June 7, 2000

Mr. Sander Levin
Acting Vice President
GPU Nuclear, Incorporated
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, New Jersey 08731

SUBJECT: NRC'S OYSTER CREEK INTEGRATED INSPECTION REPORT NO. 05000219/2000-003

Dear Mr. Levin:

On May 13, 2000, the NRC completed an integrated inspection at your Oyster Creek reactor facility. The enclosed report presents the results of that inspection. The results of this inspection were discussed on June 1, 2000, with you and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. No findings were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room and will be available on the NRC Public Electronic Reading Room (PERR) link at the NRC home page, http://www.nrc.gov/NRC/ADAMS/index.html.

We appreciate your cooperation. Please contact me at 610 337-5146 if you have any questions regarding this letter.

Sincerely,

***/RA/***

John F. Rogge, Chief
Projects Branch No. 7
Division of Reactor Projects

Docket/License Nos.: 05000219/DPR-16

Enclosure: NRC Inspection Report No. 05000219/2000-003

<u>cc w/encl:</u>
G. Busch, Manager, Nuclear Safety & Licensing
M. Laggart, Manager, Licensing and Vendor Audits
State of New Jersey

<u>Distribution w/encl:</u> **(VIA E-MAIL)**
Region I Docket Room (with concurrences)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
H. Miller, RA (to M. Fudge)
J. Wiggins, DRA (to G. Matakas)
J. Rogge, DRP
N. Perry, DRP
D. Screnci, PAO
B. Platchek, DRP

<u>HQ Distribution w/encl</u> **(VIA E-MAIL)**:
J. Shea, OEDO
E. Adensam, PD1, NRR
H. Pastis, NRR
T. Colburn, NRR
W. Scott, NRR
Inspection Program Branch, NRR (IPAS)

DOCUMENT NAME: G:\BRANCH7\Oyster Creek\OC2000003.wpd
After declaring this document "An Official Agency Record" it **<u>will</u>** be released to the Public.
**To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

| OFFICE | RI/DRP | E | RI/DRP | E | | | | |
|---|---|---|---|---|---|---|---|---|
| NAME | L. Dudes | | J. Rogge | | | | | |
| DATE | 06/7/00 | | 06/7/00 | | | | | |

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

| | |
|---|---|
| Report No. | 05000219/2000-003 |
| Docket No. | 05000219 |
| License No. | DPR-16 |
| Licensee: | GPU Nuclear, Incorporated<br>1 Upper Pond Road<br>Parsippany, New Jersey  07054 |
| Facility Name: | Oyster Creek Nuclear Generating Station |
| Location: | Forked River, New Jersey |
| Inspection Period: | April 2, 2000 - May 13, 2000 |
| Inspectors: | Laura A. Dudes,  Senior Resident Inspector<br>Thomas R. Hipschman, Resident Inspector<br>Aniello Della Greca, Senior Reactor Inspector, April 10-14<br>Kenneth Kolaczyk, Reactor Inspector, April 10-14<br>Alfred Lohmeier, Reactor Inspector, April 10-14 |
| Approved By: | John F. Rogge, Chief<br>Projects Branch 7<br>Division of Reactor Projects |

# SUMMARY OF FINDINGS

Oyster Creek Nuclear Generating Station
NRC Inspection Report 05000219/2000-003

The report covered a 6 week period of resident inspection, an engineering modification and 10 CFR 50.59 inspection by region based inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) in Inspection Manual 0609 (see Attachment 1).

- No findings were identified.

Report Details

Summary of Plant Status:

Oyster Creek remained at or near full power during the entire inspection period.

## 1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

### 1R02 Changes to License Conditions

a. Inspection Scope

The inspectors reviewed the safety evaluations (SEs) described below. The review was conducted to verify that changes to the facility or to the procedures as described in the Updated Final Safety Analysis Report (UFSAR) were reviewed and documented by the licensee in accordance with 10 CFR 50.59. The SEs were selected from the changes performed during the last two years taking into consideration safety significance of the change, risk to the structures, systems, and components affected, and impact on the three reactor safety cornerstones. The inspectors also reviewed, as applicable, GPUN's identification and resolution of problems related to SEs and associated changes.

| | |
|---|---|
| SE 000731-011 | Modification MD G075-007, Revision 0, 4160 V Switchgear 1C & 1D Protective Relay Replacement |
| SE 000615-002 | Modification MD G544-001, Revision 0, Remote shutdown Panel V-11-0034 Indication Circuit Modification |
| SE 328403- 001 | Modification MD G655, Revision 0, Thermal Dilution Gates Restoration |
| SE 000735-010 | Modification MD G803, Revision 0, Alternate DC Control Power for Switchgear 1D. |
| SE 000822-045 | Engineering Evaluation 0151-98, Revision 0, Alternate Replacement for Standby Gas Treatment System HEPA Filter |
| SE 945100-283 | Station Procedure 2000-ABN-3200.13A/B, Revision 0, Loss of DC Distribution System A and/or B |
| SE 000854-009 | Station Procedure 312.9, Revision 17, Primary Containment Control |
| SE 945100-088 | Station Procedure 310, Revision 68, Containment Spray System Operation |

The inspectors also reviewed a sample of changes, tests and experiments, as described below, for which the licensee determined that a safety evaluation was not required. This review was performed to verify that GPUN's threshold for performing safety evaluations was consistent with 10 CFR 50.59.

| | |
|---|---|
| 2000-ABN-3200.13 | Station Procedure, Revision 11, Response to Loss of All 125 Vdc. |
| 2000-ABN-3200.36 | Station Procedure, Revision 6, Loss of Offsite Power |
| 2000-ABN-3200.40 | Station Procedure, Revision 0, Stuck Open Electromatic Relief Valve |

| | |
|---|---|
| SP 304 | Station Procedure, Revision 30, Standby Liquid Control System Operation. |
| SCR-TR-IA0055 | Setpoint Change Request, Revision 0, Recirculation Pump Seal Cooler Outlet Temperature |
| SCR-TS-215-0006 | Setpoint Change Requests, High Temperature in RWCU Recirculation Pump Isolation Switch |
| SCR-LVA-001 | Setpoint Change Requests, Condenser Low Vacuum Alarm |
| 0283-98 | Commercial Grade Dedication Assessments, Alternate Replacement for Battery Chargers C1/C2/k2-K5 Relays |
| CGD-OC-98-0010 | Commercial Grade Dedication Assessments, Electrolytic Capacitor Evaluation |
| MD H227 | Modifications to Core Spray Supports NZ-2-R3 and NZ-R-11 |
| MD H305 | Modification of A and B Battery Rack Seismic Support |
| MD H335 | Permanent Power to Feedwater Pump Room Exhaust Fan |

The inspectors reviewed the licensee's problem identification and resolution program related to selected plant changes and safety evaluations. The review was conducted to verify that the licensee identified modification issues at the proper threshold and enters them in the corrective action program, and to evaluate the adequacy of the resulting corrective actions.

b. Issues and Findings

At Oyster Creek, the station safety-related dc power is supplied by a two battery system, B and C. A third nonsafety-related dc source is supplied by battery A, located in the same room as battery B. Automatic transfer switches between the A and B batteries ensure the reliability of power to some of the more important loads.

Prior to 1998, a single procedure, No. 2000-ABN-3200.13, addressed the loss of all three dc sources. In 1998, to improve the operator response to a loss dc power, the licensee developed two new procedures, one for the A & B batteries and the other for the C battery. During the development of procedure 2000-ABN-3200.13A/B, the licensee identified a concern that the coincident loss of batteries A & B could potentially result in the release of 6500 cubic feet of hydrogen to the turbine floor and approximately half of this volume into the main generator exciter cubicle.

The licensee did not identify a common cause that could result in the failure of both batteries. However, in the scenario they postulated, the loss of the A and B dc sources would result in the loss of both the normal (ac) and emergency (dc) hydrogen seal oil pumps. The consequent decay in the seal oil pressure would eventually allow the hydrogen in the generator to escape. The licensee believed that the hydrogen lost to the turbine building would have minimal consequences because of the large building volume. However, the hydrogen lost to the exciter cubicle could result in an explosion and fire. The sparking generator slip ring and exciter commutator would provide the ignition source for such a fire. The licensee developed this scenario in their Safety/Environmental Evaluation for the procedure change, No. SE-945100-283, dated February 25, 1998.

When the issue was originally discovered, the responsible engineer recommended that a modification be implemented to power the emergency seal oil pump from the C battery system. However, no formal review of the finding was performed by the appropriate disciplines and no action was taken to initiate the design change recommended by the responsible engineer. At the time of the inspection, the licensee had not evaluated the impact of the finding on the safety systems or on the fire hazard analysis. However, an evaluation was required because a fire in the A and B battery room could be the common cause for the failure of both batteries and result in a fire in another area of the plant, the turbine building. This condition had not been addressed by the Oyster Creek fire hazard analysis.

As a result of the inspectors observations, on April 18, 2000, the licensee initiated a corrective action program item, CAP No.02000-0534. Subsequently, they conducted a survey of the area and a review of the cables crossing the turbine building. GPU concluded that a fire in the turbine building would not adversely affect the safe shutdown of the plant or change the conclusions of the fire hazard analysis. The CAP also indicated that the licensee would evaluate the possibility of powering the emergency seal oil pump from the redundant DC source. This issue was identified at the end of the inspection and the inspectors did not have an opportunity to review the adequacy of GPU's assessment or evaluate the issue using the significance determination process. Consequently, this item remains unresolved pending further NRC review. **(URI 5000219/2000-003-01)**

1RO4 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial system walkdown (visual inspection) of the turbine building closed-loop cooling water (TBCCW) system during the period that the 1-2 pump was out of service for planned maintenance.

The inspectors also performed a walkdown of the No. 2 emergency diesel generator (EDG) during planned maintenance on the No. 1 EDG.

b. Issues and Findings

There were no findings identified.

1RO5 Fire Protection

a. Inspection Scope

The inspectors conducted fire protection activities consisting of plant walkdowns, discussions with fire protection personnel, a review of procedure 333, *Plant Fire Protection System,* and the Oyster Creek Fire Hazards Analysis Report. Plant walkdowns included observations of combustible material control, fire detection and equipment suppression availability, and compensatory measures. The inspectors conducted fire protection inspections for the following areas:

- A/B 480 volt switchgear room
- 23' reactor building, thermolag fire barrier upgrade areas
- C Battery room

- 4160 volt switchgear rooms C/D

b. Issues and Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

As a result of emergent work on the switchyard Bank 7 step down transformer which converts 230 KV power to 34.5 KV power in the Oyster Creek switchyard, the inspectors reviewed the implementation of Maintenance Rule (MR) as related to the operation of the start-up transformers which supply power directly to the vital busses either from the normally used offsite 230 KV lines or through the offsite power 34.5 KV lines. The inspector verified that the licensee appropriately scoped, established performance criteria and monitored the performance of the transformers in accordance with 10 CFR 50.65.

b. Issues and Findings

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

.1 Emergent Work on the 34.5 KV Transformer Bank 7

a. Inspection Scope

As a result of routine thermography on the switchyard 34.5 KV transformers, the licensee identified that a high temperature on the bank 7 transformer load tap changer may be indicative of a degraded component. The licensee took immediate action to develop a plan to troubleshoot and repair (if necessary) the degraded component. This work in the switchyard had the potential to initiate and or complicate a reactor scram. The inspector reviewed the licensee' s safety evaluation and emergent risk assessment.

b. Issues and Findings

There were no findings identified.

.2 Planned Containment Spray, Troubleshooting Action Plan Assessment

a. Inspection Scope

As part of a system engineering action plan to gather performance data on the containment spray system, a troubleshooting action plan was implemented as part of the planned maintenance outage. A risk assessment was performed because the specific actions to be taken were outside of the normal post maintenance surveillance testing. The inspector reviewed the risk assessment and action plan.

b. Issues and Findings

There were no findings identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determinations associated with the following plant equipment challenges:

- Operability of the alternate fire water supply to the core spray system (CAP 2000-0463).
- Operability of back-up offsite power while the Q-121 line was out of service. (CAP 2000-582).
- Operability of the Containment Spray heat exchangers due to loose anchor bolts. (CAP 2000-0494).
- Operability of Electromatic Relief Valves without backshell sealing (CAP 2000-632).

b. Issues and Findings

There were no findings identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the operator work-around database and associated corrective action reports and work orders.

b. Issues and Findings

There were no findings identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed selected portions of the permanent plant changes described below.

Mitigation Systems:

| | |
|---|---|
| MD G070 | ESW [Emergency Service Water] System 1 Pipe Vent Modification |
| MD G655 | Thermal Dilution Gates Restoration |
| MD H227 | Core Spray Supports NZ-2-R3 and NZ-R-11 Modifications |
| MD H335 | Permanent Power to Feedwater Pump Room Exhaust Fan |
| 1662-00-0002 | Commercial Grade Dedication - Direct replacement and Seismic Qualification of 6-inch, 600 lbs Gate Valve |
| SP 310 | Station Procedure, Revision 68, Containment Spray System Operation |
| SP 308 | Station Procedure, Emergency Core Cooling System Operation |
| 2000-ABN-3200-13A/B | Loss of dc distribution system A and or B |
| Standing Order 50 | Operation of the 1-8 Sump |
| Standing Order 21 | Allowable Bypass Configuration for APRM/LPRM System |
| C 1302730-5350-010 | Engineering Calculation, Revision 2, Thermal Overload Heater Sizing for Isolation Condenser MOVs |
| C-1302-153-5450-081 | Engineering Calculation, Cleanup System Isolation Valve Stroke Time Evaluation |
| MD G927 | Separation of Service Air From Containment Spray |

Barrier Integrity:

| | |
|---|---|
| MD G108 | Main Steam Isolation Valve Modifications |
| MD G857 | Feedwater Check Valves V-2-71,72,73, V-2-74 Disc and Hinge Alternate Replacement |
| 0151-98 | Alternate Replacement for Standby Gas Treatment System HEPA Filter, Revision 0 |
| 0241-98MSIV | Step Stud Substitution |
| SCR V-43-0155 | Dilution Pump House Setpoint Change Request |
| SP 312.9 | Station Procedure, Revision 17, Primary Containment Control |
| 083-261-BRL-1 | Engineering Calculation Core Shroud Stiffness with Failed Vertical Welds and Installed Wedges |

Event Initiators:

| | |
|---|---|
| MD G075-007 | 4160 V Switchgear 1C & 1D Protective Relay Replacement, Revision 0 |
| MD G544-001 | Remote shutdown Panel V-11-0034 Indication Circuit Modification, Revision 0 |
| MD G803 | Alternate DC Control Power for Switchgear 1D, Revision 0 |
| MD H305 | Modification of A and B Battery Rack Seismic Support |

The plant modifications were selected from approved changes that were either completed during the last two years or were scheduled to be installed during the 2000 refueling outage. The selection was based on risk significance, impact on the three reactor safety cornerstones (initiating events, mitigating systems and barrier integrity), and representative activities from various disciplined and engineering specialities. These modifications included equivalency evaluations, minor and major modifications, setpoint changes and design calculations, and a variety of normal, abnormal, and emergency plant procedures. The inspectors directed their review to selected portions of the design, implementation, post-modification testing, and closeout documentation. As appropriate, they held discussions with the responsible design engineers and other personnel familiar with the plant changes. These discussions addressed in particular the scope and extent of the changes as well as the licensee's identification and resolution of problems that initiated the changes. The inspectors also conducted field observations of the installed modifications.

b. Issues and Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector reviewed and observed portions of the following post maintenance testing:

- Emergency diesel generator testing following planned system maintenance and modification.
- Core Spray following planned system maintenance.
- Containment Spray Pump Operability and Heat Exchanger Thermal Performance following planned maintenance

b. Issues and Findings

There were no findings identified.

1R22 Surveillance Testing

.1 Containment Spray and Emergency Service Water System I Pump Operability and Inservice Test.

a. Inspection Scope

During the week of April 2, 2000, the inspector reviewed the surveillance testing conducted to demonstrate the operability of containment spray and emergency service water pumps, and selected valves. The performance of this test satisfies the requirements of technical specification sections 4.4.C.1, 4.4.D.1 and the Inservice test requirements for the systems pumps and selected valves. The inspector reviewed NUREG 1482, “Guidelines for Inservice Testing at Nuclear Power Plants,” the GPU surveillance procedure 607.4.004, “Containment Spray and Emergency Service Water System I Pump Operability and Inservice Test,” and previous CAP reports related to this system.

b. Issues and Findings

There were no findings identified.

.2 Isolation Condenser Valve Operability and Inservice Test

a. Inspection Scope

During the week of April 10, 2000, the inspector review surveillance procedure 609.4.001, “Isolation Condenser Valve Operability and Inservice Test. The review verified the stroke time acceptance criteria list in attachment 1 of the procedure met the licensee’s Inservice test program guidelines. The inspector also observed portions of the test to verify appropriate valve stroke timing techniques were used.

b. Issues and Findings

There were no findings identified.

## 4. OTHER ACTIVITIES [OA]

4OA6 Meetings

Exit Meeting Summary

On June 1, 2000, the resident inspectors presented the inspection results to Mr. Sander Levin and other members of licensee management. The engineering modification inspection presented their inspection results to the licensee on April 14, 2000. In addition, on June 7, 2000 a discussion between the Mr. Neil Della Greca and Mr. Sander Levin to present the planned NRC review of the hydrogen seal oil pump corrective actions assumptions and conclusions. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

## PARTIAL LIST OF PERSONS CONTACTED

Licensee (in alphabetical order)

G. Busch, Manager, Nuclear Safety & Licensing
R. Ewart, Site Security Manager
S. Levin, Director, Operations and Maintenance
D. McMillan, Director, Equipment Reliability
K. Mulligan, Plant Operations Director
J. Perry, Plant Maintenance Director
D. Slear, Director, Configuration Control
R. Tilton, Manager, Assessment

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | |
|---|---|
| URI 05000219/2000-003-01 | Untimely resolution of the potential to create two fire zones simultaneously due to a common mode failure resulting from a fire in the A/B battery room. (1R02) |

## LIST OF ACRONYMS USED

| | |
|---|---|
| AC | Alternating Current |
| APRM | Average Power Range Monitor |
| CAP | Corrective Action Plan |
| CFR | Code of Federal Regulations |
| DC | Direct Current |
| CGD | Commercial Grade Dedication |
| DRP | Division of Reactor Projects |
| DRS | Division of Reactor Safety |
| EDG | Emergency Diesel Generator |
| ESW | Emergency Service Water |
| GPUN | General Public Utilities (GPU) Nuclear |
| HVAC | Heating, Ventilation and Air Conditioning |
| IST | In-Service Test |
| JO | Job Order |
| LPRM | Local Power Range Monitor |
| MOV | Motor-Operated Valve |
| MR | Maintenance Rule |
| NRC | Nuclear Regulatory Commission |
| NRR | Office of Nuclear Reactor Regulation |
| OCNGS | Oyster Creek Nuclear Generating Station |
| PDR | Public Document Room |
| RCA | Radiologically Controlled Area |
| RP&C | Radiological Protection and Chemistry |
| RPS | Reactor Protection Systems |
| RWP | Radiation Work Permit |
| SCR | Setpoint Change Request |
| SE | Safety Evaluations |
| TBCCW | Turbine Building Closed Loop Cooling Water |
| TS | Technical Specifications |
| UFSAR | Updated Final Safety Analysis Report |

ATTACHMENT 1

# NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

**Reactor Safety**
- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

**Radiation Safety**
- Occupational
- Public

**Safeguards**
- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.