November 1998

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

SUBJECT: OYSTER CREEK - ISSUANCE OF AMENDMENT NO. 203 , RE: CHANGES TO THE ADMINISTRATIVE CONTROLS (TAC NO. MA2369)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No203 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated July 21,1998.

The amendment will: 1) modify Specification 6.2.2.2(a) to provide some flexibility to accommodate unexpected absence of on-duty shift crew members, 2) eliminate reference to the Manager, Plant Operations in Specification 6.2.2.2(j) as the position has been eliminated, 3) reduce the maximum time in which to forward audit reports to the responsible manager from 60 days to 30 days, 4) replace the term "Vice President" with the term "Corporate Officer" in several places in Section 6, and 5) correct several typographical errors.

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

Sincerely,

Original signed by

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

Docket No. 50-219

Enclosures: 1. Amendment No. 203 to DPR-16
2. Safety Evaluation

dfo1

cc w/encls: See next page

DISTRIBUTION: See attached page

concurrence subject to change

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M. Roche
GPU Nuclear; Inc.

cc:

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DATED: November 30, 1998

AMENDMENT NO. 203 TO FACILITY OPERATING LICENSE NO. DPR-16-OYSTER CREEK

Docket File
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PDI-3 r/f
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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. 203
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 July 21, 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, 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.203 are hereby incorporated in the license. GPU Nuclear, Inc. 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 30, 1998

ATTACHMENT TO LICENSE AMENDMENT NO203

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
2.3-7	2.3-7
3.3-5	3.3-5
3.3-8a	3.3-8a
3.7-4	3.7-4
3.8-2	3.8-2
6-1	6-1
6-2	6-2
6-2a	6-2a
6-3	6-3
6-5	6-5
6-6	6-6
6-8	6-8
6-9	6-9

The low level water level trip setting of 11'5" above the top of the active fuel has been established to assure that the reactor is not operated at a water level below that for which the fuel cladding integrity safety limit is applicable. With the scram set at this point, the generation of steam, and thus the loss of inventory is stopped. For example, for a loss of feedwater flow a reactor scram at the value indicated and isolation valve closure at the low-low water level set point results in more than 4 feet of water remaining above the core after isolation (6). The TAF definition of 353.3 inches from vessel zero is based on a fuel length of 144 inches and it is applicable to the current fuel length of 145.24 inches.

During periods when the reactor is shut down, decay heat is present and adequate water level must be maintained to provide core cooling. Thus, the low-low level trip point of 7'2" above the core is provided to actuate the core spray system (when the core spray system is required as identified in Section 3.4) to provide cooling water should the level drop to this point.*

The turbine stop valve(s) scram is provided to anticipate the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system.

The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves to a load rejection and failure of the turbine bypass system. This scram is initiated by the loss of turbine acceleration relay oil pressure. The timing for this scram is almost identical to the turbine trip.

The undervoltage protection system is a 2 out of 3 coincident logic relay system designated to shift emergency buses C and D to on-site power should normal power be lost or degraded to an unacceptable level. The trip points and time delay settings have been selected to assure an adequate power source to emergency safeguards systems in the event of a total loss of normal power or degraded conditions which would adversely affect the functioning of engineered safety features connected to the plant emergency power distribution system.

<u>References</u>

(1) FDSAR, Volume 1, Section VII-4.2.4.2
(2) FDSAR, Amendment 28, Item III.A-12
(3) FDSAR, Amendment 32, Question 13
(4) Letters, Peter A. Morris, Director, Division of Reaction Licensing, USAEC, to John E. Logan, Vice President, Jersey Central Power and Light Company
(5) FDSAR, Amendment 65, Section B.XI
(6) FDSAR, Amendment 65, Section B.IX

Transformation temperature. The minimum temperature for pressurization at any time in life has to account for the toughness properties in the most limiting regions of the reactor vessel, as well as the effects of fast neutron embrittlement.

Curves A, B and C on Figures 3.3.1, 3.3.2 and 3.3.3 are derived from an evaluation of the fracture toughness properties performed on the specimens contained in Reactor Vessel Materials Surveillance Program Capsule No. 2 (Reference 14). The results of dosimeter wire analyses (Reference 14) indicated that the neutron fluence (E>1.0 MeV) at the end of 32 effective full power years of operation is 2.36 x 10^{18} n/cm^2 at the 1/4T (T=vessel wall thickness) location. This value was used in the calculation of the adjusted reference nil-ductility temperature which, in turn, was used to generate the pressure-temperature curves A, B and C on Figures 3.3.1, 3.3.2 and 3.3.3 (Reference 15). The 250°F maximum pressure test temperature provides ample margin against violation of the minimum required temperature. Secondary containment is not jeopardized by a steam leak during pressure testing, and the Standby Gas Treatment system is adequate to prevent unfiltered release to the stack.

Stud tensioning is considered significant from the standpoint of brittle fracture only when the preload exceed approximately 1/3 of the final design value. No vessel or closure stud minimum temperature requirements are considered necessary for preload values below 1/3 of the design preload with the vessel depressurized since preloads below 1/3 of the design preload result in vessel closure and average bolt stresses which are less than 20% of the yield strengths of the vessel and bolting materials. Extensive service experience with these materials has confirmed that the probability of brittle fracture is extremely remote at these low stress levels, irrespective of the metal temperature.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surface adjacent to the "O" rings of the head and vessel flange. The original Code requirement was that boltup be done at qualification temperatures (T3OL) plus 60° F. Current Code requirements state (Ref. 16) that for application of full bolt preload and reactor pressure up to 20% of hydrostatic test pressure, the RPV metal temperature must be at RT_{NDT} or greater. The boltup temperature of 85° F was derived by determining the highest value of (T3OL + 60) and the highest value of RT_{NDT}, and by choosing the more conservative value of the two. Calculated values of (T3OL + 60) and RT_{NDT} of the RPV metal temperature were 85° F and 36° F, respectively (Ref. 15). Therefore, selecting the boltup temperature to be 85° F provides 49°F margin over the current Code requirement based on RT_{NDT}.

Detailed stress analyses(4) were made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these analyses are presented and compared to allowable stress limits in Reference (4). The specific conditions analyzed currently include 240 cycles (17) of normal startup and shutdown with a heating and cooling rate of 100° F per hour applied continuously over a temperature range of 100° F to 546° F and for 10 cycles of emergency cooldown at a rate of 300° F per hour applied over the same range. A review of the original analysis shows that the components with the highest fatigue usage factor are the reactor vessel studs and reactor vessel basin seal skirt. These components have the potential to exceed the allowable fatigue usage factor if the number of thermal cycles (i.e., heatup/cooldown) exceed design assumptions. The number of heatup and cooldown cycles was reanalyzed, as documented by Reference (17), for a higher number of cycles (240) than expected in the original analysis (120). The reanalysis confirmed that the original fatigue usage factor limit of 0.8 is maintained. All other components have relatively low usage factors and are not expected to exceed the fatigue usage factor limit of 0.8 for the design life of 40 years.Thermal stresses from this analysis combined with the primary load

References:

(1) FDSAR, Volume I, Section IV-2
(2) Letter to NRC dated May 19, 1979, "Transient of May 2, 1979"
(3) General Electric Co. Letter G-EN-9-55, "Revised Natural Circulation Flow Calculation", dated May 29, 1979
(4) Licensing Application Amendment 16, Design Requirements Section
(5) (Deleted)
(6) FDSAR, Volume I, Section IV-2.3.3 and Volume II, Appendix H
(7) FDSAR, Volume I, Table IV-2-1
(8) Licensing Application Amendment 34, Question 14
(9) Licensing Application Amendment 28, Item III-B-2
(10) Licensing Application Amendment 32, Question 15
(11) (Deleted)
(12) (Deleted)
(13) Licensing Application Amendment 16, Page 1
(14) GPUN TDR 725 Rev. 3: Testing and Evaluation of Irradiated Reactor Vessel Materials Surveillance Program Specimens
(15) GENE-B13-01769 (GE Nuclear Energy): Pressure-Temperature Curves Per Regulatory Guide 1.99, Revision 2 for Oyster Creek Nuclear Generating Station.
(16) Paragraph G-2222(C), Appendix G, Section XI, ASME Boiler and Pressure Vessel Code, 1989 Edition with 1989 Addenda, "Fracture Toughness Criteria for Protection Against Failure."
(17) GPUN Safety Evaluation, SE-000221-004, "Reactor Vessel Thermal Cycles".

As indicated in Amendment 18 to the Licensing Application, there are numerous sources of diesel fuel which can be obtained within 6 to 12 hours and the heating boiler fuel in a 75,000 gallon tank on the site could also be used. As indicated in Amendment 32 of the Licensing Application and including the Security System loads, the load requirement for the loss of offsite power would require 12,410 gallons for a three day supply. For the case of loss of offsite power plus loss-of-coolant plus bus failure 9790 gallons would be required for a three day supply.

In the case of loss of offsite power plus loss-of-coolant with both diesel generators starting the load requirements (all equipment operating) shown there would not be three days' supply. However, not all of this load is required for three days and, after evaluation of the conditions, loads not required on the diesel will be curtailed. It is reasonable to expect that within 8 hours conditions can be evaluated and the following loads curtailed:

1. One Core Spray Pump
2. One Core Spray Booster Pump
3. One Control Rod Drive Pump
4. One Containment Spray Pump
5. One Emergency Service Water Pump

With these pieces of equipment taken off at 8 hours after the incident it would require a total consumption of 12,840 gallons for a three day supply. Therefore, a minimum technical specification requirement of 14,000 gallons of diesel fuel in the standby diesel generator fuel tank will exceed the engineered safety features operational requirement after an accident by approximately 9%.

During plant cold shutdown or refueling, it may be necessary to inspect, repair and replace the 15,000 gallon standby diesel generator fuel storage tank. This would require tank partial or full drain down. An alternate fuel supply configuration may be established which consists of temporary tanker trucks capable of containing 14,000 gallons. This configuration is capable of supporting continuous operation of both diesels for at least 3 days.

The temporary configuration is acceptable since a minimal power load would be required during and following a design basis condition of a loss of offsite power while the plant is in cold shutdown or refueling. Analysis shows that in the event of a tornado or seismic event which may cause a loss of offsite power and a temporary loss of the temporary EDG fuel oil supply, power can be restored before the consequences of previously analyzed conditions are exceeded.

References:

(1) Letter, Ivan R. Finfrock, Jr. to the Director of Nuclear Reactor Regulation dated April 4, 1978.

Each condenser containing a minimum total water volume of 22,730 gallons provides 11,060 gallons above the condensing tubes. Based on scram from a reactor power level of 1950 MWt (the design basis power level for the isolation condensers) the condenser system can accommodate the reactor decay heat(2,3) (corrected for U-239 and NP-239) for 1 hour and 40 minutes without need for makeup water. One condenser with a minimum water volume of 22,730 gallons can accommodate the reactor decay heat for 45 minutes after scram from 1950 MWt before makeup water is required. In order to accommodate a scram from 1950 MWt and cooldown, a total of 107,500 gallons of makeup water would be required either from the condensate storage tank or from the fire protection system. Since the rated reactor power is 1930 MWt, the above calculations represent conservative estimates of the isolation condenser system capability.

The vent lines from each of the isolation condenser loops to the main steam lines downstream of the main steam lines isolation valves are provided with isolation valves which close automatically on isolation condenser actuation or on signals which close the main steam isolation valves. High temperature sensors in the isolation condenser and pipe areas cause alarm in the control room to alert the operator of a piping leak in these areas.

Specification 3.8.E allows reduction in redundancy of isolation capability for isolation condenser inlet (steam side) isolation valves. Reaconable assurance of isolation capability is provided by testing the operability of the redundant valve. Specification 3.8.F allows short term inoperability of the AC motor-operated isolation condenser outlet (condensate return) valve. It is not necessary to test the redundant DC motor-operated valve as this valve is normally in the closed position. These specifications permit troubleshooting and repair as well as routine maintenance, such as valve stem packing addition or replacement, to be performed during reactor operation without reducing the redundancy of the isolation condenser heat sink function. The out of service time of 4 hours is consistent with that permitted for primary containment isolation valves.(5)

Either of the two isolation condensers can accomplish the purpose of the system. If one condenser is found to be inoperable, there is no immediate threat to the heat removal capability for the reactor and reactor operation may continue while repairs are being made. Therefore, the time out of service for one of the condensers is based on considerations for a one out of two system.(4) The test interval for operability of the valves required to place the isolation condenser in operation is once/month (Specification 4.8). An acceptable out of service time, T, is then determined to be 10 days. However, if at the time the failure is discovered and the repair time is longer than 7 days, the reactor will be placed in the cold shutdown condition. If the repair time is not more than 7 days the reactor may continue in operation, but as an added factor of conservatism, the motor operated isolation condenser and condensate makeup valves on the operable isolation condenser are verified operable daily. Expiration of the 7 day period or inability to meet the other specifications requires that the reactor be placed in the cold shutdown condition which is normally expected to take no more than 18 hours. The out of service allowance when the system is required is limited to the run mode in order to require system availability, including redundancy, at startup.

6.1 RESPONSIBILITY

6.1.1 The Vice President & Director Oyster Creek shall be responsible for overall facility operation. Those responsibilities delegated to the Vice President & Director as stated in the Oyster Creek Technical Specifications may also be fulfilled by the Director – Operations and Maintenance. The Vice President & Director shall delegate in writing the succession to this responsibility during his and/or the Director – Operations and Maintenance absence.

6.2 ORGANIZATION

6.2.1 Corporate

6.2.1.1 An onsite and offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

6.2.1.2 Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including operating organization positions. These relationships shall be documented and updated as appropriate, in the form of organizational charts. These organizational charts will be documented in the Updated FSAR and updated in accordance with 10 CFR 50.71e.

6.2.1.3 The President - GPU Nuclear shall have corporate responsibility for overall plant nuclear safety and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

6.2.2 FACILITY STAFF

6.2.2.1 The Vice President & Director Oyster Creek shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

6.2.2.2 The facility organization shall meet the following:

a. Each on duty shift shall include at least the following shift staffing:

- One (1) group shift supervisor
- Two (2) control room operators
- Three (3) equipment operators - one may be a Radwaste Operator
- One (1) Shift Technical Advisor (see h. below)

Except for the group shift supervisor, shift crew composition may be one less than the minimum requirements, for a period of time not to exceed two hours, in order to accommodate unexpected absence of on-duty shift crew members. Immediate action must be taken to restore the shift crew composition to within requirements given above. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crew member being late or absent.

b. At all times when there is fuel in the vessel, at least one licensed senior reactor operator shall be on site and one licensed reactor operator should be at the controls.

c. At all times when there is fuel in the vessel, except when the reactor is in COLD SHUTDOWN or REFUEL modes, two licensed senior reactor operators and two licensed reactor operators shall be on site, with at least one licensed senior reactor operator in the control room and one licensed reactor operator at the controls.

d. At least two licensed reactor operators shall be in the control room during all reactor startups, shutdowns, and other periods involving planned control rod manipulations.

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. An individual qualified in radiation protection measures shall be on site when fuel is in the reactor.

g. (deleted)

h. Each on duty shift shall include a Shift Technical Advisor except that the Shift Technical Advisors position need not be filled if the reactor is in the refuel or shutdown mode and the reactor is less than 212 F.

i. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions.

   In the event that unforeseen problems require substantial amounts of overtime to be used or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

   a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

   b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven-day period, all excluding shift turnover time.

   c. A break of at least eight hours should be allowed between work period, including shift turnover time.

   d. In a, b, and c above, the time required to complete shift turnover is to be counted as break time and is not to be counted as work time.

   e. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Department Managers, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

j. The Plant Operations Director and the Group Shift Supervisor require Senior Reactor Operators licenses. The Control Room Operators require a Reactor Operators license.

6.2.2.3 Individuals who train the operating staff and those who carry out the health physics and quality assurance function shall have sufficient organizational freedom to be independent of operational pressures, however, they may report to the appropriate manager on site.

6.3 Facility Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1 of 1978 for comparable positions unless otherwise noted in the Technical Specifications. Licensed operators shall meet the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees. Technicians and maintenance personnel who do not meet ANSI/ANS 3.1 of 1978, Section 4.5, are permitted to perform work for which qualification has been demonstrated.

6.3.2 The management position responsible for radiological controls shall meet or exceed the qualifications of Regulatory Guide 1.8 (Rev. 1-R, 9/75). Each other member of the radiation protection organization for which there is a comparable position described in ANSI N18.1-1971 shall meet or exceed the minimum qualifications specified therein, or in the case of radiation protection technicians, they shall have at least one year's continuous experience in applied radiation protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations and shall have been certified by the management position responsible for radiological controls as qualified to perform assigned functions. This certification must be based on an NRC approved, documented program consisting of classroom training with appropriate examinations and documented positive findings by responsible supervision that the individual has demonstrated his ability to perform each specified procedure and assigned function with an understanding of its basis and purpose.

6.3.3 The Shift Technical Advisors shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining program for operators shall be maintained under the direction of the Manager responsible for plant training and shall meet the requirements and recommendation of 10 CFR Part 55. Replacement training programs, the content of which shall meet the requirements of 10 CFR Part 55, shall be conducted under the direction of the Manager responsible for plant training for licensed operators and Senior Reactor Operators.

6.5 REVIEW AND AUDIT

6.5.1 TECHNICAL REVIEW AND CONTROL

The Corporate Officers of GPU Nuclear, Inc. shall be responsible for ensuring the preparation, review, and approval of documents required by the activities described in 6.5.1.1 through 6.5.1.5 within his functional area of responsibility as assigned in the GPU Nuclear Review and Approval Matrix. Implementing approvals shall be performed at the cognizant manager level or above.

ACTIVITIES

6.5.1.1 Each procedure required by Technical Specification 6.8 and other procedures which affect nuclear safety, and substantive changes thereto, shall be prepared by a designated individual(s)/group knowledgeable in the area affected by the procedure. Each such procedure, and substantive change thereto, shall be reviewed for adequacy by an individual(s)/group other than the preparer, but who may be from the same division as the individual who prepared the procedure or change.

RECORDS

6.5.1.13 Written records of activities performed under specifications 6.5.1.1 through 6.5.1.11 shall be maintained.

QUALIFICATIONS

6.5.1.14 Responsible Technical Reviewers shall meet or exceed the qualifications of ANSI/ANS 3.1-1978 Section 4.6 or 4.4 for applicable disciplines or have 7 years of appropriate experience in the field of his specialty. Credit towards experience will be given for advanced degrees on a one-for-one basis up to a maximum of two years. These Reviewers shall be designated in writing.

6.5.2 INDEPENDENT SAFETY REVIEW

FUNCTION

6.5.2.1 The Corporate Officers of GPU Nuclear, Inc. shall be responsible for ensuring the periodic independent safety review of the subjects described in 6.5.2.5 within his assigned area of safety review responsibility, as assigned in the GPUN Review and Approval Matrix.

6.5.2.2 Independent safety review shall be completed by an individual/group not having direct responsibility for the performance of the activities under review, but who may be from the same functionally cognizant organization as the individual/group performing the original work.

6.5.2.3 GPU Nuclear, Inc. shall collectively have or have access to the experience and competence required to independently review subjects in the following areas:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Nondestructive testing
f. Instrumentation and control
g. Radiological safety
h. Mechanical engineering
i. Electrical engineering
j. Administrative controls and quality assurance practices
k. Emergency plans and related organization, procedures and equipment
l. Other appropriate fields associated with the unique characteristics of Oyster Creek

6.5.2.4 Consultants may be utilized as determined by the cognizant Corporate Officer to provide expert advice.

RESPONSIBILITIES

6.5.2.5 The following subjects shall be independently reviewed by the functionally assigned divisions:

a. Written safety evaluations of changes in the facility as described in the Safety Analysis Report, of changes in procedures as described in the Safety Analysis Report, and of tests or experiments not described in the Safety Analysis Report, which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2). Such reviews need not be performed prior to implementation.

b. Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59(c). Matters of this kind shall be reviewed prior to submittal to the NRC.

c. Proposed changes to Technical Specifications or license amendments related to nuclear safety shall be reviewed prior to submittal to the NRC for approval.

d. Violations, deviations, and reportable events which require reporting to the NRC in writing. Such reviews are performed after the fact. Review of events covered under this subsection shall include results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Written summaries of audit reports in the areas specified in section 6.5.3 and involving safety related functions.

f. Any other matters involving safe operations of the nuclear power plant which a reviewer deems appropriate for consideration, or which is referred to the independent reviewers.

QUALIFICATIONS

6.5.2.6 The independent reviewer(s) shall either have a Bachelor's Degree in Engineering or the Physical Sciences and five (5) years of professional level experience in the area being reviewed or have 9 years of appropriate experience in the field of his specialty. An individual performing reviews may possess competence in more than one specialty area. Credit toward experience will be given for advanced degrees on a one-for-one basis up to a maximum of two years.

RECORDS

6.5.2.7 Reports of reviews encompassed in Section 6.5.2.5 shall be prepared, maintained and transmitted to the cognizant Corporate Officer.

RECORDS

6.5.3.3 Audit reports encompassed by sections 6.5.3.1 and 6.5.3.2 shall be forwarded for action to the management positions responsible for the areas audited within 30 days after completion of the audit. Upper management shall be informed per the Operation Quality Assurance Plan.

6.5.4 INDEPENDENT ONSITE SAFETY REVIEW GROUP (IOSRG)

STRUCTURE

6.5.4.1 The IOSRG shall be a full-time group of engineers experienced in nuclear power plant engineering, operation and/or technology, independent of the facility staff, and located onsite.

ORGANIZATION

6.5.4.2 a. The IOSRG shall consist of a Manager responsible for Nuclear Safety Assessment and staff members who meet the qualifications of 6.5.4.5. Group expertise shall be multidisciplined.

b. The IOSRG shall report to the Director responsible for Nuclear Safety Assessment.

FUNCTION

6.5.4.3 The periodic review functions of the IOSRG shall include the following on a selective and overview basis:

1) Evaluation for technical adequacy and clarity of procedures important to the safe operation of the facility.

2) Evaluation of facility operations from a safety perspective.

3) Assessment of facility nuclear safety programs.

4) Assessment of the facility performance regarding conformance to requirements related to safety.

5) Any other matter involving safe operation of the nuclear power plant that the manager deems appropriate for consideration.

AUTHORITY

6.5.4.4 The IOSRG shall have access to the facility and facility records as necessary to perform its evaluations and assessments. Based on its reviews, the IOSRG shall provide recommendations to the management positions responsible for the areas reviewed.

QUALIFICATIONS

6.5.4.5 IOSRG engineers shall have either (1) a Bachelor's Degree in Engineering or appropriate Physical Science and three years of professional level experience in the nuclear power field which may include technical supporting functions or (2) eight years of appropriate experience in nuclear power plant operations and/or technology. Credit toward experience will be given for advance degrees on a one-to-one basis up to a maximum of two years.

RECORDS

6.5.4.6 Reports of evaluations and assessments encompassed in Section 6.5.4.3 shall be prepared, approved, and transmitted to the Director and the Corporate Officer responsible for nuclear safety assessment, Vice President & Director Oyster Creek, and the management positions responsible for the areas reviewed.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50; and

b. Each REPORTABLE EVENT shall be reported to the cognizant manager and the cognizant division Vice President and the Vice President & Director Oyster Creek. The functionally cognizant division staff shall prepare a Licensee Event Report (LER) in accordance with the guidance outlined in 10 CFR 50.73(b). Copies of all such reports shall be submitted to the functionally cognizant Corporate Officer and the Vice President & Director Oyster Creek.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. If any Safety Limit is exceeded, the reactor shall be shut down immediately until the Commission authorizes the resumption of operation.

b. The Safety Limit violation shall be reported to the Commission and the Vice President and Director Oyster Creek.

c. A Safety Limit Violation Report shall be prepared. The report shall be submitted to the Vice President and Director Oyster Creek. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components systems or structures, (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission within ten days of the violation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.203

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR, INC. AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated July 21, 1998, GPU Nuclear, Inc., (the licensee) requested an amendment to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (OCNGS) and the Technical Specifications (TS) appended to the license. The requested changes consist of: (1) modifying Specification 6.2.2.2(a) to provide some flexibility to accommodate unexpected absence of on-duty shift crew members, (2) eliminate reference to the Manager, Plant Operations in Specification 6.2.2.2(j) as the position has been eliminated (3) reduce the maximum time in which to forward audit reports to the responsible manager from 60 to 30 days, (4) replace the term "Vice President" with the term "Corporate Officer" in several places in Section 6, and (5) correct several typographical errors.

2.0 EVALUATION

The TS for OCNGS do not specify an allowable time limit for a temporary deviation from the minimum shift staffing requirements in the Table in 10 CFR 50.54(m)(2)(i) as allowed by Table footnote 1. A temporary deviation is a reasonable consideration to account for unforeseen or unexpected absences due to illnesses or accidents or the need for some flexibility in meeting operational requirements or challenges. Standard Technical Specifications (STS) allow a time period of 2 hours to accommodate unexpected absences. The licensee's TS are silent regarding this flexibility, but do include a shift staffing requirement in 6.2.2.2.a. The licensee's strict interpretation of their TS does not allow them to take advantage of the flexibility of the footnote without being in violation of their TS. The licensee's request is reasonable and consistent with the STS and is acceptable.

The Shift Technical Advisor was not included in the shift staffing list in 6.2.2.2.a, but is required by 6.2.2.2.h. The staffing position is added to 6.2.2.2.a for consistency and to accommodate the added flexibility in shift staffing and is acceptable.

The position of Manager, Plant Operations was eliminated as a result of organizational changes. The duties and responsibilities of the eliminated position were assumed by the Plant Operations Director. The TS require the Plant Operations Director to be a licensed senior reactor operator. This change is consistent with existing standards and is acceptable.

The maximum time in which to forward audit reports encompassed by TS sections 6.5.3.1 and 6.5.3.2 to the management positions responsible for the areas audited was reduced from 60 to 30 days in TS 6.5.3.3. The audits in question are performed in accordance with the Oyster Creek Operational Quality Assurance Plan or under the cognizance of the Vice President responsible for technical support. The shortening of this records requirement does not impact safe operation of the facility and is not in conflict with regulatory requirements and is acceptable.

The remaining TS changes correct typographical errors or define and clarify corporate responsibilities or lines of authority. In several instances the use of the term "Corporate Officer" replaces a specific title which may have been deleted or changed as the result of organizational changes. These changes are acceptable.

Finally, included in this amendment request are updated TS Bases pages which are not subject to NRC approval.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 45525). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

Principal Contributor: R. Eaton

Date: November 30, 1998