August 20, 1990

Docket No. 50-219

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

Dear Mr. Fitzpatrick:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 76760)

The Commission has issued the enclosed Amendment No. 141 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated May 4, 1990.

The amendment revises the Technical Specifications to accommodate implementation of a 21 month operating cycle with a 3 month outage, or a 24 month plant refueling cycle for those Technical Specification surveillances which will expire prior to the currently scheduled 13R refueling outage.

With respect to Technical Specification 4.5.E, Type "B" and "C" Local Leak Rate Tests (LLRT), the staff has concerns that the request to extend to a 24 month operation cycle does not allow sufficient time for unscheduled shutdowns within the cycle. This, we believe, will result in a high frequency of one time exemption requests from the 24 month requirement. Therefore, we recommend that you seek and provide the basis for a permanent exemption from the Appendix J regulatory requirement of 24 months to some more appropriate time period to avoid plant shutdowns or one time requests for exemptions because of the 24 month restriction.

A copy of the related Safety Evaluation is enclosed. Also enclosed is a copy of the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 141 to DPR-16
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 141
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation, et al., (the licensee), dated May 4, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 141 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 20, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 141

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
Page 4.1-6	Page 4.1-6
Page 4.1-8	Page 4.1-8
Page 4.2-2	Page 4.2-2
Page 4.5-3	Page 4.5-3
Page 4.5-6	Page 4.5-6
Page 4.5-9	Page 4.5-9
Page 4.5-11	Page 4.5-11
Page 4.7-1	Page 4.7-1
Page 6-16	Page 6-16

TABLE 4.1.1 (cont'd)

Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test and Calibration)
11. APRM Level	N/A	1/3 d	N/A	Output adjustment using operational type heat balance during power operation
APRM Scram Trips	Note 2	1/wk.	1/wk.	Using built-in calibration equipment during power operation
12. APRM Rod Blocks	Note 2	1/3 mo.	1/mo.	Upscale and downscale
13.a. High Radiation in Main Steamline	1/s	1/3 mo.	1/mo.	Using built-in calibration equipment during power operation
b. Sensors for 13(a)	N/A	Each refueling outage	N/A	Using external radiation source
14. High Radiation in Reactor Building				
Operating Floor	1/s	1/3 mo.	1/wk	Using gamma source for calibration
Ventilation Exhaust	1/s	1/3 mo.	1/wk.	Using gamma source for calibration
15. High Radiation on Air Ejector		1/3 mo.	1/wk.	Using built-in calibration equipment
Ejector Off-Gas	1/s			Channel check
	1/mo.			Source check
		1/24 mo.		Calibration according to established station calibration procedures
			1/24 mo.	Note a
16. IRM Level	N/A	Each startup	N/A	
IRM Scram	*	*	*	Using built-in calibration equipment

Amendment No.: ~~Change 7~~ 65, 74, 108, 141

TABLE 4.1.1 (cont'd)

Instrument Channel	Check	Calibrate	Test	Remarks (Applies To Test and Calibration)
27. Scram Discharge Volume (Rod Block)				
a) Water level high	N/A	Each refueling outage	Every 3 months	By varying level in switch column
b) Scram Trip bypass	N/A	N/A	Each refueling outage	
28. Loss of Power				
a) 4.16 KV Emergency Bus Undervoltage (Loss of voltage)	Daily	1/24 mos.	1/mo.	
b) 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	Daily	1/24 mos.	1/mo.	
29. Drywell High Radiation	N/A	Each refueling outage	Each refueling outage	

Amendment Nos.: 63, 80, 106, 116, 141
Change: 5,7

* Calibrate prior to startup and normal shutdown and thereafter check 1/s and test 1/wk until no longer required.

Legend: N/A = Not Applicable; 1/s = Once per shift; 1/d = Once per day; 1/3d = Once per three days; 1/wk = Once per we
1/3 mo = Once every 3 months; 1/18 mos. = Once every 18 months, 1/24 = Once per 24 months

The following notes are only for Item 15 of Table 4.1.1:
A channel may be taken out of service for the purpose of a check, calibration, test or maintenance without declaring the channel to be inoperable.

a. The channel functional test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1) Instrument indicates measured levels above the alarm setpoint.
2) Instrument indicates a downscale failure.
3) Instrument controls not set in operate mode.
4) Instrument electrical power loss.

3.	Functional test	Once every 24 months
4.	Solution volume and temperature check	Once/day
5.	Solution Boron-10 enrichment	Each refueling outage. Enrichment analyses shall be received no later than 30 days after startup from the refueling outage. If not received within 30 days, notify NRC (within 7 days) of plans to obtain test results.

F. At specific power operation conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed with equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

G. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days, except in shutdown mode*, and shall be cycled at least one complete cycle of full travel at least quarterly.

H. All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE. This will be done at least once per refueling cycle by placing the mode switch in shutdown and by verifying that:

a. The drain and vent valves close within 30 seconds after receipt of a signal for control rods to scram, and

b. The scram signal can be reset and the drain and vent valves open when the scram discharge volume trip is bypassed.

BASIS: The core reactivity limitation (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration is performed with the reactor core in the cold, xenon-free condition and will show

* These valves may be closed intermittently for testing under administrative control.

D. Frequency

1. Three Type "A" overall Integrated Containment Leakage Rate Tests shall be conducted at approximately 40 month intervals during scheduled shutdowns within each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If two consecutive periodic Type A tests fail to meet the acceptance criteria, the subsequent Type A test shall be performed at each shutdown for refueling or approximately every 18 months whichever occurs first. This schedule will remain in effect until two consecutive Type A tests meet the acceptance criteria, at which time the frequency of testing noted in D.1 above may be resumed.

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

1. Primary containment testable penetrations (Type "B" Test) and isolation valves (Type "C" Test), except as stated below, shall be tested at a pressure of least 35 psig (P_a) at intervals not to exceed 24 months.

2. The main steam line isolation valves shall be tested at a pressure of at least 20 psig at intervals not to exceed 24 months to determine if corrective action is required.

3. Isolation valve, Type "C", tests shall have each valve closed by normal operation. (e.g. no preliminary exercising or tightening of valve after closures by valve motor).

4. Bolted double gasketed seals shall be tested whenever the seal is closed after being opened, and at intervals not to exceed 24 months.

5. The drywell airlock shall be demonstrated operable by performing the following tests:

4. Reactor Building to Suppression Chamber Vacuum Breakers

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

b. Once every 24 months each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker from closed to fully open does not exceed the force specified in Specification 3.5.A.4.a. The air-operated vacuum breaker instrumentation shall be calibrated once every 24 months.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Periodic Operability Tests

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

b. Tests - Once Every 24 Months

(1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) The suppression chamber - drywell vacuum breaker position indication and alarms systems shall be calibrated and functionally tested.

(3) At least four of the suppression chamber - drywell vacuum breakers shall be inspected. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected such that Specifications 3.5.A.5.a can be met.

c. Tests - Once Every 20 Months

A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of air flow through a 2-inch orifice.

K. Reactor Building

1. Secondary containment capability tests shall be conducted after isolating the reactor building and placing either Standby Gas Treatment System filter train in operation.

2. The tests shall be performed at least once per operating cycle and shall demonstrate the capability to maintain a ¼ inch of water vacuum under calm wind conditions with a Standby Gas Treatment System Filter train flow rate of not more than 4000 cfm.

Q. Shock Suppressors (Snubbers)

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

a. Visual Inspections

All snubbers shall be visually inspected in accordance with the following schedule:

No. Inoperable Snubbers Per Inspection Period	Subsequent Visual Inspection Period*
0	24 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

*The provisions of Technical Specification 1.24 are not applicable.

The required inspection interval shall not be lengthened more than one step at a time. The snubbers may be categorized into two groups: those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not seized. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that the affected snubber is functionally tested in the as-found condition and determined operable per Specification 4.5.Q.d or 4.5.Q.e as applicable and that the cause for the rejection has been clearly established and remedied for that particular snubber.

c. Functional Tests

At least once every 24 months, a representative sample (10% of the total of each type of snubber in use in the plant) shall be functionally tested either in place or in a bench test. For each snubber that

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

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

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

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

f. Snubber Service Life Monitoring

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

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

4.7 AUXILIARY ELECTRICAL POWER

Applicability: Applies to surveillance requirements of the auxiliary electrical supply.

Objective: To verify the availability of the auxiliary electrical supply.

Specification: A. Diesel Generator

1. Each diesel generator shall be started and loaded to not less than 20% rated power every two weeks.

2. The two diesel generators shall be automatically actuated and functionally tested once every 24 months. This shall include testing of the diesel generator load sequence timers listed in Table 3.1.1.

3. Each diesel generator shall be given a thorough inspection at least once per 24 months during shutdown.

4. The diesel generators' fuel supply shall be checked following the above tests.

5. The diesel generators' starting batteries shall be tested and monitored the same as the station batteries, Specification 4.7.b.

B. Station Batteries

1. Weekly surveillance will be performed to verify the following:

   a. The active metallic surface of the plates shall be fully covered with electrolyte in all batteries,

   b. The designated pilot cell voltage is greater than or equal to 2.0 volts and

   c. The overall battery voltage is greater than or equal to 120 volts (Diesel battery; 112 volts).

   d. The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.190.

2. Quarterly Surveillance will be performed to verify the following:

   a. The active metallic surface of the plates shall be fully covered with electrolyte in all batteries.

   b. The voltage of each connected cell is greater than or equal to 2.0 volts under float charge and

activities within the area and who will perform periodic radiation surveillance at the frequency in the RWP. The surveillance frequency will be established by the Director responsible for Radiological Controls.

6.13.2 Specification 6.13.1 shall also apply to each high radiation area in which the intensity of radiation is greater than 1,000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of operations and/or radiation protection supervision on duty.

6.14 ENVIRONMENTAL QUALIFICATION

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License DPR-16 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.15 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1) Provisions establishing preventative maintenance and periodic visual inspection requirements, and

2) System leak test requirements, to the extent permitted by system design and radiological conditions, for each system at a frequency of once every 24 months. The systems subject to this testing are (1) Core Spray, (2) Containment Spray, (3) Reactor Water Cleanup, (4) Isolation Condenser, and (5) Shutdown Cooling.

6.16 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas* under accident conditions. This program shall include the following:

a. Training of personnel,

*Areas requiring personnel access for establishing hot shutdown condition.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141

TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated May 4, 1990, GPU Nuclear Corporation (GPUN/the licensee) proposed to revise the Technical Specifications to accommodate implementation of a 21 month operating cycle with a 3 month outage, or a 24 month plant refueling cycle for those surveillances which will expire prior to the currently scheduled 13R refueling outage. The staff's evaluation of the requested changes based on its review of the supporting information is given below.

2.0 EVALUATION

2.1 High Radiation on Air Ejector Off-Gas Instrument Calibration

Proposed Change

Technical Specification (TS) Table 4.1.1 Item 15, is revised to extend the High Radiation on Air Ejector Off-Gas instrument channel calibration and test from each refueling outage to once per 24 months.

Evaluation

TS Table 4.1.1, Item 15, "High Radiation on Air Ejector Off-Gas Instrumentation Calibration" is currently specified to be performed each refueling outage. This surveillance test verifies detector sensitivity to a known radioactive source. The air ejector off-gas consists of hydrogen and oxygen from radiolytic decomposition of water, air in leakage, water vapor and radioactive fission gases. These gases are drawn from the main condenser through the steam air ejectors. The air ejector off-gas radiation monitoring system monitors and records the radiation level of the noncondensible gases removed by the steam jet air ejectors. This system also initiates closure of the off-gas system isolation valves when radiation levels exceed the setpoint.

The proposed change extends the surveillance interval from 20 months to 24 months for the air ejector off-gas radiation monitoring system. To justify the extension, the licensee evaluated the surveillance results as well as maintenance history over the period 1977 through 1986. This review did not find any instance where the surveillance acceptance criteria was not met. Additionally, detector drift during each 20 month calibration interval was found to be acceptable. The TS will still provide for channel checks once per shift to verify instrument channel operability. The off-gas radiation monitoring system has also been modified by the replacement of the analog Log Radiation Monitors (LRM) with a General Electric (GE) NUMAC LRM. The GE NUMAC LRM has several advantages over the previous analog units, such as the functional circuitry is continually being self-checked in addition to the scheduled surveillance testing and the test circuitry is hard wired in place which eliminates any temporary modifications of the functional circuitry during testing. Also, the NUMAC LRM drift rate is less than the analog log radiation monitors it replaces.

Based on the above, the staff agrees with the licensee's conclusion that an increase in the surveillance interval from 20 to 24 months is justified. The staff also agrees that the new NUMAC LRM will enhance the performance of the air ejector off-gas radiation monitoring system.

Conclusion

The proposed 24 month surveillance interval for high radiation on air ejector off-gas instrumentation is acceptable to the staff. The maintenance and surveillance history performed by the licensee confirmed the reliable operation of the air ejector off-gas monitoring system. The off-gas radiation monitoring system now utilizes a GE NUMAC LRM which includes self diagnostic functions and improved drift characteristics. The TS will still provide for shift channel checks and monthly channel tests which provides added assurance of channel operability. Based on the above evaluation the proposed surveillance interval for Table 4.1.1, Item 15 is acceptable to staff.

2.2 Loss of Power Instrument Channel Calibration

Proposed Change

TS Table 4.1-1, Items 28.a and 28.b, are revised to extend the 4.16Kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage) instrument channel calibrations from once per 18 months to once per 24 months.

TS Table 4.1.1, "Legend" is also revised to include the designation: 1/24 mo.= Once every 24 months.

Evaluation

TS Table 4.1-1, Items 28a and 28b, Loss of Power Instrument Channel Calibration are currently specified to be performed once every 18 months. The proposed TS revision would revise this interval to 24 months. The "loss of power" relays monitor voltage on the 4.16Kv emergency buses. Upon loss of voltage or degraded grid voltage these buses are unloaded and the diesel generators are started. The calibration test is performed to verify the setpoints for the loss of voltage or degraded voltage relay setpoints. The licensee reviewed calibration test results for the years 1985, 1987 and 1988 and found the results to be acceptable for GE IAV 53k loss of voltage relays. The type 27H degraded grid voltage relays were found to have excessive drift and have since been replaced with type 27N relays. Based on drift data referenced by the licensee, the type 27N relay will remain within the TS limit for the proposed surveillance interval extension. The licensee also stated that the voltage logic is two out of three and thus a malfunction in one relay would not effect the operation of the instrument channel. The staff agrees that this arrangement provides for added reliability, although the original design bases does not itself provide additional justification for a surveillance extension. However, both the loss of voltage and the degraded voltage relays are checked daily and tested monthly as referenced in the TS. These surveillance requirements are not being revised and will continue to provide added assurance of channel operability.

TS Table 4.1-1 Legend is also revised to include the designation: 1/24 mo. = once every 24 months to reflect the changes in the TS Table. We find this acceptable.

Based on the above licensee evaluations of surveillance and maintenance history and the referenced 27N relay drift rates, the staff finds the surveillance interval extension from 18 to 24 months for loss of voltage and degraded voltage relays to be acceptable.

Conclusion

The proposed 24 month surveillance interval for loss of voltage instrumentation is acceptable to the staff. The maintenance and surveillance history evaluation performed by the licensee confirmed the reliable operation of the loss of voltage instrumentation. The only exception noted during the licensee evaluation was for excessive drift of the 27H degraded voltage relay. These relays were subsequently replaced with 27N relays which exhibit less drift and meet the requirements of the proposed TS revision. The TS will still provide for daily channel checks and monthly channel tests which provides added assurance of channel operability. Based on the above evaluation, the proposed surveillance interval of 24 months for Table 4.1-1, Items 28a and 28b,is acceptable to the staff.

2.3 Reactivity Control System - Standby Liquid Control System

Proposed Change

TS Section 4.2.E.3 is revised to extend the Standby Liquid Control System Functional Test from each refueling outage to once every 24 months.

Evaluation

The Standby Liquid Control System (SLCS) is designed to bring the reactor to a shutdown condition at any time in core life independent of the control rod capabilities. The safety function of the periodic surveillance testing is to assure that the SLCS will perform as designed if it is needed during a plant emergency. The proposed change will extend the SLCS functional test interval from 20 months to 24 months. The SLCS functional test verifies the following:

(1) Operability of the SLCS by manually initiating pump start
(2) Pump running indication
(3) Corresponding squibb valve with a fired indication
(4) Flow indication
(5) Annunciator alarm
(6) Automatic reactor water cleanup system isolation upon a signal from the flow indicating switch

The instrumentations used in the functional test are not part of the TS. They are only there to monitor the success of the functional test which will remain the same after the extended interval.

An operating history of SLCS functional tests demonstrated adequate operation of the system for the last ten years. Pump operability is verified once every month in accordance with the current TS and quarterly as per ASME Section XI-IST requirements. The explosives valves are not normal operating valves. They are only used in the case of emergency. They are purchased in lots with a sample test prior to installation. The vendor performs the testing in lots which could identify any defect in the component before installation. The valve's primer and trigger mechanism have a shelf life of five years which could accommodate a 24-month testing interval and would provide assurance of their operability in the case of emergency.

Based on this information, we find that the capability of the SLCS to reliably perform its safety function is not likely to be impacted by the extension of the functional test interval from 20 to 24 months.

Conclusion

We find the proposed TS change to extend the refueling outage from 20 months to 24 months is acceptable based on the adequate operating history of the system seen through functional tests and the TS current monthly operability testing requirements of mechanical equipment which provide assurance of component availability during the plant operation. This change is expected to have no significant effect on the ability of the system discussed above to perform its safety functions reliably.

2.4 Type "B" and "C" Local Leak Rate Tests (LLRT)

Proposed Change

TS Section 4.5.E is revised to extend the Type "B" and "C" Local Leak Rate Tests (LLRT) from each refueling outage to an interval not to exceed 24 months.

Evaluation

TS Section 4.5.E currently specifies that Type "B" (primary containment penetration) and Type "C" (isolation valves) containment local leak rates test shall be performed each refueling outage. TS Section 4.5.G further specifies that the LLRT shall be performed as stated in Section 4.5.E, but in no case may exceed intervals of 24 months. This specified maximum interval of 24 months is consistent with the requirements of 10 CFR Part 50, Appendix J, Primary Reactor Containment Leakage Testing, Subsections III.D.2 and D.3, which specifies that Type "B" and "C" tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years. Thus, the extension of the refueling outage from 20 months to 24 months is within the existing TS frequency requirements for the LLRT and within the regulations pertaining to the LLRT as specified in 10 CFR Part 50, Appendix J. Therefore, the proposed change has no effect on the specified frequency for performing LLRT.

Conclusion

The staff has reviewed the licensee's proposed change to extend the Type "B" and "C" LLRT from each refueling outage to an interval not to exceed 24 months. Based on the review, the staff concludes that the licensee's justification to support the proposed change to Section 4.5.E to accommodate the implementation of a 24 month refueling cycle is adequate. There were no technical concerns identified and all regulatory requirements were met, therefore, the extension of the surveillance testing period for TS 4.5.E to 24 months is acceptable.

2.5 Reactor Building to Suppression Chamber Vacuum Breakers

Proposed Change

TS Section 4.5.5.J.b is revised to extend the Reactor Building to Suppression Chamber Vacuum Breakers test and instrument calibration from each refueling outage to once every 24 months.

Evaluation

TS Section 4.5.J.4.b specifies that the Reactor Building to Suppression Chamber Vacuum Breakers shall be tested each refueling outage and the air-operated vacuum breaker instrumentation shall be calibrated each refueling outage. This surveillance verifies the force required to open the vacuum breaker from closed to fully open, and calibrates the vacuum breaker actuation instrumentation.

The proposed change will extend the interval between successive tests from 20 months to 24 months. Evaluation of test results has indicated no degradation of valve parts. TS Section 4.5.J.4.a requires that the Reactor Building to Suppression Chamber Vacuum Breakers and associated instrumentation, including setpoint be checked for proper operation every 3 months. The proposed refueling outage interval change from 20 to 24 months will have no effect on the Reactor Building to Suppression Chamber Vacuum Breakers and actuation instrumentation since the vacuum breakers and instrumentation have demonstrated reliable operation, and quarterly operability testing in accordance with TS requirements provides additional assurance of availability between refueling inspections. Therefore, the proposed change has no effect on the safety function of the Reactor Building to Suppression Chamber Vacuum Breakers and actuation instrumentation.

Conclusion

The staff has reviewed the licensee's proposed change to extend the Reactor Building to Suppression Chamber Vacuum Breakers test from each refueling outage to once every 24 months. Based on the review, the staff concludes that the licensee's justification to support the proposed extension to 24 months is adequate. There were no technical concerns identified and all regulatory requirements were met therefore, the extension of the surveillance testing period for TS 4.5.J.4.b is acceptable.

2.6 Pressure Suppression Chamber-Drywell Vacuum Breaker Test

Proposed Change

TS Section 4.5.J.5.b(1), (2) and (3) are revised to extend the Suppression Chamber-Drywell Vacuum Breakers test, position indication and alarms calibration and test, and inspections from each refueling outage to once every 24 months.

Evaluation

TS Section 4.5.J.5.b (1), (2), and (3), currently specify that all suppression chamber-drywell vacuum breakers shall be tested each refueling outage, the suppression chamber-drywell vacuum breaker position indication and alarms shall be calibrated and tested each refueling outage, and at least four of the suppression chamber-drywell vacuum breakers shall be inspected each refueling outage. This surveillance verifies the force required to open each valve from fully closed to fully open, functionally tests and calibrates alarms and position indication instrumentation, and inspects for valve deficiencies.

The proposed change will extend the interval between successive tests from 20 months to 24 months. Evaluation of test results have indicated no degradation of valve parts. TS 4.5, Subsection J.5.a, requires each operable suppression chamber-drywell vacuum breakers exercised once each month and following any release of energy which would tend to increase pressure to the suppression chamber. Operation of position switches, indicators and alarms are also

verified monthly by operation of each vacuum breaker in accordance with TS requirements. The proposed refueling outage interval change from 20 to 24 months will have no effect on the suppression chamber-drywell vacuum breakers and associated instrumentation since the vacuum breakers and instrumentation have demonstrated reliable operation, and monthly operability testing is performed in accordance with TS requirements which will provide additional assurance of availability between refueling inspections. Therefore, the proposed change has no effect on the safety function of the suppression chamber-drywell vacuum breakers.

Conclusion

The staff has reviewed the licensee's proposed change to extend the suppression chamber-drywell vacuum breakers test, position indication and alarms calibration and test and inspections from each refueling outage to once every 24 months. Based on the review the staff concludes that the licensee's justification to support the proposed extension to 24 months is adequate. There were no technical concerns identified and all regulatory requirements were met, therefore, the extension of the surveillance testing period for TS 4.5.J.5.b(1), (2) and (3) is acceptable.

2.7 Drywell to Suppression Chamber Leak Rate Test

Proposed Change

TS Section 4.5.J.5.b(4) is clarified to specify that the Drywell to Suppression Chamber leak rate test shall be performed each refueling outage (interval not to exceed 20 months as presently required).

Evaluation and Conclusion

TS 4.5.J.5.b(4) currently specifies that a drywell to suppression chamber leak rate test shall be performed each refueling outage. This specification has been renumbered as TS 4.5.J.5.c. TS 4.5.J.5.c specifies that the leak rate test be performed once every 20 months. TS 4.5.J.5.b(4) specified that the leak rate be performed each refueling outage (interval not to exceed 20 months). Therefore, the requirements have been maintained the same as the original specification. On this basis, the staff finds the change acceptable.

2.8 Snubber Testing

Proposed Change

TS 4.5.Q.1.a is revised to extend the subsequent visual inspection period from 18 months to 24 months when zero inoperable snubbers are detected per inspection period. TS Section 4.5.Q.1.c is revised to extend the functional test of 10% of each type of snubber in the plant from each refueling cycle to every 24 months. TS Section 4.5.Q.1.f is revised to extend the review of installation and maintenance records from 18 months to 24 months.

Evaluation

The visual examination results for snubbers of Oyster Creek over the period 1978 to 1988 indicate two (2) individual mechanical snubber failures, and three (3) individual hydraulic snubber failures. Evaluation of the functional test results, performed on 10% of the total population, over the period 1983 to 1989 indicate no failures of mechanical snubbers, and only one (1) individual hydraulic snubber failure. The proposed change from 18 months to 24 months and from each refueling cycle (20 months) to 24 months will have no effect on snubber reliability since both mechanical and hydraulic snubbers have demonstrated reliable operation over the periods cited above. This data provides a sufficient basis to conclude that the snubber program of Oyster Creek is effective in minimizing snubber failure.

In addition, the procedural requirements in the TS for snubber surveillance will result in a more frequent examination schedule if the failure rate per examination period increases, and is therefore self-correcting. The extension of the installation and maintenance record reviews maintains the snubber service life monitoring program consistent with the visual inspection and functional test periods and does not affect snubber operability. This program will continue to ensure that the indicated service life will not be exceeded prior to the next schedule review. Therefore, the proposed change has no effect on the safety function of the snubbers.

Although the changes in examination intervals being proposed by the licensee do not appear in the current ASME/ANSI documents on the subject (ASME/ANSI OM-1987, Part 4, including OMc-1990 Addenda), they are consistent with the draft revision of this document which is scheduled to be published next year.

Conclusions

Based on staff review and on considerations discussed above, the staff concludes that there is reasonable assurance that the safety of the public will not be endangered by extending the intervals between successive visual inspections from 18 months to 24 months, successive functional tests from 20 months to 24 months, and successive maintenance and installation record reviews from 18 months to 24 months as proposed by the licensee and the proposed changes are, thus, acceptable.

2.9 Diesel Generator Testing

Proposed Change

TS 4.7.A.2 is revised to extend the emergency diesel generator automatic actuations and functional test from each refueling outage to once every 24 months. TS Section 4.7.A.3 is revised to extend the diesel generator inspection from once per 18 months to once per 24 months.

Evaluation

The proposed TS 4.7.A.2 will extend the interval between functional testing of the diesel generators from 20 months to 24 months, and TS 4.7.A.3 will extend the interval between diesel generator inspections from 18 months to 24 months. The licensee states that their evaluation of diesel generator test results has indicated that reliability is well within NRC guidelines and that they have committed to maintaining a 0.975 target reliability in response to the Station Blackout Rule. The actual operating time of approximately 120 hours per year for a diesel generator is far less than the 2000 hours specified by the manufacturer. The diesel generator manufacturer has stated that the increased maintenance and inspection interval is acceptable.

The proposed TS would not change the testing of the diesel generators every 2 weeks to demonstrate that they can be started and loaded to at least 20% of rated power. This should disclose any substantial decrease in reliability should any occur from the extension in the inspection and maintenance interval. For this reason, and the factors discussed above, we find the extension in the functional testing and inspection intervals to be acceptable.

The proposed TS 4.7.A.5 would continue to require the diesel generators' starting batteries to be tested and monitored the same as the station batteries. A TS change pertaining to the testing of the station batteries is still under consideration. However, we find that testing and monitoring of the diesel generators' starting batteries should be consistent with the testing monitoring of the station batteries. We therefore find this TS to be acceptable.

Conclusion

A proposed change to the Oyster Creek TS would lengthen the intervals between successive functional and inspection tests of the emergency diesel generators. The staff has reviewed the proposed TS changes and concludes that they would not adversely affect the reliability of the safety systems to perform their safety function. We therefore find the proposed TS changes to be acceptable.

2.10 Integrity of Systems Outside Containment

Proposed Change

TS Section 6.15, Subsection (2), is revised to extend performance of system leak tests for the Core Spray Containment Spray, Reactor Water Cleanup, Isolation Condenser, and Shutdown Cooling Systems from refueling cycle intervals to a frequency of once every 24 months.

Evaluation

TS 6.15, Subsection (2), currently requires performance of system leak tests at a frequency not to exceed refueling cycle intervals for the following systems:

a. Core Spray
b. Containment Spray
c. Reactor Water Cleanup
d. Isolation Condenser
e. Shutdown Cooling

This surveillance implements the program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The proposed change will extend the interval between successive inspections from 20 months to 24 months. Evaluation of test results for the Core Spray System, Containment Spray System, Isolation Condenser System and Shutdown Cooling System over the period of 1980 thru 1988 indicates only minor leakage was identified in these systems with the exception of the Shutdown Cooling System in which one test in 1989 resulted in unacceptable system external leakage.

With respect to the Reactor Water Cleanup System (RWCU), evaluation of test results over the period of 1982 thru 1989 indicate several instances when the 0.20 gpm leak rate inspection criteria was not met as a result of valve leakage. The 0.20 gpm external leakage criteria represents 20% of the total integrated ESF Systems leak rate assumed for control room habitability analyses. The ESF leak rate is the smallest contributor to the dose consequences to the control room operators following a design basis LOCA. In each case maintenance requests were initiated to address the leaking component. No major source of leakage capable of challenging system operability had been recorded. In addition, the RWCU is visually inspected for leaks periodically during power operation.

Conclusion

Based on the licensee's operating experience and inspection results of the Core Spray System, Containment Spray System, Isolation Condenser System, and Shutdown Cooling System during the period 1980 thru 1988, the TS changes to these systems as discussed above have been found acceptable. Also, based on the licensee's operating and inspection experience, and test results of the Reactor Water Cleanup System during the period 1982 thru 1989 and the licensee's actions taken to repair the source (valve) of leakage as the result of leak rate inspection criteria not being met, the TS changes as discussed have been found to be acceptable.

The staff concludes that the proposed TS changes will not affect the safety functions of the systems discussed above in the Oyster Creek Nuclear Generating Station.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on August 8, 1990 (55 FR 32321). Accordingly, based upon the environmental assessment, we have determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Dated: August 20, 1990

Principal Contributors: C. Dout, J. Harold, K. DeSai, T. McLellan, A. Toalston and J. Rajan

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

NOTICE OF ISSUANCE OF AMENDMENT TO

PROVISIONAL OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. to Facility Operating License No. DPR-16 issued to GPU Nuclear Corporation (the licensee), which revised the Technical Specifications for operation of the Oyster Creek Nuclear Generating Station located in Ocean County, New Jersey. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to accommodate implementation of a 21 month operating cycle with a 3 month outage, or a 24 month plant refueling cycle for those Technical Specification surveillances which will expire prior to the currently scheduled 13R refueling outage.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on June 5, 1990 (55 FR 22977). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated May 4, 1990, (2) Amendment No. 141 to License No. DPR-16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street N.W., Washington, D.C. and at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, New Jersey 08753. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects I/II.

Dated at Rockville, Maryland this 20th day of August 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W. Dromerick

Alexander W. Dromerick, Sr. Project Manager,
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

DATED: August 20, 1990

AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DRP-16

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