November 1996, which was approved by the staff in a safety evaluation dated August 20, 1998. The licensee also stated that post maintenance testing of the new strainers validated the clean strainer pressure drop used in its analysis.

The staff has divided the review of the GE Stacked Disk Strainer topical report into two parts. The first part concerns the methodology used to estimate the head loss across the strainer for postulated debris loadings. The staff has issued a safety evaluation report, dated February 3, 1999, which approves this portion of the topical report. The second part involves the review of the methods used to determine the structural loads on the ECCS penetrations, piping and strainers caused by hydrodynamic forces generated by the LOCA. This portion of the staff review is continuing and will be documented upon completion in a separate safety evaluation report. The staff has determined that operation of Oyster Creek with these strainers is acceptable in the interim. ECCS strainer blockage has actually occurred at operating BWRs on several occasions (see Bulletin 96-03 for a discussion of these events). Installing these larger strainers will ensure that such events are unlikely to occur at Oyster Creek. In addition, the licensee has evaluated the induced hydrodynamic loads and has stated that the new strainers can withstand the expected loads without additional structural supports (see licensee's November 23, 1998 letter). For these reasons, interim operation is acceptable, pending completion of the staff's review of the topical report.

The licensee's May 5, 1998, letter to the NRC stated that the calculation of the amount of debris dislodged from piping due to the LOCA was determined using pipe break locations "with the greatest potential for failure under cyclic loading conditions." These methods were used in the FSAR for pipe break, pipe whip, and jet impingement calculations. However, since the basis of Bulletin 96-03 is to assure compliance with 10 CFR 50.46, the licensee must consider the worst case break location. In an August 3, 1998, letter from the licensee to the NRC staff, the licensee clarified that "[t]he strainer was designed to accommodate breaks" outside those identified in the OC [Oyster Creek] FSAR." A location was selected "which maximizes the debris delivered to the suppression pool." The staff finds this to be acceptable.

The staff noted that the licensee's May 5, 1998, submittal did not account for any "Additional Operational Debris." This category includes items left in containment that cannot withstand the forces generated by a LOCA and may be transported to the suppression pool where they may contribute to the strainer debris loading. In response to the staff's concern, the licensee stated that "[t]here is sufficient conservatism in the other debris loading assumed in the design to accommodate any reasonable amount of unexpected debris within the containment." However, the licensee also adjusted the predicted amount of debris to include 25 pounds of operational debris. The staff finds this acceptable since any operational debris found exceeding this limit should be removed to maintain the assumptions of the calculations.

Protective coatings that are dislodged by pressure, temperatures, or stresses caused by a LOCA can be a source of debris. The licensee has followed the guidance of the Boiling Water Reactor Owners Group Utility Resolution Guidance to determine the amount of inorganic zinc drywell coating (47 pounds) assumed to be dislodged by the LOCA. In addition, the licensee has made an allowance for paint chips from drywell equipment and wetwell coatings. The licensee's May 5, 1998, submittal noted that the wetwell containment coating was blistered. In an August 3, 1998, letter to the NRC, the licensee described tests performed on the coatings which demonstrated adherence of the coating in areas away from and between blisters and also demonstrated that the blistered areas were unlikely to generate significant amounts of debris. In addition, the licensee pointed out that in the period between January 1984 and December 1986, there were 11 Electromagnetic Relief Valve (EMRV) actuations as part of plant surveillance tests which resulted in steam blow down through the quenchers in 4 wetwell bays. A subsequent visual inspection revealed no spalling of the coating or the blisters. Based on this information, the licensee has concluded in its August 3, 1998, letter that "coating degradation should not become a source of debris during a DBA LOCA." The NRC Office of Nuclear Regulatory Research (RES) is currently evaluating the potential for coating to clog strainers. In addition, the NRC has issued Generic Letter 98-04[1] which requests information from licensees in order to determine whether their protective coating programs are in compliance with their existing licensing bases. In the interim, the licensee's conclusions regarding coating debris are acceptable.

3.2 Ensuring Availability of Adequate Containment Pressure

3.2.1 Control of Containment Spray Pumps

The licensee has determined that 1.25 psig of containment pressure is necessary to provide an adequate margin in NPSH (i.e., the difference between available and required NPSH) for 1 hour after the LOCA. The acceptability of crediting this amount of containment pressure is addressed above with respect to the unreviewed safety question. In order to ensure that this amount of overpressure will be available, the licensee, in its May 5, 1998, submittal proposed changing the setpoint of a switch which automatically trips the containment spray pumps so that the containment pressure will not fall below a drywell pressure of 1.25 psig. The setpoint of the switch is presently set at 0.6 psig. The setpoint modification was considered to be a backup to operator action to control containment spray. However, in a letter dated December 22, 1998, the licensee informed the staff that it was modifying this commitment by leaving the pressure switch setpoint at its present value. The licensee stated that this would ensure containment spray availability later in the accident when the containment overpressure was no longer necessary to meet NPSH requirements but might assist in managing the LOCA. The staff considers the licensee's change in commitment to be prudent and acceptable.

3.2.2 Containment Analysis

The licensee performed transient calculations of the containment pressure using assumptions intended to produce a minimum value. This provides assurance that adequate containment pressure will be available when required. These calculations were done with the GOTHIC computer code. The GOTHIC computer code has not been approved by the NRC, but it has been extensively validated for containment calculations and the staff finds the licensee's use of this code for this application to be acceptable.

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[1] Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated July 14, 1998.

The assumptions used by the licensee to minimize the containment pressure are discussed in Section 9.1 of the licensee's May 5, 1998, submittal. The staff has reviewed these assumptions and finds that they have been selected in a manner that tends to underestimate the containment accident pressure. For example, the initial drywell and wetwell pressures were minimized by minimizing the initial amount of noncondensible gases in the drywell and wetwell. This is done by assuming an initial pressure of 0 psig, an initial humidity of 100%, and an initial drywell temperature of 150 °F. This is acceptable for this application since it is conservative to demonstrate that the required containment pressure will be available while minimizing the estimated pressure.

The ANS 1979 decay heat model with an uncertainty of 2-sigma was used for the containment calculations. This is conservative and acceptable.

The staff has reviewed the licensee's EOP and design basis NPSH analyses for the core spray pumps. The staff finds that with credit for 1.25 psig of containment overpressure for the first hour following a LOCA, NPSH for the ECCS pumps will be available to meet the short and long term EOP case scenario. Additionally, the staff finds it reasonable that the 1.25 psig credit be applied to the design basis case for the first hour following a LOCA. Additionally, the staff has reviewed the debris load on the strainers, the operation of containment spray during a design basis LOCA and the calculation of the minimum containment pressure. The staff concludes that there is reasonable assurance that plant operation in this manner poses no undue risk to health and safety of the public and is therefore acceptable.

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. 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 (63 FR 56250). 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 Contributors: K. Kavanagh
R. Lobel

Date: May 13, 1999

7.0 REFERENCES

1. Roche, M. B., GPU Nuclear, to USNRC, "Oyster Creek Nuclear Generating Station Docket No. 50-219 - Request for Change to the Licensing Basis," May 5, 1998.

2. Roche, M. B., GPU Nuclear, to USNRC, "Oyster Creek Nuclear Generating Station Docket No. 50-219 - Request for Change to the Licensing Basis - Response to Request for Additional Information - #1940-98-20329," August 3, 1998.

3. Roche, M. B., GPU Nuclear, to USNRC, "Oyster Creek Nuclear Generating Station Docket No. 50-219 - Request for Change to the Licensing Basis - Response to Request for Additional Information - #1940-98-20430," August 3, 1998.

4. Roche, M. B., GPU Nuclear, to USNRC, "Oyster Creek Nuclear Generating Station Docket No. 50-219 - Request for a Change to the Licensing Basis - Response to a Request for Additional Information, " September 14, 1998.

5. Roche, M. B., GPU Nuclear, to USNRC, "Oyster Creek Nuclear Generating Station Docket No. 50-219 - Request for a Change to the Licensing Bases - ECCS Overpressure (TAC NO. MA1719)," December 22, 1998.

6. NRC BULLETIN 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.

7. NRC Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.

8. Roche, M. B., GPU Nuclear, to USNRC, "Oyster Creek Nuclear Generating Station Docket No. 50-219 - Generic Letter 97-04; 90 Day Response," January 5, 1998.

Table 1 EOP Case (Case 8)

Time (min)	Action	Flow (gpm)
0 - 10	"B" core spray and booster pump running "C" core spray and booster pump running 1 containment spray pump running 1 containment spray pump running	5000 5000 3650 4200
10 - 60	"B" core spray and booster pump running "C" booster pump tripped "C" core spray pump running 1 containment spray pump running 1 containment spray pump running	5000 4350 4200 4200
60 - end	"B" booster pump tripped "B" core spray pump running "C" core spray pump running 1 containment spray pump running 1 containment spray pump running	 4350 4350 4200 4200

Table 2 Design Basis Case (Case 9)

Time (min)	Action	Flow (gpm)
0 - 10	"B" core spray and booster pump running "C" core spray and booster pump running	5000 5000
10 - 60	1 containment spray pump running "C" booster pump tripped "C" core spray pump running "B" core spray and booster pump running	3200 4350 5000
60 - end	1 containment spray pump running "B" booster pump tripped "B" core spray pump running "C" core spray pump running	3200 4350 4350