OYSTER CREEK NUCLEAR GENERATING STATION

OPERATING LICENSE
NO. DPR-16

LICENSE AMENDMENT
REQUEST NO. 263
DOCKET NO. 50-219

Applicant submits by this License Amendment Request No. 263 to the Oyster Creek Nuclear Generating Station.

By:

Michael B Roche

Michael B. Roche
Vice President and Director
Oyster Creek

Sworn to and Subscribed before me this 19th day of September, 1998.

Diana M. DeBlasio

A Notary Public of New Jersey

DIANA M. DeBLASIO
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 6-13-2001

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF |
GPU NUCLEAR, INC. | DOCKET NO. 50-219

CERTIFICATE OF SERVICE

This is to certify that a copy of License Amendment No. 263 for the Oyster Creek Nuclear Generating Station, filed with the U.S. Nuclear Regulatory Commission on September 19 has this day of 1998 been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the U.S. mail, addressed as follows:

The Honorable Louis A. Amato
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

By: Michael B Roche

Michael B. Roche
Vice President & Director
Oyster Creek

Attachment 1

Oyster Creek Nuclear Generating Station
License Amendment Request No. 263

I. Change Requested

GPU Nuclear requests that Updated Final Safety Analysis Report (UFSAR) Section 5.4.8.2 be revised to address the use of a freeze seal as a temporary part of the reactor coolant pressure boundary.

II. Discussion of Proposed Change

In preparation for the upcoming refueling outage GPUN has identified an unreviewed safety question as defined in 10 CFR 50.59(a)(2). The proposed activity involves isolating the Reactor Water Cleanup System (RWCS) from the Reactor Pressure Vessel (RPV) by means of a temporary freeze seal. The proposed activity constitutes an unreviewed safety question in that it introduces a malfunction of a different type than has been previously evaluated in the UFSAR. Accordingly, the purpose of License Amendment Request No. 263 is to revise Section 5.4.8 of the UFSAR such that it incorporates the use of a freeze seal as a temporary part of the reactor coolant pressure boundary.

The RWCS is a filtration and demineralization system for maintaining the purity of the water in the Reactor Coolant System (RCS). The system includes a regenerative heat exchanger, a non regenerative heat exchanger, a pressure reducing station, cleanup filters and auxiliaries, a cleanup demineralizer, cleanup pumps, a surge tank, a flow control station, a reactor drain station, isolation valves, piping, instrumentation and controls. Under normal operation, reactor coolant flows under reactor pressure from the suction of reactor Recirculation Pump B and is cooled to 120°F in the regenerative and non regenerative heat exchangers (in series). The coolant is then reduced to 110 psig, filtered, demineralized, and pumped through a flow control valve and the regenerative heat exchanger to the discharge of reactor Recirculation Pump B. The supply line from the suction of reactor Recirculation Pump B to the RWCS contains a motor operated isolation valve inside the drywell and two parallel motor operated valves outside the drywell. The return line to the discharge of reactor Recirculation Pump B contains one motor operated valve outside the drywell and one check valve inside the drywell. Between the return line check valve inside the drywell and the reactor vessel there is a 6" normally locked open manual valve (V-16-63) that isolates the RWCS return line from the reactor vessel.

As part of the scope of work for the upcoming 17R refueling outage, the valve bonnet of V-16-63 may need to be removed to facilitate repairs or inspections. In order to isolate the RWCS return line from the reactor vessel and implement repairs, GPUN intends to use a nitrogen freeze seal as a temporary reactor coolant pressure boundary barrier. Moreover, the existing disc/stem of V-16-63 will be used to provide an additional barrier. A disc/stem restraining device will be used to physically wedge the disc into the seat of the valve body during the repair activity.

Consistent with NRC Information Notice No. 91-41 "Potential Problems With The Use Of Freeze Seals", considerations with respect to temperature detection devices, proper personnel training, use of freeze seal mockups, use of contingency plans and evaluating the need for additional makeup supplies have been incorporated into the development of the valve repair activity. To provide additional assurance that the freeze seal barrier will not be jeopardized, a second nitrogen supply will be located as close to the freeze seal chamber as practical and will be available in the event the primary nitrogen supply is lost.

A freeze seal mockup test was performed to demonstrate the capability of the freeze seal methodology and barrier. Venting the freeze seal mockup piping downstream of the freeze seal barrier for 30 minutes validated the isolation integrity of the freeze seal methodology and barrier. In addition, a loss of nitrogen event was simulated during the mockup test. A freeze seal barrier was formed, its integrity verified and held for one (1) hour prior to removal of the nitrogen supply. The freeze seal was then exposed to water at 99° F and approximately 100 psig on the RPV side. Based on the above it was concluded that, under conditions more severe than expected in the plant, there would be no loss of the freeze seal for 55 minutes after a loss of nitrogen.

While the valve repair activity is underway, two reliable isolation barriers will be provided, except for the case in which the valve disc must be removed. If it becomes necessary to remove the disc in order to repair or replace the stem or disc, mockup testing has demonstrated that removal and replacement of the disc can be completed within 55 minutes. The mitigating action to be taken upon a loss of nitrogen supply with the stem/disc removed is to install a valve bonnet seal plate assembly and thereby establish integrity of the reactor coolant pressure boundary. It is estimated that installation of the valve bonnet seal plate assembly will take 15 minutes. If at any time following disc removal it is determined that redundant isolation cannot be established within 55 minutes of the initial disc removal, the valve bonnet seal assembly will be installed to minimize the duration that a single isolation barrier is in place.

The above discussion assumes the following plant conditions:

- Plant mode switch in either the SHUTDOWN or REFUELING position.
- RPV is reassembled with fuel in the vessel.
- The RCS bulk temperature, as measured by TE-31D, shall not exceed a temperature of 99°F prior to the initiation of the valve repair activity and no evolutions involving removal of the valve disc shall occur if the RCS bulk temperature exceeds 99°F.
- RPV water level shall not exceed 210 inches above TAF.
- Decay heat from the RPV will be removed by the Shutdown Cooling System via the RBCCW/Service Water Systems.
- Both Core Spray Systems shall be operable in accordance with Technical Specifications.
- Fire Protection System shall be operable and capable of delivering water to the Core Spray System.
- At least one loop of the Reactor Recirculation System must remain open with the suction and discharge valves in the full open position.
- The Emergency Diesel Generators and their respective Safety Related electrical distribution systems are returned to service and available to support emergency operation.
- Offsite power shall be available.
- Secondary Containment shall be operable in accordance with the Technical Specifications.
- Both Standby Gas Treatment Systems will be operable.
- The RPV will be vented to the atmosphere and the established vent path will be tagged in the open position.
- RWCU System pumps P-16-001A and P-16-001B will be electrically de-energized.

- RWCU System valves V-16-1, V-16-2, V-16-14 and V-16-61 will be tagged in the closed position.

- Provisions will be in place to remove any equipment traversing the Drywell airlock to ensure that the airlock can be closed in a timely fashion should the need arise.

- The seal plate assembly for V-16-63 shall be staged and ready for fit-up and installation.

III. Safety Assessment

This License Amendment Request requests that UFSAR Section 5.4.8.2 be revised to address the use of a freeze seal as a temporary part of the reactor coolant pressure boundary.

A failure of the proposed freeze seal barrier constitutes a malfunction of equipment important to safety that has not been previously evaluated in the UFSAR. However, the resultant consequences (an unisolatable leak of reactor coolant into the Drywell) and mitigation actions are bounded by UFSAR evaluations and plant Technical Specification requirements.

Technical Specifications require the Core Spray Systems (CSSs) and Fire Protection System to be operable with irradiated fuel in the vessel, that adequate makeup is available when work is performed on the reactor coolant pressure boundary or connected systems that could result in an inadvertent loss of reactor coolant. GPUN has determined that the CSSs are capable of supplying sufficient water to maintain the RPV water level at or above 56 inches TAF assuming a 6 inch opening in the reactor coolant system. In the highly unlikely event of a loss of reactor coolant system isolation, a loss of RPV inventory would result in the actuation of the CSSs and the maintenance of water level at or above 56 inches TAF.

The proposed repair activity will occur with the Drywell airlock and primary containment open; however, the secondary containment will be intact. As discussed in UFSAR Section 6.2.3 "Secondary Containment Functional Design", the secondary containment functions as the primary containment during periods when the primary containment is open, such as during a refueling outage. The plant was designed to accommodate this mode of operation, which is in accordance with the plant Technical Specifications. In the event of a loss of the freeze seal and subsequent leak, the Drywell airlock will be closed. The leak would be predominately a liquid release with any gaseous effluent contained within the secondary containment.

An inadvertent loss of reactor coolant inventory with the plant not operating at power was not specifically considered in the spectrum of design basis accidents previously evaluated in the UFSAR. However, the consequences of such an event are bounded by those analyzed in the UFSAR.

Loss of reactor coolant accidents analyzed in Section 15 of the UFSAR bound the consequence of this accident. Pipe break accidents analyzed in UFSAR Section 15.6.5 "Decrease in Reactor Coolant Inventory Events" consider the effects of a blow down of an operating plant and the releases from primary containment due to failed fuel aggravated by the driving pressure inside containment. The scenario postulated here releases low energy reactor coolant (sub-cooled at atmospheric pressure) to the containment which has no impact on containment pressure. All required safety related equipment internal to the Drywell is qualified to DBA LOCA conditions and, therefore, the DBA LOCA conditions (temperature, pressure, humidity, flooding) bound the conditions associated with this scenario. For equipment external to the Drywell, no unusual conditions are expected, therefore, all external equipment will remain operable. Given that the energy of the reactor coolant is significantly less and no safety related systems, structures or components will be challenged by the direct or indirect effects of the event, any loss of RPV water inventory during the proposed repair activity will be less severe than those evaluated in UFSAR Section 15.6.5.

UFSAR Section 15.7.4 "Design Basis Fuel Handling Accidents in the Containment" describes accidents that occur when the secondary containment is performing the function of primary containment. This section of the UFSAR considers the worst case to be when the Drywell and reactor heads are off and a fuel handling accident occurs. In that event, the radiological release develops from the effects of a failed fuel bundle. The proposed repair activity does not involve fuel movement and therefore is bounded by the above UFSAR Section.

Based on the above, it is concluded that the proposed change does not adversely affect nuclear safety or safe plant operation.

IV. <u>Information Supporting a Finding of No Significant Hazards Consideration</u>

GPU Nuclear has determined that this License Amendment Request poses no significant hazards as defined by 10 CFR 50.92. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. <u>The License Amendment Request does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

The proposed repair activity involves the placement of temporary isolation barriers, including a freeze seal, in the RWCU System piping in order to isolate valve V-16-63 from the RCS while repairs are being made. The isolation barriers fulfill the function of the valve body, which is passive integrity. The repair activity is similar to other activities routinely performed during refueling outages that depend upon single isolation barriers. The plant was designed to permit such work with appropriate isolation barrier(s) in place. The work associated with the proposed repair activity is consistent with this premise.

The accident considered in this evaluation is a maintenance repair activity with a RCS leak that, without adequate makeup, would uncover the reactor core. Effective isolation provisions have been incorporated into the scope of the proposed repair activity which will minimize the probability that a RCS leak will occur. The freeze seal barrier has been demonstrated to last 55 minutes following a loss of nitrogen. The mitigating action to be taken upon a loss of nitrogen supply with the stem/disc removed is to install a valve bonnet seal plate assembly and thereby establish integrity of the reactor coolant pressure boundary. In addition, sufficient makeup capacity is provided to maintain the RPV water level at or above 56" TAF.

Failure of the freeze seal barrier with the valve disc/stem removed would result in a loss of RCS water inventory. The proposed repair activity is bounded by the events evaluated in UFSAR Sections 15.6.5 "Decrease in Reactor Coolant Inventory Events" and 15.7.4 "Design Basis Fuel Handling Accidents in the Containment".

Based on the above, the proposed activity does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. <u>The License Amendment Request does not create the possibility of a new or different kind of accident from any previously evaluated</u>

As indicated above, the accident considered in this evaluation is a maintenance repair activity with a RCS leak that, without adequate makeup, would uncover the reactor core. The proposed repair activity is bounded by the events evaluated in UFSAR Sections 15.6.5 "Decrease in Reactor Coolant Inventory Events" and 15.7.4 "Design Basis Fuel Handling Accidents in the Containment". As such, the proposed License Amendment does not create a new or different kind of accident from any previously evaluated.

3. <u>The License Amendment Request does not involve a significant reduction in a margin of safety.</u>

With respect to the piping subjected to the freeze seal, an evaluation of stress and materials issues concluded that the ductility and notch toughness of the pipe base metal, weld metal, and weld heat affected zone will remain high during the operation. In addition, no permanent changes to the base metal, weld metal or heat affected zone material properties or corrosion resistance are expected. Moreover, the maximum stress intensity in the cooled weld is acceptable per ASME Codes or B31.1 requirements. In light of the above, it was concluded that the pipe condition will not change as a result of the freeze seal and that it will retain its capabilities to meet its design loading.

A decrease in reactor coolant inventory caused by a leak or rupture is a LOCA condition that has been evaluated in the UFSAR. The proposed repair activity is bounded by the events evaluated in UFSAR Sections 15.6.5 "Decrease in Reactor Coolant Inventory Events" and 15.7.4 "Design Basis Fuel Handling Accidents in the Containment". The proposed repair activity will be performed with at least one loop of the Reactor Recirculation System in the open position whereas the bounding events include all loops open. However, since the potential energy release from the primary systems is significantly less than that which would be released for the DBA event, the conditions with closed loops are bounded. One train of the Core Spray System is capable of providing sufficient water to restore the RPV water level, both trains will be operable during the proposed repair activity.

Based on the above, the proposed License Amendment does not involve a significant reduction in a margin of safety.

V. Information Supporting an Environmental Assessment

An environmental assessment is not required for the License Amendment Request as the proposed change conforms to the criteria for "actions eligible for categorical exclusion" as specified in 10 CFR 51.22(c)(9). The License Amendment Request will have no impact on the environment. The License Amendment Request does not involve a significant hazards consideration as discussed in the preceding section. The proposed activity does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed activity does not involve a significant increase in individual or cumulative occupational radiation exposure.

V. Conclusion

As defined in 10 CFR 50.59(a)(2), the proposed repair activity constitutes an unreviewed safety question because the use of a freeze seal introduces a malfunction of a different type than has been previously evaluated in UFSAR. In accordance with 10 CFR 50.59(c), License Amendment Request No. 263 seeks the Commission's approval of the proposed repair activity by revising Section 5.4.8 of the UFSAR such that it incorporates the use of a freeze seal as a temporary part of the reactor coolant pressure boundary. An inadvertent loss of reactor coolant inventory with the plant not operating at power was not specifically considered in the spectrum of design basis accidents previously evaluated in the UFSAR. However, the consequences of such an event are bounded by those analyzed in the UFSAR.

The proposed activity has been reviewed in accordance with Section 6.5 of the Oyster Creek Technical Specifications. As discussed above, using the standards in 10 CFR 50.92, GPU Nuclear has determined that there are no Significant Hazards Considerations involved with this Request for a Change to the Licensing Basis.

Attachment 2

Oyster Creek Nuclear Generating Station
License Amendment Request No. 263
Proposed Revision - UFSAR Section 5.4.8.2

If the Requested Change to the Licensing Basis is approved, the following Final Safety Analysis Report changes will be submitted in the next FSAR Update:

5.4.8.2 System Description

The system includes a regenerative heat exchanger, a non regenerative heat exchanger, a pressure reducing station, cleanup filters and auxiliaries, a cleanup demineralizer, cleanup pumps, a surge tank, a flow control station, a reactor drain station, isolation valves, piping, instrumentation and controls (Drawing GE148F444).

Under normal operation, reactor coolant flows under reactor pressure from the suction of reactor Recirculation Pump B, is cooled to 120°F in the regenerative and non regenerative heat exchangers (in series), its pressure reduced to 110 psig, filtered, demineralized, and pumped through a flow control valve and the regenerative heat exchanger to the discharge of reactor Recirculation Pump B. When reactor pressure is insufficient to maintain the required suction pressure at the cleanup recirculation pump, an auxiliary cleanup pump is placed in operation.

For draining the reactor, some of the cleanup system effluent flow is directed to the hotwell or to radwaste via a second flow control valve at the reactor drain station. The normal drain path is through the recirculation loop and cleanup system. There is also a drain line, from the bottom of the reactor vessel, which has normally open manual valves to the cleanup system.

The system is operated to maintain low levels of reactor water conductivity and undissolved solids. Conductivity is monitored at the influent (two cells) and at the effluent (one cell) of the demineralizer. Recording capability is provided, and abnormal conditions alarmed in the Control Room. The design flow rate is 380,000 lbs/hr.

The system supply line has a motor operated isolation valve inside the drywell and two parallel motor operated valves outside the drywell. The return line has one motor operated valve outside the drywell and one check valve inside the drywell. The isolation valves will close, and the cleanup pumps will stop automatically under any of the following conditions:

- Low flow, or outlet valve shut, for the cleanup filter in service.
- High auxiliary pump cooling water outlet temperature.

- High non regenerative heat exchanger outlet temperature (reactor coolant).
- High pressure from the pressure reducing station.
- Liquid poison system flow into the reactor vessel.
- High drywell pressure.
- Low-low reactor water level.

An exception to the above is valve V-16-61 which closes only on low-low reactor water level and high drywell pressure.

Reactor Water Cleanup System safety-related motor-operated valves are included in the Generic Letter (GL) 89-10 Motor-Operated Valve (MOV) Program as noted in the OCNGS Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program. This program has reestablished the design basis for safety-related motor-operated valves. Critical design bases assumptions such as design bases differential pressure, safety function - open vs. close, minimum available AC/DC voltages, actuator gearing, torque switch control logic, valve factors, stem friction coefficients, and valve stroke times have been established in assessing GL 89-10 design bases capability. Plant changes or activities which can affect these design bases assumptions must consider the affect on the capability of GL 89-10 motor-operated valves to perform their safety function and on safety margins established for these valves.

The major characteristics of the Reactor Cleanup System components are presented in Table 5.4-1. The equipment is designed to ASME B&PV Code, Section III, Class C on the primary side. The tube side of the heat exchanger is ASME VIII.

The pressure reducing station consists of a pressure control valve and a bypass control valve, and relief valves. Pressure is maintained at or below 110 psig in the filter and demineralizer portion of the cleanup system. High pressure from the pressure reducing station trips the cleanup system isolation valves and pumps.

The pressure control valve is a 4 inch, globe, single seat valve, air diaphragm operated, air to open, spring and flow to close, with pressure controller. A pressure relief valve located just downstream of the pressure control valves protect the low pressure portions of the cleanup system. One 6 inch valve can discharge up to 125 pounds per second through a 20 inch line and one isolation check valve to the torus. A remote operated solenoid leakoff valve is used to detect relief valve leakage. One 1 inch valve is provided in line to the Reactor Building Equipment Drain Tank. The filters, the demineralizers, and other isolable portions of the system have relief valves which discharge to the Reactor Building Equipment Drain Tank.

Cleanup system flow is normally maintained at approximately 400 gpm. The system flow is measured by a flow element at the cleanup demineralizer inlet. Low flow at this element automatically starts the filter precoat pump to recycle, in order to hold the filter cake.

The flow control valve is a 4 inch globe, air diaphragm operated valve, air to open, spring and flow to close. This valve is located between the cleanup pumps and the regenerative heat exchanger. An electropneumatic converter supplies control air to the valve diaphragm, and is controlled by the flow controller, with feedback from the flow element. Cleanup pump suction header low pressure results in a flow reduction demand to the flow controller. A manually operated bypass valve is installed around the valve as backup to the automatic flow control. A flow integrator is provided to monitor system performance. The system is normally operated in the manual mode.

The reactor drain station consists of an auxiliary pressure-reducing valve and orifice in series, a motor operated orifice bypass valve, a flow element, and motor-operated shutoff valves to the hot well and to radwaste. The takeoff line for this station is located upstream of the cleanup recirculation pumps.

A normally locked open 6 inch manual valve (V-16-63) is located inside the drywell downstream of the return line check valve and provides a means of isolating the return line from the RPV. In order to implement repairs to V-16-63 with the plant mode switch in either the SHUTDOWN or REFUELING position, the use of a freeze seal positioned between the Reactor Vessel and V-16-63 may be necessary, subject to the following restrictions:

- **RPV is reassembled and fuel is in the vessel.**

- **The reactor coolant system bulk temperature, as measured by TE-31D, shall not exceed a temperature of 99°F prior to the initiation of the valve repair activity and no evolutions involving removal of the valve disc shall occur if the reactor coolant system bulk temperature exceeds 99°F.**

- **RPV water level shall not exceed 210 inches above TAF.**

- **Decay heat from the RPV will be removed by the Shutdown Cooling System via the RBCCW/Service Water Systems.**

- **Both Core Spray Systems shall be operable in accordance with the Technical Specifications.**

- **Fire Protection System shall be operable and capable of delivering water to the Core Spray System.**

- **At least one loop of the Reactor Recirculation System must remain open with the suction and discharge valves in the full open position.**

- **The Emergency Diesel Generators and their respective NSR electrical distribution systems are returned to service and available to support emergency operation.**

- **Offsite power shall be available.**

- **Secondary Containment shall be operable in accordance with the Technical Specifications.**

- **Both Standby Gas Treatment Systems will be operable.**

- **The RPV will be vented to the atmosphere and the established vent path will be tagged in the open position.**

- **RWCU System pumps P-16-001A and P-16-001B will be electrically de-energized.**

- **RWCU System valves V-16-1, V-16-2, V-16-14 and V-16-61 will be tagged in the closed position.**

- **Provisions will be in place to remove any equipment traversing the Drywell Airlock to ensure that the airlock can be closed in a timely fashion should the need arise.**

- **The seal plate assembly for V-16-63 shall be staged and ready for fit-up and installation.**