February 21, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M89684)

Dear Mr. Barton:

The Commission has issued the enclosed Amendment No. 177 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated June 15, 1994, as supplemented September 23, 1994, and November 3, 1994.

The amendment revises Technical Specification 2.3.D to change the setpoints "Reactor High Pressure, Relief Valve Initiation" by increasing the setpoint value by 15 psig for each of the Electromatic Relief Valves in the Automatic Depressurization System.

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

Docket No. 50-219

Enclosures: 1. Amendment No. 177 to DPR-16
2. Safety Evaluation
3. Notice

cc w/encls: See next page

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

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Mr. John J. Barton
Vice President and Director

Oyster Creek Nuclear
Generating Station

cc:

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

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. 177
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 June 15, 1994, as supplemented September 23, 1994, and November 3, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

    (2) Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177, 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, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronald W. Hernan for

Phillip F. McKee, 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: February 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 177

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

Remove	Insert
2.3-2	2.3-2

FUNCTION		LIMITING SAFETY SYSTEM SETTINGS
B.	Neutron Flux, Control Rod Block	The Rod Block setting shall be $S \leq [(0.90 \times 10^{-6}) W + 53.1] \left[\frac{FRP}{MFLPD}\right]$ with a maximum setpoint of 108% for core flow equal to 61×10^{6} lb/hr and greater. The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.
C.	Reactor High, Pressure, Scram	$\leq$1060 psig
D.	Reactor High Pressure, Relief Valves Initiation	2 @ $\leq$ 1085 psig 3 @ $\leq$ 1105 psig
E.	Reactor High Pressure, Isolation Condenser Initiation	$\leq$1060 psig with time delay $\leq$3 seconds
F.	Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig ±12 psi 5 @ 1221 psig ±12 psi
G.	Low Pressure Main Steam MSIV Closure	$\geq$825 psig (initiated in IRM Line, range 10)
H.	Main Steam Line Isolation Valve Closure, Scram	$\leq$10% Valve Closure from full open
I.	Reactor Low Water Level, Scram	$\geq$11'5" above the top of the active fuel as indicated under normal operating conditions
J.	Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	$\geq$7'2" above the top of the active fuel as indicated under normal operating conditions

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 177

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 June 15, 1994, as supplemented September 23, 1994, and November 23, 1994, GPU Nuclear Corporation (GPUN), the licensee for Oyster Creek Nuclear Generating Station (OCNGS) requested a change to Technical Specification (TS) 2.3.D. This change request involved raising the relieving setpoints on the Electromatic Relief Valves (EMRVs) by 15 psig. Two valves will now have a setpoint of 1085 psig, and three will have a setpoint of 1105 psig. This change in setpoint pressure is needed to accommodate setpoint drift that occurs during the cycle that may cause the plant to exceed its TS setpoints during the cycle. This change only affects the relief valves and not the spring safety valves.

2.0 BACKGROUND

The Oyster Creek facility has five EMRVs connected to its main steam lines. The discharge lines from three EMRVs are combined into a south EMRV discharge header. The discharge lines from the remaining two EMRVs are combined into a north EMRV discharge header. Each header leads to a quencher submerged in the suppression pool. The EMRVs are provided for the purpose of relieving primary system steam to the suppression pool in the event of an overpressure transient in the main steam system or in the event of an emergency that requires reactor vessel depressurization (Automatic Depressurization System (ADS) blowdown). In the event of an overpressure transient, an EMRV will automatically open at its high pressure actuation setpoint under the setpoint control of a Bourdon-tube type pressure switch. This is a somewhat different control scheme than that used with the Target Rock pilot-operated safety relief valves (SRVs) at most Mark I facilities. However, the purpose of the Oyster Creek EMRVs is similar to that of SRVs.

Nine "safety" valves are provided for ASME Code overpressure protection. Those valves are set at higher pressures than the EMRVs. However, they do not discharge to the suppression pool and thus do not create hydrodynamic loads on the containment. The safety valves would not be affected by the proposed amendment.

The dynamic loads that result from EMRV discharges are described in NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program." NUREG-0661 was issued July 1980 and promulgated to licensees by Generic Letter 80-78. EMRV dynamic loads include water clearing loads associated with the expulsion of the submerged water in the discharge line, and air clearing loads associated with the expulsion of compressed air (actually nitrogen in inerted containments) from the discharge line into the suppression pool. These loads were not originally considered to be significant, and were thus not considered in the original design of Mark I pressure suppression containments. It was later discovered that such loads can be significant. Under the guidance of GL 80-78 and NUREG-0661 (Mark I Containment Long-Term Program), licensees analyzed the EMRV discharge loads and accounted for them in their revised containment analyses which were submitted to the staff in Plant Unique Analysis Reports (PUARs). The generic load definition and modeling methodology used by licensees was defined in a Load Definition Report (LDR), NEDO-21888, and in Application Guides developed by General Electric for the BWR Owners Group.

3.0 EVALUATION

3.1 Analysis of Effect on Dynamic Loads for Higher High Pressure Actuation Setpoint

Whereas; in most Mark I facilities, each SRV has an independent discharge line to the suppression pool, the Oyster Creek facility utilizes a unique discharge piping design in which multiple EMRV discharge lines converge into headers. Also, two Y-quenchers, one on each header, are used instead of one T-quencher on each discharge line as used by most other Mark I facilities. The original Oyster Creek PUAR load definitions were based on calculations performed in accordance with methods prescribed in the GE Load Definition Report and associated "NEDO" documents using GE computer models (such as RVFOR, RVRIZ, TEEQFOR and QBUBS), and the results of August 1977 inplant EMRV steam discharge tests. The inplant tests were conducted with the torus shell and supports instrumented and the reactor vessel at a pressure of 1035 psia. The inplant tests were necessary because the generic T-quencher load definitions from the Monticello tests were not directly applicable to the Oyster Creek header arrangement with Y-quenchers. The staff previously evaluated and approved the alternative methodology. The staff's Safety Evaluation Report, *Oyster Creek Long-Term Program - Pool Dynamic Loads* was issued on January 13, 1984. The EMRV setpoints used in the inplant test were lower than the new setpoints the licensee has requested and thus do not bound the proposed modification.

In 1983, additional calculations of EMRV discharge loads were performed for each Mark I facility during development of the plant-specific Emergency Operating Procedures (EOPs). The EOPs require calculations of "EMRV Tailpipe Level Limits." These calculations determine the maximum water level allowable in the EMRV discharge piping such that, should an EMRV be opened, the piping is not likely to fail. Such calculations are done for a range of reactor vessel pressure conditions. As part of these calculations, an SRV tailpipe system load variation with reactor pressure vessel (RPV) pressure function (%/PSI) is derived (Ref.: OEI Document 8390-4C, Appendix C to NEDO-31331).

The licensee's June 15, 1994 amendment application, and subsequent letters dated September 23, 1994, and November 3, 1994, provide information to demonstrate that it is a conservative assumption that a 1.033 multiplier can be applied to the PUAR loads to account for higher pressure actuation. The objective of using a multiplier applied to the previous loads is to eliminate the need for complete reanalysis of loads and new inplant tests.

3.1.1 EMRV Discharge Loads on the EMRV Discharge Piping

Following an EMRV actuation, the pressure in the discharge line undergoes a transient prior to reaching a post-clearing steady-state value. A transient pressure wave travels back and forth in the line as the pressure continues to increase until the inertia of the water leg is overcome. The pressure differential across the wave front and fluid momentum changes create varying thrust loads on the various piping segments. These loads were originally calculated for the Oyster Creek PUAR in 1982 using the approved load definition report (LDR) methodology (which uses RVFOR) with adjustments to account for the fact that multiple EMRVs discharge into a common header. The fact that the Oyster Creek quenchers have a "Y" configuration was and is not considered significant for these specific loads. The LDR calculations produced a set of load-time histories for each segment of discharge piping. The inplant test data indicated that the calculations were conservative.

To account for the effects of increased EMRV setpoint pressure, the licensee now proposes to apply the 1.033 multiplier to these loads. The magnitude of this multiplier is based on the assertion that a linear variation of dynamic load vs. pressure is conservative. The licensee confirmed the validity of this assertion by examination of the 1983 EOP calculations. Those calculations encompassed both high and low reactor pressure conditions (i.e., 1133 psig, 412 psig and 206 psig) and thus provided the necessary sensitivity information to confirm the assertion. The assertion was further confirmed by a simple "first principles" analysis. In this analysis, a set of two differential equations, (1) the Newtonian equation-of-motion for a segment of water in the EMRV line water leg, and (2) a mass conservation equation for the steam/air space were solved simultaneously to relate setpoint pressure with line pressure. The resulting relationship (MPR Calc 83-187-02),

$$P_{(DISCH\ LINE)} = C\ P^{0.79}_{(SETPOINT)}$$

where C is a constant, has a less-than-1.0 exponent. The fractional exponent indicates that if the EMRV setpoint is raised, the pressure in the EMRV discharge line does not increase proportionally, but to a lesser degree. This relationship further indicates that the 1.033 multiplier associated with a linear variation is conservative. Based on the above, the staff finds that the use of a 1.033 multiplier applied to the PUAR loads is acceptable for defining the increased loads on the EMRV piping.

It is also noted that NEDO-31331, Supplement A, which prescribes methodology for calculating "SRV Tailpipe Level Limits" for Emergency Operating Procedures states that SRV tail pipe system loads decrease by 10% for each 100 psig reduction in RPV pressure. Also, GE was contacted and asked if any sensitivity studies had been performed for the power uprate program that might relate SRV line pressure changes to dynamic loads. The reviewer was informed that such studies have been conducted and that tests have been conducted. It was found that loads vary linearly with flowrate (Ref.: Telecon W. Long of NRC with Dan Pappone of GE of 10/27/94). This is consistent with the 1.033 multiplier.

3.1.2 Thrust Loads on the Y-Quencher

During water slug clearing, transient axial and perpendicular thrust loads are imposed on the Y-quencher arms due to acceleration forces imposed on the water slug. In the long-term program (LTP) analyses, results obtained from the clearing model (RVFOR) were used as input to the "BDIF" code (a licensee contractor's code) to calculate these thrust loads. BDIF calculations were also performed for the EOP "SRV Tailpipe Level Limit" analyses. The quencher thrust loads were also found to be bounded by the 1.033 multiplier. The use of the 1.033 factor is therefore acceptable for these loads also.

3.1.3 Water Jet Loads on Internal Structures

Water jets emanating from the quencher and having sufficient penetration distance, will create drag forces on submerged structures. These drag forces are proportional to the drag coefficient of the structure and the square of the jet velocity. It was found from jet length penetration studies that the jets would penetrate a distance of 7-8 feet and that the vent header support columns would thus experience such loads. To address the effects of increased actuation pressure the licensee examined the results of BDIF code output developed during the PUAR and EOP analyses. The information indicates that the square of the jet velocity, in the range of pressure concerned, is a linear function of the EMRV setpoint and, as a result, the magnitude of the support column drag loads can be considered to vary directly with EMRV setpoint pressure. On this basis, the 1.033 multiplier is appropriate for the jet impingement loads.

3.1.4 Bubble Loads on the Torus Shell

Following clearing of the water leg in the discharge line, the compressed air is accelerated into the suppression pool and forms a high pressure air bubble. This bubble expands and contracts a number of times before it rises to the pool surface. The associated fluid movement creates oscillatory drag loads on submerged structures as well as pressure loads on the torus shell and torus supports. These loads are referred to as EMRV air-clearing loads. For

Oyster Creek, the torus shell loads for the PUAR were calculated using the QBUBS code calibrated with information from the inplant tests.

The factors that affect shell pressure loads include bubble pressure, quencher location, pool geometry, and pool temperature. Only the bubble pressure is affected by the EMRV actuation pressure. The QBUBS output for the EOP calculations indicated that bubble pressure (and thus torus loads) varies directly and linearly with pressure. The use of the 1.033 multiplier is thus conservative, assuming dynamic amplification due to frequency shift is neglected. The EOP calculations indicated that very little frequency shift results due to EMRV discharge pressure changes in the range of concern. Frequency shift is thus not significant. The use of a 1.033 multiplier for these loads is acceptable.

3.1.5 Air Bubble Loads on Submerged Structures

As stated above, the velocity and acceleration fields produced by oscillating air bubbles rising to the pool surface also induce drag loads on internal submerged structures. The GE code TEEQFOR was used to calculate the bubble-induced drag loads on submerged structures for the LTP. These loads vary with bubble pressure according to the relationship

$$\text{Drag Load} \approx [\text{Bubble Pressure}]^{BFAC}$$

where "BFAC" is an empirical bubble charging factor determined by the inplant EMRV discharge tests.

The 1.033 multiplier is used for the bubble drag loads also. This has been determined to be appropriate for bubble drag loads because BFAC is 0.6 (considerably less than one), and because bubble pressure is known from QBUBS to vary directly with setpoint pressure. This provides the basis for acceptability of the 1.033 multiplier for these loads.

3.1.6 EMRV Steam Flow Rate Correction

The licensee proposes to increase the 1.033 multiplier to 1.04 to correct for an error discovered in the original PUAR analyses. The PUAR analyses assumed an EMRV design flow rating of 600,000 lbm/hr at 1250 psig pressure. The value should have been 602,900 lbm/hr (0.5% higher).

3.1.7 Conclusion

The licensee has provided information supporting use of a 1.04 multiplier. This multiplier is applied to pool dynamic loads previously calculated for the PUAR, to account for the proposed EMRV setpoint increase and to account for errors in the calculations of the PUAR loads due to use of an incorrect EMRV flow rating. The staff has reviewed the licensee's basis for use of the multiplier and finds it acceptable.

It is further noted that the magnitude of the proposed setpoint increase is within the range typically associated with power uprate amendments for Mark I facilities. In power uprate analyses, it has been found that the increased pool dynamic loads associated with SRV setpoint increases, necessary to accommodate higher vessel pressures, are readily accommodated by existing margins.

3.2 OVERPRESSURE ANALYSIS AND PRESSURIZATION TRANSIENTS

The overpressure analysis is not changed by this request because the relief mode of the SRVs, as well as the isolation condenser, are not credited in the analysis. The overpressure protection limit of 1375 psia is provided by the spring safety valves only. The operation limit minimum critical power ratio (OLMCPR) and the peak transient pressure are the only limits that will be affected by this change, and must be evaluated to assure that the limits will not be exceeded for transients and accidents.

The licensee reanalyzed the limiting pressurization transients with the new increased relief setpoints to provide assurance that no safety limits will be violated as a result of this setpoint increase. The limiting transient is the turbine trip without bypass (TTWOBP) and this continues to be the limiting transient with the change. The results show that the maximum reactor coolant boundary pressure is 1290 psia. The delta- CPR remains at 0.314 for the transient because it is reached prior to the relief valve setpoint. The feedwater controller failure transient results in a peak pressure of 1178 psia and the delta-CPR is 0.232 and occurs prior to EMRV opening. These remain the bounding transients for OCNGS, and the results are acceptable since no safety limits are violated.

The anticipated transients without scram (ATWS) were not evaluated for this change because previous analysis are still valid. The main steamline isolation valve (MSIV) closure transient for safety valve sizing bounds the ATWS event. That analysis demonstrates that the pressurization limits are not violated for pressurization transients. The MSIV event with no recirculation pump trip (RPT), no credit for EMRV actuation, and an indirect high flux scram is the bounding analysis for demonstrating that the reactor has adequate pressure relieving capacity with just the safety valves. This analysis is still valid, even with the EMRV setpoint increase, because no credit is taken with RPT and EMRV actuation is less severe than the bounding pressurization transient. The licensee had demonstrated this in a previous amendment. This amendment change request does not change the previous conclusions for the ATWS analysis. The staff finds the TS change request acceptable with respect to safety system settups and analysis.

3.3 TORUS - STRUCTURAL ANALYSIS

3.3.1 Torus Effects of Increased Loads on Torus

As resolution of the Generic Technical Activity A-7 (NUREG-0661, Ref. 2), the licensee's consultant; MPR Associates, Inc. (MPR), performed detailed stress analyses of components of the OCNGS containment affected by the actuation of EMRVs in MPR Report 733 (Ref. 3). The staff had accepted the analyses results in the staff's safety evaluation reports (Refs. 4,5).

The proposed setpoint pressures increase the existing setpoint pressures by about 3.3 percent. The licensee has used a multiplier of 1.04 to calculate the new pool dynamics loads. The NRC staff evaluated the increases in loads on the components of the OCNGS containment resulting from the proposed EMRV setpoint increases (Ref. 6) and concluded that the licensee proposed multiplier is acceptable for calculating the increased loads on the containment torus and its support structures since the responses of the torus and its support structures are linear.

In MPR Report 1434 (Ref. 6), the MPR has recalculated the stresses calculated in Ref. 3, and compared them against the ASME allowables used in the existing analyses. Two relevant outliers are identified where the recalculated stresses were found to be slightly above the allowables; (1) at torus shell between the straps, and (2) in torus support columns and in sway braces. MPR resolved the outliers by performing specific (instead of generalized) analyses compatible with the increased loads and demonstrated that the stresses in the components will not exceed the allowables. The staff finds the resolution of the outliers acceptable.

3.3.2 Effects of Torus Corrosion:

In Topical Report 101 (Rev. 0), attached to Ref. 1 and in response to the staff's request for additional information (Ref. 7), the licensee discussed the effects of corrosion related metal loss. In the discussion, the licensee stated that the torus corrosion found during 1983-84 outage was repaired by weld overlay, or shell stresses with the reduced shell thickness (due to corrosion) were shown to comply with the allowable stresses. During the same outage, the interior of the torus shell and the interior piping were coated with epoxy coating to protect them from additional corrosion. Since then, the licensee has performed four periodic inspections to assure coating integrity and absence of additional corrosion. Based on these inspections, the licensee concludes that the torus shell thickness is virtually unchanged since the repair and coating efforts performed in 1983-84 outage.

Table 1 attached to Ref. 7 indicated the maximum unrepaired corrosion depth in the interior surface of the torus varies between 35 and 50 mils. After considering the effects of increased pool dynamics loads, the licensee demonstrated that the margin thickness (defined as nominal thickness of 385 mils minus the maximum thickness required to meet allowables) is greater than the maximum metal loss found in each of the unrepaired areas. For example, the maximum thickness required to meet ASME Level B allowable was 0.341 in. under the load combination containing Intermediate Break LOCA (condensation oscillation load), increased EMRV load, and Operating Basis Earthquake load in the shell areas around the straps. The maximum unrepaired corrosion depth in this area was found to be 40 mils giving a margin of 4 mils before the corresponding allowable stress could be exceeded under the controlling load combination.

The exterior of the shell was coated in 1987 after shell rusting was observed during a routine inspection in 1986. The rusting was categorized as uniform and superficial with no evidence of rust scale. The metal loss was estimated to be no more than 2 mills.

The staff finds the explanations provided by the licensee to be acceptable provided the conditions are monitored and maintained in accordance with its commitment in Section 4.5.P.2 of the OCNGS Technical Specification.

3.3.3 Torus Internal Structures

The OCNGS torus consists of 20 mitered cylindrical shell segments (Bays). The individual segments are welded together at their intersections. At each intersection, the torus is stiffened with an internal T-shaped ring girder. At each ring girder location, the vent header has a welded ring collar. The vent header is supported by two columns pinned to the ring collar and to the web of the T-shaped ring girder. Another structure inside the torus is the catwalk which provides a continuous walkway in each bay of the torus. It consists of a walkway grating attached to a framework which is supported at each ring girder. Additionally, the torus ring girders are supporting spray header piping system, the core spray suction header restraint snubbers, the demineralized relief valve discharge piping system, and the safety relief valve discharge piping system.

The torus internal structures were analyzed during the OCNGS Mark I containment long-term program (Ref. 3) and accepted by the staff in 1984 (Ref. 4). In MPR report 1434 (Ref. 6), the MPR recalculated the stresses to account for the 4% increase in the EMRV setpoint pressures and found that the stresses in the support structures to be within the allowables for each of the load combinations considered in the original analysis. The staff reviewed the recalculated stresses, including the outlier evaluations, and found them to be acceptable.

The torus internal structures are subjected to the same type of environment as the torus shell, and thus, susceptible to corrosion. The licensee is monitoring their condition in accordance with its commitment in Section 4.5.P.2 of the OCNGS Technical Specification.

3.3.4 Torus Support Structures

The torus is supported by columns welded to the torus shell at the reinforced ring-girder locations. Saddle supports to the shell are provided at the center of each bay. The outside columns are laterally supported by cross bracings. The column and saddle supports are anchored to the concrete foundation by cast-in-place anchor bolts. In Ref. 6, MPR has recalculated the loads on the supporting structural components, and through one outlier evaluation demonstrated that the supporting structures could withstand the loads generated by the higher EMRV setpoints without exceeding the corresponding allowables. The staff finds the recalculations acceptable.

The torus support structures, particularly the lubrite plates, other base plates, the grout underneath, and the anchor-bolts are subjected to fluctuating stresses (due to torus movements under temperature and pressure variations) under normal operating conditions and EMRV actuations. These structures are also subjected to high temperature and high humidity environment and are susceptible to degradation. In Section 4.5.P.4 of the OCNGS Technical Specification (TS), the licensee has committed to visually examine the exterior of the torus shell whenever operation of a relief valve is indicated, and when the suppression pool temperature is indicated to be 160°F (and higher), and the primary coolant system pressure is greater then 180 psig. The OCNGS detailed procedure (Procedure No. 604.4.013) indicates that the visual examination requirement of the TS is applicable to the torus shell as well as to its support structures.

3.3.5 Conclusion

Thus, based on the review of the licensee's submittal, the responses to the staff's requests for additional information, and the load evaluation performed by the NRC staff, the staff concludes that the torus shell, its internal structures, and its support structures can withstand the increased pressure loads without exceeding the acceptance criteria established by NRC in NUREG-0661 (Safety Evaluation Report - Mark I Containment Long Term Program), provided these structures are monitored for potential degradation on periodic bases. The licensee's surveillance requirements in the Technical Specification and in the OCNGS plant procedure ensure that the licensee is monitoring and maintaining the shell, its internal structures, and its support structures on a periodic basis.

3.4 EFFECTS OF INCREASED PRESSURE AND THRUST LOADS ON EMRV DISCHARGE HEADER AND DISCHARGE Y-QUENCHER

In order to support the above stated TS change, the licensee evaluated the changes to the stresses and fatigue usage previously determined as part of the original Oyster Creek Nuclear Generating Station Mark I Containment Long-Term Program Plant Unique Analysis (OCPUA). The definition of the loads associated with EMRV actuation was originally performed with all five EMRVs at a set pressure of 1070 psig. In order to support this TS Change Request, the licensee performed an analysis with a set pressure of 1105 psig for all five EMRVs. The results of the OCPUA had determined that simultaneous actuation bounded sequential actuation, and the licensee performed an evaluation of the effects of the proposed TS change consistent with these original results.

The licensee determined that raising the EMRV setpoints to 1105 psig increases the EMRV actuation-induced loads by less than 3.3%. The licensee stated that this 3.3% factor was derived by analysis and that an additional 0.5% had to be added to this factor to account for a minor error found in the OCPUA Reports resulting in a correction to the design steam flow rate used in the original reports. The total 4.0% increase was then used to evaluate loading effects of the proposed EMRV actuation setpoints. The EMRV actuation-induced loads include:

(1) transient pressure and thrust loads on the EMRV discharge header;

(2) transient pressure and thrust loads on the EMRV discharge Y-quencher;

Components which were originally found to be below allowable stress values and/or loads in the OCPUA were increased by 4.0% irrespective of the amount of EMRV contribution to the total load. If these components were still below the allowable values with the 4.0% increase in load, they were considered acceptable. The majority of components fell into this category. The licensee stated that after applying the 1.04 factor to all stresses and loads reported in the OCPUA, a total of ten outliers were identified in which the factored stress and/or load exceeded the OCPUA allowable. For eight of these ten outliers, stress or loads were found to be acceptable by examining the OCPUA calculations and determining the effect of the EMRV load increase by increasing only the portion of the total stress or load that was due to EMRV discharge, as opposed to increasing the original OCPUA total load values.

The licensee then determined that for the EMRV discharge piping and the vent line/vent header intersections, the stresses reported in the original OCPUA analyses slightly exceeded allowable values for several load combinations, but were considered acceptable due to conservatism in the original analysis methods. To determine the acceptability of increasing the EMRV activation-induced loads on these components, the licensee reviewed the stress analyses which were performed to determine whether more realistic (less conservative) analysis methods could be used to more accurately determine the stresses for these components. The licensee stated that in these cases, the square root sum-of-the-squares summation method (SRSS) was used in accordance with NUREG-0484, for independent dynamic loads such as earthquake, EMRV discharge and loss of coolant accidents. Use of SRSS resulted in stresses below allowable values for the increased EMRV discharge loads for the vent line/vent header intersection load combinations. Also, use of SRSS resulted in stresses considered acceptable for all but one location on the EMRV discharge piping (i.e., stresses were equal to or less than stresses considered acceptable in the original OCPUA analyses.) At this one location, which is the connection between the main steam line and the south header discharge line, the licensee found it to be stressed to 21.66 ksi for the load case consisting of EMRV discharge plus deadweight loads. This stress is about 3.1% above the OCPUA allowable stress of 21.0 ksi. (Use of SRSS did not reduce the stress since there was only one dynamic load in this load combination.)

Because of the conservatisms built into the OCPUA load definition for EMRV discharge, and recognizing the conservatisms built into the allowable stress limits, the licensee considers this currently calculated overstress by 3.1% to be acceptable. The EMRV flow calculation as specified by NUREG-0661 includes a 5% general purpose conservatism. Therefore, the actual EMRV loading with the 1.04 factor applied, but with the 5% general purpose conservatism removed, results in a stress value which is less than the allowable stress for this load combination by at least 1%. In addition, the licensee identified an additional conservatism since the OCPUA demonstrates that the Oyster Creek torus could tolerate a five valve initial simultaneous actuation followed by a five valve subsequent simultaneous actuation without taking credit for staggered setpoints or operator action. The assumed time between actuations was 12 seconds or greater in the load definition for the OCPUA, and the licensee determined that multiple challenges would continue to be greater than 12 seconds apart for the proposed TS change. Since the proposed setpoints for all five valves are not equal and are not likely to actuate simultaneously, this results in additional conservatism. Therefore, the licensee determined that because of the conservative approaches taken in the OCPUA, this overstress is judged to be acceptable.

The licensee also determined the EMRV loads, resulting from the setpoint increase, affected the calculated fatigue usage by a factor of about 1.11. Consequently, the fatigue usages reported in the OCPUA were checked. The licensee determined that the total usage would not exceed the allowable total usage of 1.0.

The staff agrees with the licensee's assessment that the increase in stresses and fatigue usage, resulting from the proposed increase in the EMRV setpoints, will result in acceptable structural response of the EMRV branch connection to the Main Steam Line, EMRV discharge header and "Y-quenchers", and torus attached piping consistent with the Mark I Containment Long-Term Program methodologies.

3.4.1 Conclusion

Based on the above evaluation, the staff agrees that the analysis which the licensee has provided demonstrates the adequacy of the plant EMRV Main Steam Line branch connection, the EMRV discharge header, the Y-quencher, the torus attached piping, for the increased EMRV loads for the proposed EMRV setpoint TS change. The licensee has shown that for the increase in the loads due to the increase in the EMRV setpoints by 15 psi, the allowable stresses and allowable fatigue usage in these plant components will not be exceeded. Therefore, the proposed EMRV setpoint TS change has no significant safety impact and is acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

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 February 16, 1995, (60 FR 9056) Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Technical Specification Change Request from J. Barton (GPUN) to NRC, dated June 15, 1994

2. NUREG-0661, "Safety Evaluation Report, Mark I Containment Long Term Program - Resolution of Generic Technical Activity A-7," Dated June 1980.

3. MPR Report 733, "Plant-Unique Analysis Report, Suppression Chamber and Vent System - OCNGS," Dated August 1982.

4. Letter LS05-84-01-015; NRC to GPUN, "Safety Evaluation Report Associated with Mark I Containment Long Term Program - Structural Review Including the TER Prepared by the Franklin Research Center," Dated January 13, 1984.

5. Letter LS05-84-01-016; NRC to GPUN, "Safety Evaluation Report Associated with the Mark I Containment Long Term Program - Pool Dynamics Loads Including the TER Prepared by the Brookhaven National Laboratory," Dated January, 13, 1984.

6. MPR Report 1434 (Rev. 0), "Evaluation of Proposed Increase in Technical Specification Limits for EMRV Setpoint pressure on Mark I Containment Long-Term Program Analyses," Dated March 1994.

7. Letter from R. Keaton (GPUN) to NRC, "Responses to Request for Additional Information," Dated November 3, 1994.

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Date: February 21, 1995

7590-01

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. 177 to Facility Operating License No. NPF-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, to be implemented within 60 days of issuance.

The amendment revises Technical Specification 2.3.D to change the setpoints "Reactor High Pressure, Relief Valve Initiation" by increasing the setpoint value by 15 psig for each of the Electromatic Relief Valves in the Automatic Depressurization System.

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 July 5, 1994 (59 FR 34453). 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 the amendment will not have a significant effect on the quality of the human environment (60 FR 9056).

For further details with respect to the action see (1) the application for amendment dated June 15, 1994, as supplemented September 23, and November 23, 1994, (2) Amendment No. 177 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 NW., Washington DC, and at the local public document room located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Dated at Rockville, Maryland, this 21st day of February 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

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