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10 CFR 50.90

June 27, 2001
2130-01-20135

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

Subject: Oyster Creek Generating Station (OCGS)
Docket No. 50-219
Facility License No. DPR-16
Technical Specification Change Request No. 286
Replacement Pages

Reference: AmerGen Letter No. 2130-01-20041 dated March 1, 2001, "Technical Specification Change Request No. 286"

The referenced letter requested NRC review and approval of a proposed change to the Appendix A Technical Specifications contained on page 4.5-3. The purpose of this letter is to forward replacement pages associated with the requested change. Enclosed is replacement page 4.5-3, which reflects the requested change to the Technical Specifications and conforms to the marked-up page contained in the Reference. Also enclosed is bases page 4.5-11 reflecting a bases change associated with the requested Technical Specification change.

Should you have any questions or require any additional information please contact the undersigned at 610-765-5516.

Very truly yours,

Paul F. Gaya

for George B. Rombold
Manager, Licensing

Enclosure: Replacement pages for the Oyster Creek Technical Specifications

c: H. J. Miller, Administrator, USNRC Region I
L. A. Dudes, USNRC Senior Resident Inspector, Oyster Creek
H. N. Pastis, USNRC Senior Project Manager, Oyster Creek
File No. 01036

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<u>Enclosure</u>

Oyster Creek Generating Station
Technical Specification Change Request No. 286

Replacement Technical Specification Page 4.5-3

and

Replacement Technical Specification Bases Page 4.5-11

the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel and verifying the isolation time limit is met. Following maintenance, repair or replacement work on the control or power circuit for the valves, the affected component shall be tested to assure it will perform its intended function in the circuit.

3. During each COLD SHUTDOWN, each main steam isolation valve shall be closed and its closure time verified to be within the limits of Specification 4.5.F.1 above unless this test has been performed within the last 92 days.

4. Reactor Building to Suppression Chamber Vacuum Breakers

   a. The reactor building to suppression chamber vacuum breakers and associated instrumentation, including setpoint, shall be checked for proper operation every three months.

   b. During each REFUELING OUTAGE, each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker from closed to fully open does not exceed the force specified in Specification 3.5.A.4.a. The air-operated vacuum breaker instrumentation shall be calibrated during each REFUELING OUTAGE.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic OPERABILITY Tests

      Once every 3 months and following any release of energy which would tend to increase pressure to the suppression chamber, each OPERABLE suppression chamber - drywell vacuum breaker shall be exercised. Operation of position switches, indicators and alarms shall be verified every 3 months by operation of each OPERABLE vacuum breaker.

   b. REFUELING OUTAGE Tests

      (1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

      (2) The suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163, "Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J, "Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements". The Primary Containment Leakage Rate Testing Program conforms with this guidance.

The maximum allowable leakage rate for the primary containment (L_a) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a). As discussed below, P_a for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time be kept to a practical minimum.

Automatic primary containment isolation valves are provided to maintain PRIMARY CONTAINMENT INTEGRITY following the design basis loss-of-coolant accident. Closure times for the automatic primary containment isolation valves are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant accident. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each REFUELING OUTAGE is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). Closure times restrict coolant loss from the circumferential rupture of any of these lines outside primary containment to less than that for a main steam line break (the design basis accident for outside containment line breaks). The minimum time for main steam isolation valve (MSIV) closure of 3 seconds is based on the transient analysis that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time for MSIV closure of 10 seconds is based on the value assumed for the main steam line break dose calculations and restricts coolant loss to prevent uncovering the reactor core. Per ASME Boiler and Pressure Vessel Code, Section XI, the full closure test of the MSIVs during COLD SHUTDOWNs will ensure OPERABILITY and provide assurance that the valves maintain the required closing time. The provision for a minimum of 92 days between the tests ensures that full closure testing is not too frequent. The MSIVs are partially stroked quarterly as part of reactor protection system instrument surveillance testing.