April 6, 1993

Docket No. 50-219

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Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M81848)

The Commission has issued the enclosed Amendment No. 163 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated October 4, 1991, as supplemented December 11 and 24, 1991, May 19, June 3 and 24, November 5, 1992, and March 25, 1993.

The amendment extends the expiration date of the facility operating license from December 15, 2004 to April 9, 2009. This extension provides an effective operating license term of 40 years from the beginning of plant operation rather than 40 years from the issuance of the construction permit.

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

Sincerely,

Original signed by

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. 163 to DPR-16
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

BC:EMEB*	OGC*
JNorberg	EHoller
3/1/93	3/8/93

*See previous concurrence

OFFICE	LA:PDI-4*	PM:PDI-4*	D:PDI-4*	BC:ESGB*	BC:SCSB*
NAME	SNorris 4/5/93	ADromerick:cn	JStolz	GBagchi	JStrosnider
DATE	2/17/93	2/17/93	3/4/93	2/17/93	3/1/93

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Document Name: G:\DROMERIC\M81848.AMD

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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 FACILITY OPERATING LICENSE

Amendment No. 163
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 October 4, 1991, as supplemented December 11 and 24, 1991, May 19, June 3 and 24, November 5, 1992, and March 25, 1993, 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, Facility Operating License No. DPR-16, Paragraph 3 is hereby amended to read as follows:*

   (3) This license is effective as of the date of issuance and shall expire at midnight on April 9, 2009.

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:
Page 6 of license

Date of Issuance: April 6, 1993

*Page 6 is attached, for convenience for the composite license to reflect this change.

F. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

3. This license is effective as of the date of issuance and shall expire at midnight on April 9, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed
by

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendices A and B -
Technical Specifications

Date of Issuance: July 2, 1991

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 163

TO FACILITY 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 October 4, 1988, as supplemented December 11 and 24, 1991, May 19, June 3 and 24, November 5, 1992, and March 25, 1993, GPU Nuclear Corporation (the licensee/GPUN) requested an amendment to the Facility Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station (OCNGS). The proposed amendment would extend the expiration date of the facility operating license from December 15, 2004 to April 9, 2009.

2.0 DISCUSSION

Title 10 CFR 50.51 specifies that each license will be issued for a fixed period of time not to exceed 40 years from the date of issuance. The currently licensed term for OCNGS is 40 years, commencing with the issuance of the construction permit on December 15, 1964. Therefore, the current expiration date is December 15, 2004. However, due to construction time, the effective facility operating license is about 35.7 years. Consistent with Section 50.51 of the Commission's regulations, the licensee has requested an extension of the facility operating license so that the facility operating license would end 40 years after the date of issuance (April 9, 2009) of Provisional Operating License No. DPR-16. The Provisional Operating License was superseded in its entirety by full-term Facility Operating License No. DPR-16 which was issued on July 2, 1991.

The licensee's request for extension of the facility operating license is based on the fact that a 40-year service life was considered during the design and construction of the plant. Although this does not mean that some components will not wear out during the plant lifetime, design features were incorporated which maximize the inspectability of structures, systems, and equipment. Surveillance and maintenance practices which are implemented in accordance with the ASME Code and the facility Technical Specifications provide assurance that any unexpected degradation in plant equipment will be identified and corrected.

3.0 EVALUATION

The staff has evaluated the safety issues associated with issuance of the proposed license amendment which would allow 4.3 additional years of plant operation. The issues addressed consist of additional radiation exposure to the licensee's operating staff, impacts on the offsite population, nonradiological impacts, and the general aging of plant structures and equipment. The impact of additional exposure to the operating staff, the impact on the offsite general population, and the nonradiological impacts are addressed in the NRC staff Environmental Assessment dated January 25, 1993.

3.1 Mechanical Equipment

The components of the reactor coolant pressure boundary were designed, built and tested in accordance with the ASME Boiler and Pressure Vessel Code Section I Power Boilers, 1962 Edition and the ASA B31.1-1955 Piping Code, plus the Nuclear Code Cases in effect December 1963 (when the vessel was purchased). Furthermore, the vessel manufacturer was directed by the purchase specification of the buyer to use Section VIII of the Code for Unfired Pressure vessels where Section I Power Boilers did not cover specific details.

The reactor coolant pressure boundary components within the Nuclear Steam Supply System (NSSS) scope were designed and constructed for a 40-year design life. The equipment design life is based on the time period of exposure to an operating environment. The 40-year design life is equivalent to 32 Effective Full Power Years (EFPYs). During the plant construction, materials were not exposed to the operating environment except for system functional tests. The system components were not subjected to a radiation environment until after the operating license became effective. GPUN is committed to a periodic inservice inspection program for the Reactor Coolant System per Technical Specification Section 4.3.

The OCNGS Inspection Program is based on the 1974 Edition through Summer 1975 Addenda of Section XI of the ASME Code, and included the 120-month inspection interval from December 8, 1979 through December 7, 1989. Oyster Creek has received an extension of 22 months (October 7, 1989) to this inspection interval. The program incorporates some provisions for pressure testing and ultrasonic examination which correspond to the requirements of the 1980 Edition of the ASME Code through Winter 1981 Addenda and the 1977 Edition through Summer 1978 Addenda, respectively. Relief has been granted by NRC in accordance with regulations. On April 16, 1992, the licensee submitted the OCNGS Inservice Inspection Program for the Third Ten Year Interval. The staff is in the process of reviewing the program.

In 1982, intergranular stress corrosion cracking (IGSCC) was first identified in the large diameter recirculation piping at OCNGS. Inspections of similar piping at other boiling water reactors (BWRs) disclosed additional instances of IGSCC in large diameter stainless steel pipes. Since the NRC staff considered this a generic problem, Generic Letter 84-11 was issued to require a reinspection program at all BWRs. This program involved welds in stainless steel pipes greater than 4 inches in diameter, in systems that are part of or connected to the reactor coolant pressure boundary. If IGSCC was discovered, repair, analysis, and additional surveillance were required to ensure the continued integrity of the affected pipe.

Generic Letter (GL) 88-01, issued on January 25, 1988, superseded GL 84-11, and included a copy of NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." NUREG-0313, Revision 2, described methods acceptable to the staff to control the susceptibility of BWR ASME Boiler and Pressure Vessel Code Class 1, 2, and 3 pressure boundary piping and safe ends to intergranular stress corrosion cracking. For piping that does not conform to the staff positions, varying degrees of inservice inspection are required to ensure structural integrity of the pressure boundary piping system, pursuant to 10 CFR 50.55a(g)(6)(ii). The licensee responded to GL 88-01 by letter dated August 12, 1988, as supplemented on January 31, 1989, June 7, 1990, October 18, 1990, June 11, 1991, February 26, 1992, October 2, 1992, and November 25, 1992. The NRC staff has reviewed these responses and forwarded its safety evaluation to the licensee by letters dated June 28, 1991, September 12, 1991, July 30, 1992, August 25, 1992, and January 14, 1993. The NRC staff concluded that the licensee's proposed IGSCC inspection and mitigation program will provide reasonable assurance of maintaining the long term structural integrity of austenitic stainless steel piping at OCNGS. License Amendment No. 154 which incorporates the requirements of GL 88-01, was issued on September 12, 1991.

Technical Specification 4.5.P - Suppression Chamber Surveillance requires periodic inspection of the coating on the interior surfaces of the torus to ensure its continued integrity. This inspection can be performed visually without difficulty. With respect to the drywell, there is no specific requirement for inservice inspection. The only requirement is the cursory visual examination of the drywell interior surface before integrated leak rate testing (ILRT). The concrete shield wall, located 3 inches from the drywell shell, prevents visual inspection of the drywell outside surface. Since 1986, after the discovery of extensive corrosion of the drywell in the sand cushion area, the licensee has instituted an inservice inspection program of the drywell shell through ultrasonic testing (UT) thickness measurements. The licensee has a program to remove the sand. The effects of sand removal on the structural behavior were analyzed by the licensee and reviewed by the staff. In a letter dated April 24, 1992, the staff accepted the licensee's analyses conclusions with the condition that the licensee continue to perform UT thickness measurements at outages of opportunity and at refueling outages so that the integrity of the drywell is maintained and assured. In a letter dated May 26, 1992, GPUN committed to continue taking UT drywell measurements at refueling outages and at other outages of opportunity for the life of the plant. The measurements will be at areas previously inspected and also at other accessible areas not previously inspected. In the letter, the licensee indicated that during the 14R refueling outage, GPUN would take UT thickness measurements in the drywell sandbed region, from the torus room side (outside the drywell), at shell locations not readily accessible from inside the drywell. They also indicated that assuming these measurements confirm that GPUN has bounded the corrosion problem with GPUN's current inspection locations, GPUN does not plan to make repeat measurements at these specific locations. During the 14R outage, GPUN took a sample of outside drywell foilation and had it analyzed at GPUN's Reading, Pennsylvania. laboratory. This analysis included an evaluation of the amount of base metal required to produce the observed foilation. GPUN was then able to extrapolate the remaining drywell wall thickness and this analysis agreed with the measured UT readings on the inside of the drywell. Drywell measurements will continue for life. They also indicated their current plan for OCNGS UT thickness measurement. In a letter dated June 30, 1992, the staff found that GPUN commitments regarding UT inspection of the OCNGS drywell containment are acceptable.

The NRC is currently reviewing the licensee's request to revise the drywell design pressure from 62 psig to 44 psig. As of this time, no decision has been reached on this matter. The current license requirement exists which requires the drywell containment to satisfy ASME Boiler and Pressure Vessel Code, Section VIII requirements for the licensed drywell design pressure (OCNGS Technical Specification 5.2.5). Failure to meet these Code requirements would require corrective actions by the licensee regardless of license expiration date.

From the investigation of corrosion in different areas of the drywell, the licensee indicated that corrosion in some of the areas may have occurred during the time of construction. The staff's position is that the problem may have originated from deficiencies in design and construction and is irrelevant to aging or the effects of plant operation. The drywell corrosion is a continuing concern irrespective of the extension of the duration of the operating license. The licensee has adopted a comprehensive program of maintaining the integrity of the drywell and the torus with both short term and long term measures. The staff has approved this program which contains elements of upgrading and repairing, if necessary, areas weakened by corrosion. If the licensee implements its containment integrity plan properly, the staff believes that containment integrity can be maintained well beyond the period of extension requested by the licensee.

The NRC requires licensees to develop and implement an inservice testing (IST) program for demonstrating the continued operability of OCNGS pumps and valves in response to the requirements of 10 CFR 50.55a. By letter dated October 11, 1991, the licensee submitted its third ten-year interval IST program plan for demonstrating the continued operability of the OCNGS pumps and valves. By letter and enclosed safety evaluation dated September 24, 1992, the NRC staff approved the licensee's IST program.

From the staff evaluation, the staff concludes that compliance with the Codes, standards, and regulatory requirements to which the mechanical equipment for OCNGS was originally analyzed, constructed, tested and inspected, including the inservice inspection programs in compliance with Section XI of the ASME Boiler and Pressure Vessel Code and the other augmented inspections of austenitic stainless steel piping, drywell and torus provide adequate assurance that the structural integrity of components important to safety will be maintained during the additional periods authorized by the proposed amendment. Additionally, the testing program for pumps and valves will enable early detection of degradation which could affect their operability. Any significant degradation by an active mechanism would be discovered and the mechanical equipment or component restored to an acceptable condition. Therefore, the age of the mechanical equipment or components should not be a concern in the proposed extension of the facility operating license.

3.2 Electrical Equipment and Environmental Qualification

The Environmental Qualification (EQ) program for electrical equipment operating in a harsh environment is described in the OCNGS Updated Final Safety Analysis Report (UFSAR), Section 3.11. The program ensures that EQ is maintained for required electrical equipment within the scope of 10 CFR 50.49.

Aging analyses have been performed for safety-related electrical equipment in accordance with 10 CFR 50.49, Environmental Qualification, to identify qualified lifetimes for this equipment. These lifetimes are incorporated into plant equipment maintenance and replacement practices to ensure that safety-related electrical equipment remains qualified and available to perform its safety function regardless of the overall age of the plant. Therefore, the electrical system design, electrical equipment selection and application, and the environmental qualification of electrical equipment has adequately considered or is not impacted by a 40-year operational lifetime.

The OCNGS program was evaluated by the staff and found acceptable in a safety evaluation dated May 28, 1985, and in Inspection Report 50-219/86-08, dated August 5, 1986.

3.3 Structures

The design and construction of structures was in accordance with various Codes and standards applicable at the time of plant construction. The design bases, fabrication, construction, and quality assurance criteria for the plant were reviewed and found acceptable by the NRC staff. Industrial experience with such structures confirms that a service life in excess of 40 years can be anticipated.

The UFSAR states that the reactor vessel was fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code Section I, Power Boilers, 1962 Edition and Addenda plus the Nuclear Code Cases applicable in December 1963.

The integrity and performance capability of the ferritic materials in the reactor vessel is assured because the fracture toughness is monitored with a surveillance program in conformance, to the extent practical, with the requirements of Appendix H, 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," and ASTM 185, "Standard Practice for

Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The vessel material and surveillance sample withdrawal schedule is specified in Technical Specification 4.3.A. The ferritic materials must meet the fracture toughness properties of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR Part 50, "Fracture Toughness Requirements."

A comprehensive vessel material surveillance test program is maintained in accordance with 10 CFR Part 50, Appendix H. The program provides the means for continuously monitoring the cumulative effects of the neutron exposure on the materials of the reactor vessel throughout the life of the plant. The analysis of the OCNGS plant specific surveillance capsules have confirmed that the predictions used in the analytical techniques for establishing operating limitations for the reactor vessel are conservative. Future OCNGS plant specific material surveillance capsules will be analyzed at specified times throughout plant life in order to ensure that the predictions used in the analytical techniques for establishing future operating limitations for the reactor vessel remain conservative.

Temperature and pressure changes during heatup, cooldown and normal operation of the reactor coolant system are limited to protect against non-ductile failure of the reactor coolant system. These limits are established in accordance with the requirements of 10 CFR Part 50, Appendix G, and calculated utilizing the procedures defined in Regulatory Guide 1.99, Revision 2. Amendment No. 151, dated April 11, 1991, revised the Technical Specifications Section 3.3.A, pressure/temperature limits of the reactor coolant system up to 17 effective full power years.

Paragraph IV.A.1 of 10 CFR Part 50, Appendix G states that reactor beltline materials must have Charpy upper-shelf energy of no less than 75 ft-lb (102J) initially and must maintain upper-shelf energy (USE) throughout the life of the vessel of no less than 50 ft-lb (68J) unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.

On May 28, 1992, GPUN met with the NRC staff to discuss reactor vessel USE for OCNGS. In the meeting, GPUN stated that a review of GL 92-01, "Reactor Vessel Integrity" indicates that three beltline plates at OCNGS currently have less than 50 ft-lb USE if the conservative 65% conversion factor given in Standard Review Plan 5.3.2 were used for converting Charpy V Notch longitudinal USE values to transverse USE values.

GPUN also stated during the meeting that based on preliminary analysis, ASME Section XI acceptance criteria for low USE are satisfied for the OCNGS reactor pressure vessel beyond the end of the licensed life.

In a letter dated September 22, 1992, GPUN attached report (GE-NE-52370-0692) which presents a fracture mechanics evaluation of the OCNGS reactor pressure vessel using methods and acceptance criteria provided in a draft Code Case

(Appendix XX) which has been approved by the ASME Section XI Subcommittee on Nuclear Inservice Inspection in August 1992. The results of GPUN's evaluation show that, even for the conservative case of 35 ft-lb USE (a projected USE at much beyond the end of the licensed life for the worst beltline plate), the acceptance criteria are satisfied for all four Service Level Loadings.

GPUN further concluded "that even though some beltline plates are currently below 50 ft-lb USE value, these lower values are acceptable. Thus, the Oyster Creek RPV will continue to meet the requirements of 10 CFR 50, Appendix G. This conclusion would also remain valid for any realistic plant life extension that may be considered by GPU Nuclear for OCNGS in the future."

The staff is in the process of reviewing the licensee's analysis. If the staff does not agree with GPUN's conclusion, the staff will require that GPUN take the appropriate corrective actions.

The NRC staff concludes that there are no special considerations regarding degradation of structures due to the proposed operating lifetime extension. The structural integrity of the reactor vessel is assumed because it was originally designed assuming 40-year lifetime; it is monitored, inspected and tested to detect degradation processes at an early stage of development; and it is operated in accordance with procedures to assure that its design conditions are not exceeded. As discussed above, if GPUN cannot show to the staff's satisfaction that OCNGS meets Paragraph IV.A.1 of 10 CFR Part 50, Appendix G, the NRC staff will require that GPUN take the appropriate corrective actions.

4.0 SPENT FUEL STORAGE

For the period from 2004 to 2009, OCNGS will rely on a combination of wet and dry storage technology for its spent nuclear fuel. Without the addition of incremental dry storage capacity at OCNGS, full core discharge capacity to the spent fuel pool will be available through the 1996 refueling outage and the plant will exhaust all spent fuel pool storage capacity in 2000. The OCNGS spent fuel pool has been reracked with high density poison racks and no further rack capacity expansion is possible. GPU Nuclear recognizes the eventuality and is currently soliciting proposals to augment the existing pool capacity with dry storage technology to provide sufficient onsite spent fuel storage beginning in 1996 to maintain the plant's full core reserve margin and to provide storage for all spent fuel projected to be discharged by OCNGS through 2009. Incremental modular dry storage capacity would be added each operating cycle to support continuing plant operation. It is projected that some 1250 spent fuel assemblies will be in dry storage at OCNGS by 2009.

OCNGS is implementing 24-month operating cycles beginning with Cycle 13. As a consequence, the extended operating period represented by the proposed license amendment is expected to result in an insignificant increase in the total number of spent fuel generated when compared to the total quantity of spent fuel that would have been generated at the current license expiration date assuming previously utilized 18-month cycle lengths.

5.0 STATE OF NEW JERSEY - DEPARTMENT OF ENVIRONMENTAL PROTECTION AND ENERGY BUREAU OF NUCLEAR ENGINEERING COMMENTS REGARDING THE PROPOSED AMENDMENT REGARDING AN OPERATING LICENSE EXTENSION FROM DECEMBER 15, 2004 TO APRIL 9, 2009

By letter dated November 27, 1991, the State of New Jersey Department of Environmental Protection and Energy's Bureau of Nuclear Engineering (BNE) advised the NRC that they reviewed GPUN's Amendment Request No. 199 to extend the duration of the OCNGS Operating License to forty (40) years from the date of the full power license. BNE also indicated that as a result of their review, the BNE recommends that the change request be approved contingent on resolution of the following issues:

1. Drywell thinning due to corrosion;

2. Hydrogen mitigation requirements of 10 CFR 50.44;

3. Structural integrity of the cracked Core Spray System Sparger; and

4. Assessment of the frequency of hazardous material shipping on U.S. Route 9 and potentially the level of risk associated with the shipment.

5.1 Drywell Thinning Due To Corrosion

In BNE's letter of November 27, 1992, BNE stated that:

> ... one of the major concerns is the drywell thinning due to corrosion, which is an ongoing safety assessment activity for OCNGS. Based on the current corrosion rate for the sandbed region, a critical region, the vessel would attain its minimum required thickness by 1995. Although, GPUN is planning to remove the sand from this region, there is no assurance that the corrosion will be eliminated. In other regions, such as the upper spherical region, one current thinnest reading is approaching the minimum required thickness. Since the drywell structural integrity is based on a design pressure of 62 psig, GPUN has recently submitted an amendment request, revising the peak design pressure from 62 psig to 44 psig. This design criteria revision, if approved by the NRC, will be unique to the nuclear industry. Since the drywell, and its integrity, is essential to the safe operation of the plant, the drywell corrosion mitigation program must be a license requirement.

As discussed in Section 3.1 of this evaluation, the staff indicates that since 1986, after the discovery of extensive corrosion of the drywell in the sand cushion area, the licensee has instituted an inservice inspection program of the drywell shell through UT thickness measurements. In a letter dated May 26, 1992, GPUN committed to continue taking drywell measurements at refueling outages and at other outages of opportunity for the life of the plant. In a letter dated March 25, 1993, GPUN further committed to notifying the staff before changing the drywell thickness measurement program. The staff also indicated that the licensee has adopted a comprehensive program of maintaining the integrity of the drywell with both short term and long term measures. The staff has approved this program which contains elements of upgrading and repairing, if necessary, areas weakened by corrosion. As discussed in Section 3.1 of this report, the staff is in the process of reviewing the licensee's request to revise the containment peak design pressure from 62 psig to 44 psig. Based on the commitments by GPUN, the staff has determined that a condition to the license regarding the drywell corrosion mitigation program is not required.

5.2 Hydrogen Mitigation Requirements of 10 CFR 50.44

In BNE's letter of November 27, 1992, BNE states:

> Another significant safety issue for OCNGS is the hydrogen mitigation requirements of 10 CFR 50.44. This issue, open since the early 1980's, arose from the accident at the Three Mile Island Unit 2. The Commission on December 24, 1980, in a special report to the Congress, identified "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment" as an Unresolved Safety Issue (USI A-48). Recently, the resolution of USI A-48 was published in the Federal Register dated February 15, 1990. It stated that "Ignition of hydrogen after the production of additional combustible gases from radiolysis is to be mitigated by recombiners, purge repressurization system or combination." However, GPUN is still treating this issue as plant specific. BNE believes that since NRC's resolution was published in the Federal Register, no exception should be allowed for any plant.

By letter dated November 6, 1990, the staff issued a position to GPUN concerning compliance with 10 CFR 50.44 combustible gas control requirements.

GPUN responded by proposing to upgrade their nitrogen inerting system (NIS). Other modifications were also proposed to further enhance NIS capability. These include installation of a hardened vent, an alternate portable nitrogen supply, and alternate power source capability and installation of a bypass switch in the control room to permit valve operation when a containment isolation signal is present.

Based on the above, the staff, in a safety evaluation dated November 18, 1992, found the proposed changes acceptable and considers this issue resolved.

5.3 Structural Integrity of the Cracked Core Spray System Sparger

In BNE's letter of November 27, 1992, BNE stated that:

> OCNGS is one of the oldest operating nuclear plants in the United States. Although it meets the minimum Emergency Core Cooling System (ECCS) design criteria specified in 10 CFR 50.46, it does not have modern design features such as High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems. Its Core Spray System sparger, an ECCS system, has a crack in the piping and operates with a clamp installed on the sparger. In 1988, BNE expressed its concerns on this issue and recommended that the structural integrity of this sparger, under severe accident conditions, should be analyzed.

The Core Spray System (CSS) is one of the Emergency Core Cooling Systems (ECCS). It provides emergency core cooling flow at low reactor pressure conditions via two system loops, each consisting of a core spray sparger, two main pumps and two booster pumps. During a scheduled 1978 inservice inspection, GPUN discovered a through-wall crack in the System No. 2 sparger. After performing structural and hydraulic analyses, GPUN concluded that the cracked sparger, in its as-found condition, was adequate for continued operation, but nevertheless, opted to install an additional mechanical support and a repair bracket to prevent further growth of the crack. By letter dated March 31, 1980, the licensee submitted Amendment No. 47 that proposed continuous operation with a cracked sparger along with the additional support and repair bracket. The NRC staff reviewed the request and determined that the construction of the bracket assembly was in accordance with accepted engineering practices, and that the analysis of the structural loads adequately demonstrated the repair bracket's capability to limit the confirmed crack to within an acceptable range. Therefore, the staff granted the licensee's request in a letter dated May 29, 1980.

By letter dated May 21, 1982, as supplemented November 5, 1982, May 13, 1983, and October 11, 1983, the licensee proposed License Amendment No. 70 for the sparger condition. The requested amendment would revise paragraph 2.C(7) of the license to permit the licensee to inspect all accessible surfaces and welds of both core spray spargers and to repair the assemblies by a method acceptable to NRC, in lieu of replacing the sparger. The licensee's review revealed no evidence of significant crack progression, and concluded that any minor cracking which escaped detection was insignificant in terms of both structural integrity or flow distribution. Based on these findings, the staff granted the licensee's amendment request to conduct an augmented inspection program regularly in lieu of replacing the spargers, in a letter dated January 26, 1984.

In a letter dated March 30, 1988, as supplemented April 12, 1988, and September 22, 1988, License Amendment No. 129 was proposed. By performing transient analyses (GPU-TR-033 and GPU-TR-040) and design basis loss of coolant accident (LOCA) analyses (NEDC-31462P) according to NRC accepted methodologies, OCNGS proposed TS revisions to change the core limits delineated in TS Section 3.10, for limits consistent with a new fuel combination for its Cycle 12 operation. A 70% rated ECCS flow and 102% power level were assumed in the analyses. Conformance to the ECCS acceptance criteria was shown. The analysis assumed a coincident flow of 3400 gpm (1 main pump and 1 booster pump) from one CSS sparger and 2200 gpm (1 main pump) from the other CSS sparger. Because the System No. 2 sparger could have a potential flow loss through the crack, higher flow requirements of 3640 and 2360 gpm were imposed on System No. 2 to meet the 10 CFR 50.46 acceptance criteria. The proposed amendment was found acceptable by the staff, as documented in a October 31, 1988, letter.

In a submittal dated August 14, 1990, as revised on June 18, 1991, GPUN proposed to revise TS Sections 3.4.A.3, 3.4.A.4, and 3.4.D.2 regarding new limiting conditions for operation and surveillance requirements for the CSS. These were revised conservatively to assure the operability of the CSS for a postulated design basis LOCA. Prior to NRC approval of this request in Amendment No. 153 on September 5, 1991, GPUN had voluntarily revised Plant Procedures to implement the more restrictive operating conditions mandated by the LOCA analysis.

Since GPUN continues to perform augmented inspections and regular testing of the core spray spargers at regular intervals to meet a set of conservative acceptance criteria, and that further degradation would have to be identified, evaluated and dispositioned, the likelihood of continuous undetected degradation of the sparger is remote. The staff has previously reviewed GPUN's inservice surveillance program to maintain the sparger integrity, and concluded that it was adequate and acceptable. An Operating License Condition also requires that the NRC staff review the inspection and testing methods and results, and authorize restart from each refueling outage. As of the 13R outage (1991), there were no new indications and the repair bracket assembly was in good condition.

During the 14R refueling outage (November 1992 through February 1993), inspection and testing of the downcomer to core spray sparger system No. 1, the licensee found a small leak in one of the transition welds of the downcomer. At a meeting with the staff on January 6, 1993, the licensee indicated that during an air test of core spray sparger system No. 1, the licensee detected a steady stream of air bubbles from the system No. 1 transition pipe coupling. A visual examination of the coupling indicated that there was a 1/8" diameter blow hole in the fillet weld. They also stated that there was no sign of fatigue cracking or intergranular stress corrosion cracking (IGSCC) and that there was no other indication detected in the remainder of the weld.

The licensee also determined that the 1/8" diameter blow hole defect in the 1/4" fillet weld is not a structural problem and that the blow hole leakage is 1 to 3 gpm via the annulus. Even with this leakage there will still be a large flow margin above technical specification flow requirements.

Based on the above, the licensee concluded that in GPUN's best judgement the leak is due to a weld defect due to 20+ years of heatup and cooldown. They also concluded that a clamp over the joint is not needed and that they prefer not to install the clamp because of problems that could arise during installation. They indicated, however, that clamp, tooling design and fabrication is ongoing. The licensee provided a safety evaluation for staff review justifying their actions. The staff reviewed the licensee's response and in a safety evaluation dated February 5, 1993, found that actions taken by the licensee were acceptable.

Based on the above, we conclude that the system is capable of performing its safety function of mitigating a designed basis LOCA during an operating term of 40 years.

Analysis for severe accident conditions is not part of the design basis for many plants, including OCNGS. In its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" published August 8, 1985 (50 Fed Reg 32133), the Commission concluded that existing plants posed no undue risk to public health and safety and, because of the low severe accident risk, did not see a need for immediate action on generic rulemaking for existing plants. The Commission stated its intentions for rulemakings and other regulatory actions for resolving severe accident safety issues. Among other things, the Commission has initiated a program for Individual Plant Examination (IPE) for severe accident vulnerabilities, as addressed by Generic Letter 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities (IPE)" (November 23, 1988). The staff requested that GPUN conduct an IPE to assess OCNGS' vulnerability to a severe accident. In a letter dated August 14, 1992, GPUN provided the OCNGS IPE which the staff is reviewing to assure that GPUN met the intent of Generic Letter 88-20.

Neither current NRC regulations nor the IPE program requires that a licensee perform analyses under "severe accident conditions" for a plant-specific equipment issue such as the OCNGS CSS sparger. In fact, lacking specific information regarding CSS sparger failure rates, a probabilistic risk analyses (such as used in the IPE for various systems) would be too uncertain to be used by the NRC for regulatory purposes. As discussed above, the CSS can mitigate a design basis LOCA which is the design basis for OCNGS and the NRC staff has accepted the as-modified CSS sparger as meeting current regulatory requirements.

5.4 <u>Assessment of the Frequency of Hazardous-Material Shipping on U.S. Route 9 and Potentially the Level of Risk Associated with the Shipment</u>

In BNE's letter of November 27, 1992, BNE stated that:

> ... in the Full Term Operating License (FTOL) Safety Evaluation Report (SER), issued in January, 1991 by the NRC, it was stated that there is sufficient truck traffic on U.S. Route 9 to require an assessment of the frequency of hazardous-material shipping, and potentially, the level of risk associated with the shipment. By letter dated May 23, 1990, the NRC staff requested that the GPUN address within one year of the issuance of the SER the transportation issue in order to verify that the risk due to nearby truck transportation along Route 9 is acceptably low. By letter dated August 9, 1990, GPUN committed to perform an assessment of transportation in the vicinity of Oyster Creek and submit it as requested. To date, this issue has not been resolved.

In January 1991, the staff issued a Safety Evaluation Report (SER) related to the full-term operating license for Oyster Creek Nuclear Generating Station. One of the issues identified by the staff in Section 2.2 of the SER (NUREG-1382) refers to potential shipments of hazardous materials near the plant. Specifically, the staff had found that current estimates of truck traffic on the nearby U.S. Route 9 were significantly greater than what was estimated previously, at the time of the application for a Provisional Operating License. The staff noted in the SER that the current truck traffic rate on Route 9 had the potential for exceeding the screening criterion given in Regulatory Guide 1.78. In view of the above, the staff had obtained from the licensee a commitment to assess the transportation on the route and submit the results to the staff.

In May 1992, the licensee met with the staff to describe their assessment of the issue and the means for addressing the potential hazards. As indicated by the licensee, specific shipping data for hazardous materials are not maintained by State or Federal transportation authorities and, hence, they are not readily available.

The staff indicated that in absence of supporting data the potential for frequent shipment of hazardous materials cannot be dismissed. As an alternative, the staff indicated that a reasonable safeguard against traffic accidents involving release of hazardous materials is to provide early warning to the plant. Specifically, early notification to the control room operators would provide sufficient time to implement emergency procedures relating to the potential effects of the hazardous material (e.g., safeguards against airborne toxic or flammable materials).

In response, the licensee requested and obtained from the New Jersey State Police a commitment (New Jersey State Police interoffice memo, dated April 13, 1992) regarding hazardous material accident notification. Specifically, current New Jersey State Police policy requires their duty officer or hazardous material emergency response personnel to notify promptly the Oyster Creek Group Shift Supervisor or the Control Room in the event of a transportation accident involving the release of airborne hazardous material occurring in Lacey or Ocean Townships.

In view of the above, and the licensee's Procedure 2000-ABN-3200.33, "Toxic Material/Flammable Gas Release - No Radiation Involved," the staff finds that the risk due to hazardous truck shipments near the OCNGS is acceptably low and the staff considers this issue resolved.

6.0 SUMMARY

The NRC staff concluded in the Environmental Assessment dated January 25, 1993, that the annual radiological effects during the additional years of operation that would be authorized by the proposed license amendment are not more than were previously estimated in the Final Environmental Statement, and are acceptable. The NRC staff concludes from its considerations of the design, operation, testing, and monitoring of the mechanical equipment, electrical equipment, structures, and the reactor vessel that an extension of the facility operating license for OCNGS to a 40-year service life is consistent with the UFSAR, SER, and submittals made by the licensee, and that there is reasonable assurance that the plant will be able to continue to operate safely for the additional period authorized by the proposed amendment. The staff also concludes that the plant is operated in compliance with the Commission's regulations, and that issues associated with plant degradation have been adequately addressed.

Based on the above, the NRC staff concludes that extension of the facility operating license for OCNGS to allow a 40-year service life from issuance of the initial operating license is consistent with the Final Environmental Statement and Safety Evaluation Report for the plant and that the Commission's previous findings are not changed.

7.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of New Jersey, Department of Environmental Protection and Energy's Bureau of Nuclear Engineering (BNE) was notified of the proposed issuance of the amendment. In a letter dated November 27, 1992, BNE provided their position regarding the issuance of the proposed amendment. For the staff response to BNE's letter, see Section 5.0 of this evaluation.

8.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, a notice of environmental assessment and finding of no significant impact was published in the Federal Register on February 2, 1993, (58 FR 6814). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

9.0 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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Date: April 6, 1993

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 163 to Facility Operating License No. DPR-16 issued to GPU Nuclear Corporation (the licensee), 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 extends the expiration date of the facility operating license from December 15, 2004 to April 9, 2009. This extension provides an effective operating license term of 40 years from the beginning of plant operation rather than 40 years from the issuance of the construction permit.

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 November 1, 1991 (56 FR 56251). 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 (58 FR 6814).

For further details with respect to the action see (1) the application for amendment dated dated October 4, 1991, as supplemented December 11 and 24, 1991, May 19, June 3 and 24, and November 5, 1992, and March 25, 1993, (2) Amendment No. 163 to License No. DPR-16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment dated January 25, 1993. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, New Jersey 08753.

FOR THE NUCLEAR REGULATORY COMMISSION

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Dated at Rockville, Maryland
this 6th day of April 1993