June 4, 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 RE: CLEARIFICATION TO TECHNICAL SPECIFICATIONS (TAC NO. MA0280)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 195 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated December 10, 1997.

The amendment clarifies sections of the Technical Specifications which have been demonstrated to be unclear or conflicting as a result of a licensee review.

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. 195 to DPR-16
2. Safety Evaluation

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DATED: June 4, 1998

AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. DPR-16-OYSTER CREEK NUCLEAR GENERATING STATION

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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. 195
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 December 10, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-16 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 195 , 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 of issuance.

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: June 4, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 195

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
2.3-6	2.3-6
2.3-7	2.3-7
3.1-11	3.1-11
3.1-14	3.1-14
3.1-16	3.1-16
3.4-8	3.4-8
3.8-2	3.8-2
3.8-3	3.8-3
4.3-1	4.3-1
4.5-13	4.5-13
6-1	6-1

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of main steam isolation valve closure while operating at 1930 MWt, followed by (1) a reactor scram on high neutron flux, (2) failure of the recirculation pump trip on high pressure, (3) failure of the turbine bypass valves to open, and (4) failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 9 safety valves are required to turn the pressure transient. The ASME B&PV Code allows a ± 1% of working pressure (1250 psig) variation in the lift point of the valves. This variation is recognized in Specification 4.3.

The low pressure isolation of the main steam line at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. The low-pressure isolation protection is enabled with entry into IRM range 10 or the RUN mode. In addition, a scram on 10% main steam isolation valve (MSIV) closure anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. Bypass of the MSIV closure scram function below 600 psig is permitted to provide sealing steam and allow the establishment of condenser vacuum. Advantage is taken of the MSIV scram feature to provide protection for the low-pressure portion of the fuel cladding integrity safety limit. To continue operation beyond 12% of rated power, the IRM's must be transferred into range 10. Reactor pressure must be above 825 psig to successfully transfer the IRM's into range 10. Entry into range 10 at less than 825 psig will result in main steam line isolation valve closure and MSIV closure scram. This provides automatic scram protection for the fuel cladding integrity safety limit which allows a maximum power of 25% of rated at pressures below 800 psia. Below 600 psig, when the MSIV closure scram is bypassed, scram protection is provided by the IRMs.

Operation of the reactor at pressure lower than 825 psig requires that the mode switch be in the STARTUP position and the IRMs be in range 9 or lower. The protection for the fuel clad integrity safety limit is provided by the IRM high neutron flux scram in each IRM range. The IRM range 9 high flux scram setting at 12% of rated power provides adequate thermal margin to the safety limit of 25% of rated power. There are few possible significant sources of rapid reactivity input to the system through IRM range 9: effects of increasing pressure at zero and low void content are minor; reactivity excursions from colder makeup water, will cause an IRM high flux trip; and the control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. In the unlikely event of a rapid or uncontrolled increase in reactivity, the IRM system would be more than adequate to ensure a scram before power could exceed the safety limit. Furthermore, a mechanical stop on the IRM range switch requires an operator to pull up on the switch handle to pass through the stop and enter range 10. This provides protection against an inadvertent entry into range 10 at low pressures. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram.

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).

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.

References

(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

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function		Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Min. No. of OPERABLE or OPERATING [tripped] Trip Systems	Min. No. of Instrument Channels Per OPERABLE Trip System	Action Required*
D.	Core Spray								
1	Low-Low Reactor Water Level	**	X(t)	X(t)	X(t)	X	2	2(pp)	Consider the respective core spray loop inoperable and comply with Spec. 3.4
2	High Drywell Pressure	≤ 3.5 psig	X(t)	X(t)	X(t)	X	2(k)	2(k)(pp)	
3	Low Reactor Pressure (valve permissive)	≥ 285 psig	X(t)	X(t)	X(t)	X	2	2(pp)	
E.	Containment Spray								
	Comply with Technical Specification 3.4								
F.	Primary Containment Isolation								
1	High Drywell Pressure	≤ 3.5 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)(oo)	Isolate containment or PLACE IN COLD SHUTDOWN CONDITION
2	Low-Low Reactor Water Level	≥ 7'2" above TOP OF ACTIVE FUEL	X(u)	X(u)	X(u)	X	2	2(oo)	
G.	Automatic Depressurization								
1	High Drywell Pressure	≤ 3.5 psig	X(v)	X(v)	X(v)	X	2(k)	2(k)	See note h
2	Low-Low-Low Reactor Water Level	≥ 4'8" above TOP OF ACTIVE FUEL	X(v)	X(v)	X(v)	X	2	2	See note h
3	Core Spray Booster Pump d/p Permissive	>21.2 psid	X(v)	X(v)	X(v)	X	Note i	Note i	See note i.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function		Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Min. No. of OPERABLE or OPERATING [tripped] Trip Systems	Min. No. of Instrument Channels Per OPERABLE Trip System	Action Required*
M.	Diesel Generator Load Sequence Timers	Time delay after energization of relay							
1	CRD pump	60 sec ± 15%	X	X	X	X	2(m)	1(n)(kk)	Consider the pump inoperable and comply with Spec. 3.4.D (see Note q)
2	Service Water Pump (aa)	120 sec.± 15% (SK1A) (SK2A) 10 sec. ± 15% (SK7A) (SK8A)	X	X	X	X	2(o)	2(p)(kk)	Consider the pump inoperable and comply within 7 days (See Note q)
3	Reactor Building Closed Cooling Water Pump (bb)	166 sec.± 15%	X	X	X	X	2(m)	1(n)(kk)	Consider the pump inoperable and comply within 7 days (See Note q)
N.	Loss of Power								
a	4.16KV Emergency Bus Undervoltage (Loss of Voltage)	**	X(ff)	X(ff)	X(ff)	X(ff)	2	1(kk)	
b	4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	**	X(ff)	X(ff)	X(ff)	X(ff)	2	3(kk)	See note ee

TABLE 3.1.1 (CONT'D)

* Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. A channel may be placed in an inoperable status for up to six hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE instrument channel in the same trip system is monitoring that parameter.

** See Specification 2.3 for Limiting Safety System Settings.

NOTES:

a. Permissible to bypass, with control rod block, for reactor protection system reset in REFUEL MODE.

b. Permissible to bypass below 600 psig in REFUEL and STARTUP MODES.

c. One (1) APRM in each OPERABLE trip system may be bypassed or inoperable provided the requirements of Specification 3.1.C and 3.10.C are satisfied. Two APRMs in the same quadrant shall not be concurrently bypassed except as noted below or permitted by note.

   Any one APRM may be removed from service for up to six hours for test or calibration without inserting trips in its trip system only if the remaining OPERABLE APRMs meet the requirements of Specification 3.1.B.1 and no control rods are moved outward during the calibration or test. During this short period, the requirements of Specifications 3.1.B.2, 3.1.C and 3.10.C need not be met.

d. The IRMs shall be inserted and OPERABLE until the APRMs are OPERABLE and reading at least 2/150 full scale.

e. Offgas system isolation trip set at $\leq$2,000 mRem/hr. Air ejector isolation valve closure time delay shall not exceed 15 minutes.

f. Unless SRM chambers are fully inserted.

g. Not applicable when IRM on lowest range.

h. With one or more instrument channel(s) inoperable in one ADS trip system, place the relay contact(s) for the inoperable initiation signal in the tripped condition within 4 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

   With one or more instrument channel(s) inoperable in both ADS trip systems, restore ADS initiation capability in at least one trip system within 1 hour, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

   Relief valve controllers shall not be bypassed for more than 3 hours (total time for all controllers) in any 30-day period and only one relief valve controller may be bypassed at a time.

The containment spray system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. Actuation of the containment spray system in accordance with plant emergency operating procedures ensures that containment and torus pressure and temperature conditions are within the design basis for containment integrity, EQ, and core spray NPSH requirements. The flow from one pump in either loop is more than ample to provide the required heat removal capability(2). The emergency service water system provides cooling to the containment spray heat exchangers and, therefore, is required to provide the ultimate heat sink for the energy release in the event of a loss-of-coolant accident. The emergency service water pumping requirements are those which correspond to containment cooling heat exchanger performance implicit in the containment cooling description. Since the loss-of-coolant accident while in the cold shutdown condition would not require containment spray, the system may be deactivated to permit integrated leak rate testing of the primary containment while the reactor is in the cold shutdown condition.

The core spray main pump compartments and containment spray pump compartments were provided with water-tight doors(4). Specification 3.4.E ensures that the doors are in place to perform their intended function.

Similarly, since a loss-of-coolant accident when primary containment integrity is not required would not result in pressure build-up in the drywell or torus, the containment spray system may be made inoperable under these conditions.

References

1. NEDC-31462P, "Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis," August 1987.

2. Licensing Application, Amendment 32, Question 3

3. (Deleted)

4. Licensing Application, Amendment 18, Question 4

5. GPUN Topical Report 053, "Thermal Limits with One Core Spray Sparger" December 1988.

6. NEDE-30010A, "Performance Evaluation of the Oyster Creek Core Spray Sparger", January 1984.

7. Letter and enclosed Safety Evaluation, Walter A. Paulson (NRC) to P. B. Fiedler (GPUN), July 20, 1984.

8. APED-5736, "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", April 1969.

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 fo 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 tested 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.

References:

1. FDSAR, Volume I, Section IV-3
2. K. Shure and D. J. Dudziak, "Calculating Energy Release by Fission Products," U.S. AEC Report, WAPD-T-1309, March 1961.
3. K. Shure, "Fission Product Decay Heat," in U.S. AEC Report, WAPD-BT-24, December 1961.
4. Specification 3.2, Bases.
5. Specification 3.5.3.a.1.

4.3 REACTOR COOLANT

Applicability: Applies to the surveillance requirements for the reactor coolant system.

Objective: To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Materials surveillance specimens and neutron flux monitors shall be installed in the reactor vessel adjacent to the wall at the midplane of the active core. Specimens and monitors shall be periodically removed, tested, and evaluated to determine the effects of neutron fluence on the fracture toughness of the vessel shell materials. The results of these evaluations shall be used to assess the adequacy of the P-T curves A, B, and C in Figure 3.3.1, 3.3.2 and 3.3.3. New curves shall be generated as required.

B. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 systems and components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(g)(6)(i).

C. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR, Section 50.55a(f), except where specific written relief has been granted by the NRC pursuant to 10 CFR, Section 50.55a(f)(6)(i).

D. 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 in accordance with Article 5000, Section XI. The requirements of specification 3.3.A shall be met during the test.

E. Each replacement safety valve or valve that has been repaired shall be tested in accordance with subsection IWV-3510 of Section XI of the ASME Boiler and Pressure Vessel Code. Setpoints shall be as follows:

Number of Valves	Set Points (psig)
4	1212 $\pm$ 12
5	1221 $\pm$ 12

F. A sample of reactor coolant shall be analyzed at least every 72 hours for the purpose of determining the content of chloride ion and to check the conductivity.

of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening- closing cycle, the position indicating lights are designed to function as follows:

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

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

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

The in-place testing of charcoal filters is performed using halogenated hydrocarbon refrigerant which is injected into the system upstream of the charcoal filters. Measurement of the refrigerant concentration upstream and downstream of the charcoal filters is made using a gas chromatograph. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the test also gives an indication of the relative efficiency of the installed system. The test procedure is an adaptation of test procedures developed at the Savannah River Laboratory which were described in the Ninth AEC Cleaning Conference.*

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential releases of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP at testing medium.

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

ADMINISTRATIVE CONTROLS

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

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195

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

By letter dated December 10, 1997, GPU Nuclear, Inc. (the licensee) requested an amendment to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek) and the Technical Specifications (TS) appended to the license. The requested changes consist of administrative changes to clarify potentially unclear and ambiguous wording in the Bases and to correct other errors noted as a result of a systematic review by the licensee.

The proposed TS modifications will:

1. reword TS Bases 2.3 to more clearly define the justification for the existing limits,
2. correct a typographical error in Table 3.1.1 at G.1,
3. correct a typographical error in Table 3.1.1 at M.2,
4. correct a typographical error in Section 4.3.C,
5. change a position title in Section 6.1.1,
6. change the scram bypass reactor pressure setpoint in Table 3.1.1, note b,
7. change Section 3.4 Bases,
8. change Section 3.8 Bases, and
9. change Section 4.5 Bases.

2.0 EVALUATION

2.1 Proposed change: Change the wording in Specification 2.3 Bases which more clearly defines the justification for the existing limits.

Evaluation: This change to the Bases expands and clarifies the protections associated with low pressure isolations and makes the Bases consistent with the changes associated with proposed change 2.6. The staff finds this acceptable.

2.2 Proposed change: Change the setpoint in Table 3.1.1, item G.1 from <3.5 to ≤3.5.

Evaluation: The High Drywell Pressure setpoint in Table 3.1.1, item G.1 was inadvertently changed from ≤3.5 to <3.5 as a result of a typographical error when issuing Amendment 171 to the facility TSs. The correct value of the set point is ≤3.5, which was established as a result of Amendment 112 to the facility TSs and as reflected in the Updated Final Safety Analysis Report (UFSAR), Table 7.3-1, item 7 and 13. This change corrects the typographical error and is acceptable.

2.3 Proposed change: Change the location of the relay designated SK2A in the 10 second timer column in Table 3.1.1, item M.2, to the 120 second timer column.

Evaluation: As a result of a typographical error in issuing Amendment 160 to the facility TSs, relay designator SK2A, was mistakenly placed in the column under 10 second timer instead of in the column under 120 second timer. This change will correct a typographical error and is acceptable.

2.4 Proposed change: In Section 4.3.C, change 10 CFR 50.55a(g)(6)(I) to 10 CFR 50.55a(f)(6)(I).

Evaluation: A typographical error was introduced when Amendment 82 was issued to the facility TSs. Changing 10 CFR 50.55a(g)(6)(I) to 10 CFR 50.55a(f)(6)(I) will correct the typographical error and is acceptable.

2.5 Proposed change: In Section 6.1.1, change the title of the Deputy Director to Director-Operation and Maintenance.

Evaluation: The proposed change is the result of a change in management title only and does not represent a change in any of the duties or responsibilities of the position previously described in the TS and is acceptable.

2.6 Proposed change: Change the scram bypass reactor pressure setpoint in Table 3.1.1, note b, from <800 psia to <600 psig.

Evaluation: The set point was changed prior to restart from refueling outage 10R as a result of the lower bound of the Minimum Critical Power Ratio (MCPR) safety limit pressure range being raised from 600 psig to 800 psia. As part of the Cycle 10 reload submittal the TS Table 3.1.1, note b was changed from <600 psig to <800 psia, however, the TS Bases specifying <600 psig was not changed. An evaluation was completed to justify raising the MCPR lower bound but an evaluation to raise the set point was not done and in fact the licensee maintained the <600 psig value through use of procedures and standing orders.

Plant ooerations have not been impacted by not having use of the < 800 psia value and it is more conservative in that it provides a lower level below which protection can be passed. Bypassing the low condenser vacuum and main steam isolation valve closure anticipatory scrams is permitted to allow for establishing turbine seals and condenser vacuum while starting up with reactor pressure below the predetermined setpoint value. The staff finds the lower value of <600 psig acceptable.

2.7 Proposed change: Removal of a paragraph from Section 3.4 Bases

Evaluation: The referenced paragraph describes the ability of the Control Rod Drive System (CRD) pumps to provide high pressure injection capabilities for Small Break Loss of Coolant Accidents (LOCA) below the size of .002 square feet. The CRD pumps are not safety related and no credit is taken for them in their Appendix K analysis. Small break LOCA analysis are calculated down to .05 square feet which is well beyond the capabilities of the CRD pumps. The removal of the paragraph is acceptable to the staff.

2.8 Proposed change: Revision to the description of the Isolation Condenser System in Bases Section 3.8 to reflect the removal of the radiation monitors.

Evaluation: The Isolation Condenser radiation monitors were removed pursuant to 10 CFR 50.59. Their removal had no impact on the operation of any plant system and they were not relied upon for any post-accident evaluations. Accordingly, the Bases is updated to reflect this change in the system. The staff finds this acceptable.

2.9 Proposed change: Revision of Bases Section 4.5 to remove reference to a brand specific product.

Evaluation: The current bases page in Section 4.5 contains the name of a brand specific chemical used in the performance of a surveillance test. This change will remove the reference to the specific brand name whereby allowing the licensee to use other brands and equivalent chemicals. The staff finds this acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 (62 FR 4313). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

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