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10 CFR 50
10 CFR 51
10 CFR 54

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Supplemental Information Related to Oyster Creek Generating Station License Renewal Application (TAC No. MC7624)

Reference: AmerGen letter 2130-06-20292, dated April 18, 2006, "Response to NRC Request for Additional Information, dated March 20, 2006, Related to Oyster Creek Generating Station License Renewal Application (TAC No. MC7624)"

In the referenced letter, AmerGen Energy Company, LLC (AmerGen) provided additional information to the NRC related to Section 3.5 of the Oyster Creek Generating Station License Renewal Application (LRA). Following NRC staff review of this information, it was determined that additional clarifications were needed to the responses to RAIs 3.5-4 and 3.5-6. The Enclosure to this letter provides these clarifications.

If you have any questions, please contact Fred Polaski, Manager License Renewal at 610-765-5935.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on 07-10-2006

Michael P. Gallagher
Vice President, License Renewal
AmerGen Energy Company, LLC

Enclosure: Clarification of 4/18/06 AmerGen Responses to RAIs 3.5-4 and 3.5-6

A114

cc: Regional Administrator, USNRC Region I, w/o Enclosure
USNRC Project Manager, NRR - License Renewal, Safety, w/Enclosure
USNRC Project Manager, NRR - License Renewal, Environmental, w/o Enclosure
USNRC Project Manager, NRR - OCGS, w/o Enclosure
USNRC Senior Resident Inspector, OCGS, w/o Enclosure
Bureau of Nuclear Engineering, NJDEP, w/Enclosure
File No. 05040

ENCLOSURE

UPDATED RESPONSES TO RAI 3.5-4 AND 3.5-6

AMERGEN ENERGY COMPANY, LLC
OYSTER CREEK GENERATING STATION

RAI 3.5-4

Component type "Shielding Blocks and Plates," uses patented material "Permali," for which no aging effects are indicated in Table 3.5.2.1.1. The staff requests the applicant to provide a brief description of the material, and the AMR results that justified that it does not need aging management during the period of extended operation.

Original AmerGen Response

Permali consists of vacuum impregnated material based on wood veneers (rosewood) and phenolic resin. The material was provided in Oyster Creek original design in combination with steel blocks to provide neutron shielding around recirculation piping nozzles at biological shield wall penetrations. The material is designed for its operating environment and aging management reviews did not identify aging effects requiring aging management during the period of extended operation.

Supplemental Clarification Requested:

In a conference call with NRC Staff on July 7, 2006, the staff noted that the hydrogen content in wood veneers would make the material susceptible to neutron radiation, and high temperatures around the penetration could affect the stability of phenolic resin. AmerGen is requested to provide a detailed justification of its aging management review for this material that resulted in the conclusion that no aging management is required for this material. The Staff requested AmerGen provide material test reports that show the material is capable of performing its intended function through the period of extend operation, or monitor aging effects of the material.

AmerGen Clarification

AmerGen stated during the conference call that the material was provided in the original plant design specifically for shielding purposes around penetrations in the biological shield wall. Industry and plant specific operating experience have not identified any aging effects requiring management. Also, available vendor data, not specific to Oyster Creek, shows that the material is designed for neutron attenuation in a high temperature environment. But it is unlikely that Oyster Creek will be able to produce specific material test reports for the original material.

AmerGen therefore has elected to monitor the "Permali" material associated with the penetration shielding blocks for potential aging effects that could impact their intended function. The blocks will be monitored for loss of material and cracking through the Structures Monitoring aging management program. The inspection frequency will coincide with the ASME Section XI inspection of reactor vessel nozzles, where the material is applied. The revised Structures Monitoring Program UFSAR Supplement (A.1.31) is provided below.

Note: The following is an update to the FSAR Supplement program description for the Structures Monitoring Program, A.1.31. Additions made to the program description as a result the update to the response to RAI 3.5-4 are shown in bold font.

A.1.31 Structures Monitoring Program

The Structures Monitoring Program is an existing program that was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2 and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The program includes elements of the Masonry Wall Program and the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants aging management program.

The program relies on periodic visual inspections to monitor the condition of structures and structural components, structural bolting, component supports, masonry block walls, water-control structures, the Fire Pond Dam, exterior surfaces of mechanical components that are not covered by other programs, and HVAC ducts, damper housings, and HVAC closure bolting. The program relies on procurement controls and installation practices, defined in plant procedures, to ensure that only approved lubricants and proper torque are applied to bolting in scope of the program.

The scope of the program will be enhanced to include structures and structural components that are not currently monitored, but determined to be in the scope of license renewal, including Station Blackout System structures and phase bus enclosure assemblies, Meteorological Tower Structures, submerged structures, component supports not covered by other programs, the Fire Pond Dam, and exterior surfaces of Oyster Creek and Forked River Combustion Turbine mechanical components that are not covered by other programs, including exterior surfaces of HVAC ducts, damper housings, and closure bolting. The inspections will look for leakage from or onto external surfaces, worn, flaking or oxide-coated surfaces, corrosion stains on thermal insulation, and protective coating degradation (cracking and flaking). The program will also be enhanced to require removal of piping and component insulation on a sampling basis to allow visual inspection of insulated surfaces, and to require sampling and testing of groundwater every 4 years to confirm that the soil environment is non-aggressive to below-grade concrete structures. Other program scope enhancements include, but are not limited to, inspection of piping components associated with the Radio Communications system located at the meteorological tower site, and inspection of Reactor Building Closed Loop Cooling, Feedwater, and Main Steam piping located inside the Drywell. The enhancements will be made prior to entering the period of extended operation.

Inspection criteria will be enhanced to provide reasonable assurance that change in material properties, cracking, loss of material, loss of form, reduction or loss of isolation function, reduction in anchor capacity due local degradation, and loss of preload are adequately managed so that the intended functions of structures and components within the scope of the program are maintained consistent with the current licensing basis during the period of extended operation.

Inspection frequency is every four (4) years maximum; except for **Permali shield blocks and** submerged portions of the water-control structures. **Permali shield blocks will be**

inspected on a frequency to coincide with the ASME Section XI inspection of reactor vessel nozzles, where the material is applied. A baseline inspection of submerged water-control structures will be performed prior to entering the period of extended operation. A second inspection will be performed six years after this baseline inspection and a third inspection eight years after the second inspection. After each inspection, an evaluation will be performed to determine if identified degradation warrant more frequent inspections or corrective actions.

The Structures Monitoring Program will be enhanced to include the following specific elements:

- *Buildings, structural components and commodities that are not in scope of* maintenance rule but have been determined to be in the scope of license renewal. These include miscellaneous platforms, flood and secondary containment doors, penetration seals, sump liners, structural seals, and anchors and embedment.

- Component supports, other than those in scope of ASME XI, Subsection IWF.

- Inspection of Oyster Creek external surfaces of mechanical components that are not covered by other programs, HVAC duct, damper housings, and HVAC closure bolting. The scope of this enhancement includes the Reactor Building Closed Cooling Water System carbon steel piping and piping elements located inside the primary containment drywell. Inspection and acceptance criteria of the external surfaces will be the same as those specified for structural steel components and structural bolting.

- The visual inspection of insulated surfaces will require the removal of insulation. Removal of insulation will be on a sampling basis that bounds insulation material type, susceptibility of insulated piping or component material to potential degradations that could result from being in contact with insulation, and system operating temperature.

- Inspection of electrical panels and racks, junction boxes, instrument racks and panels, cable trays, offsite power structural components and their foundations, and anchorage.

- Periodic sampling, testing, and analysis of ground water to confirm that the environment remains non-aggressive for buried reinforced concrete.

- Periodic inspection of components submerged in salt water (Intake Structure and Canal, Dilution structure) and in the water of the fire pond dam, including trash racks at the Intake Structure and Canal.

- Inspection of penetration seals, structural seals, and other elastomers for change in material properties.

- Inspection of vibration isolators, associated with component supports other than those covered by ASME XI, Subsection IWF, for reduction or loss of isolation function.

- The current inspection criteria will be revised to add loss of material, due to corrosion for steel components, and change in material properties, due to leaching of calcium hydroxide and aggressive chemical attack for reinforced concrete. Wooden piles and sheeting will be inspected for loss of material and change in material properties.

- Periodic inspection of the Fire Pond Dam for loss of material and loss of form.

- Inspection of Station Blackout System structures, structural components, and phase bus enclosure assemblies.

- Inspection of Forked River Combustion Turbine power plant external surfaces of mechanical components that are not covered by other programs, HVAC duct, damper housings, and HVAC closure bolting. Inspection and acceptance criteria of the external surfaces will be the same as those specified for structural steel components and structural bolting.

- The program will be enhanced to include inspection of Meteorological Tower Structures. Inspection and acceptance criteria will be the same as those specified for other structures in the scope of the program.

- The program will be enhanced to include inspection of exterior surfaces of piping and piping components associated with the Radio Communications system, located at the meteorological tower site, for loss of material due to corrosion. Inspection and acceptance criteria will be the same as those specified for other external surfaces of mechanical components.

- The program will be enhanced to require visual inspection of external surfaces of mechanical steel components that are not covered by other programs for leakage from or onto external surfaces, worn, flaking, or oxide-coated surfaces, corrosion stains on thermal insulation, and protective coating degradation (cracking and flaking).

- To confirm that there is no significant age related degradation occurring on the external carbon steel surfaces of the main steam system located inside containment, a one-time visual inspection for loss of material due to corrosion will be performed.

- To confirm that there is no significant age related degradation occurring on the external carbon steel surfaces of the feedwater system located inside primary containment drywell, a one-time visual inspection for loss of material due to corrosion will be performed.

- **Permali shield blocks will be visually inspected for loss of material and cracking. Inspection frequency will coincide with the ASME Section XI inspection of reactor vessel nozzles, where the material is applied.**

RAI 3.5-6

The through-wall cracking of Fitzpatrick torus indicates a need for closer examination of the highly restrained and structurally discontinuous areas subjected to operational cyclic loads. The prime aging management program used for managing degradation of the primary containment structure is Subsection IWE (AMP B.1.27). The program is focused towards detecting loss of material. The staff requests the applicant to discuss how the program would detect initiation of such cracking in the Oyster Creek primary containment.

Original AmerGen Response

The Oyster Creek ASME Section XI, Subsection IWE aging management program (B.1.27) is not credited for managing crack initiation and growth. The program is based on visual examinations that may not detect cracking experienced at Fitzpatrick. However crack initiation and growth mechanism experienced at Fitzpatrick is not applicable to Oyster Creek as explained below.

The initial review (2005) of the Fitzpatrick torus leak operating experience determined that the crack was related to design and operating conditions that are not applicable to Oyster Creek. Analysis performed by Fitzpatrick indicated that the most likely cause for the initiation and propagation of the crack was the hydrodynamic loads of the turbine exhaust pipe during HPCI operation coupled with highly restrained condition of the torus shell at the torus column support. The cracking occurred in the heat-affected zone of the lower gusset plate of the ring girder at the torus column support. Fitzpatrick concluded that the crack was initiated by cyclic loading due to condensation oscillation during HPCI operation. The condensation oscillations induced on the torus shell may have been excessive due to lack of a HPCI pipe sparger. The combined operation of the HPCI system and safety relief valve (SRV) discharges during the northeast grid blackout disturbance of August 2003 may have initiated the crack. The HPCI system operated approximately 14.5 hours and SRVs lifted five times over a period of 28 hours following the grid disturbance. Oyster Creek does not have a HPCI system and was not subject to events described above.

Since the initial review, NRC issued Information Notice 2006-01, Torus Cracking in BWR Mark I Containment, on January 12, 2006, to alert licensees of Fitzpatrick condition. Exelon is currently reviewing the impact of the Fitzpatrick experience on Oyster Creek. Corrective actions will be initiated if it is determined the condition described in the information notice are applicable to Oyster Creek.

Supplemental Clarification Requested:

In a conference call with NRC Staff on July 7, 2006, the staff noted that loads similar to those generated by the Fitzpatrick event are also generated during SRV discharges in Mark I containments. The Staff requested AmerGen to provide the results of its current review of the event and an explanation of why SRV discharges won't be a concern for Oyster Creek.

AmerGen Clarification:

AmerGen's final review of the NRC Information Notice 2006-01, "Torus Cracking in BWR Mark I Containment", issued on January 12, 2006 concluded that the torus crack identified by the Fitzpatrick Operating Experience is not applicable to Oyster Creek. The crack was considered

event driven, caused by design configuration of the HPCI discharge line into the torus with no spargers. Oyster Creek does not have a HPCI system or a steam discharge line to the torus with the same design configuration as the Fitzpatrick HPCI system.

The SRV discharges won't be a concern for Oyster Creek because unlike the Fitzpatrick event driven HPCI discharges, Mark I containment SRV discharges into the torus are design basis events evaluated in accordance with the Oyster Creek Plant Unique Analysis Report (PUAR). Oyster Creek has five-safety relief valves (EMRVs) installed in the main steam system. When opened, steam discharge from each EMRV is through piping routed inside the vent lines that enter the torus from penetrations in the vent header. The steam lines are then routed to a Y-Quencher that discharges underwater. The SRV discharge pipes do not penetrate the torus shell directly.

The Y-Quenchers were provided as a part of the Mark I containment hydrodynamics loads assessment to minimize the consequences of loads that result from blowdown of SRV lines into the torus. Components of the torus that are affected by the cyclic loads, due to blowdowns, were analyzed as described in Oyster Creek PUAR for the current term. The analysis was determined to be a TLAA for the period of extended operation and evaluated as described in LRA Section 4.6.1. Thus, the concern with SRV discharge cycles and their impact on the torus have been addressed in the LRA.