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10CFR50.55a

2130-02-20325
November 8, 2002

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: Proposed Relief Request to the Requirements of 10CFR50.55a Concerning the Third Ten-Year Interval Inservice Inspection Program (TAC No. MB6573)

Dear Sir/Madam:

Attached for your review and approval is a proposed relief request concerning the four (4) hour pressure test, in accordance with 10CFR50.55a, associated with the Third Ten-Year Interval Inservice Inspection (ISI) Program for the Oyster Creek Generating Station (OCGS). The Third Ten-Year Interval Inservice Inspection (ISI) Program concluded on October 14, 2002. The OCGS ISI Program complied with the 1986 Edition of the ASME, Section XI Code.

We request your review and approval by October 14, 2003.

If you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

Michael P. Gallagher

Michael P. Gallagher
Director, Licensing & Regulatory Affairs
Mid-Atlantic Regional Operating Group

Attachment - Oyster Creek Generating Station Proposed Relief Request

cc: H. J. Miller, Administrator, USNRC, Region I (w/attachment)
R. J. Summers, USNRC Senior Resident Inspector, OCGS (w/attachment)
P. Tam, Senior Project Manager, USNRC (w/attachment)
File No. 01042

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ATTACHMENT

OYSTER CREEK GENERATING STATION

PROPOSED RELIEF REQUEST

RR-33

AmerGen Energy Company
Oyster Creek Generating Station
Third 10-Year Interval
Request Relief RR-33

1. COMPONENT IDENTIFICATION

The components affected by this request are in the Class 1 Reactor Coolant Pressure Boundary, and include insulated portions of the Feedwater, Main Steam, Isolation Condenser, Reactor Water Cleanup, Shutdown Cooling, and Reactor Recirculation Systems.

2. APPLICABLE CODE EDITION

The Oyster Creek Generating Station (OCGS) Inservice Inspection (ISI) Program complied with the 1986 Edition of the ASME, Section XI Code. The Third Ten-Year Interval Inservice Inspection (ISI) Program concluded on October 14, 2002. In a safety evaluation report dated May 23, 2002 (letter from R. J. Laufer (NRC) to O. D. Kingsley (Exelon Nuclear)), the NRC granted a 90-day extension to ISI tests performed for the third interval at OCGS.

3. APPLICABLE CODE REQUIREMENT

Relief is requested from ASME Section XI, Paragraph IWA-5213 (d), which requires a hold time of four hours after attaining pressure and temperature prior to inspection of insulated systems. This test may be performed at a reduced pressure in accordance with Code Case 498-1.

4. IMPRACTICALITY OF COMPLIANCE

The one in ten-year hydrostatic test for the Reactor Coolant Pressure Boundary was performed using ASME Code Case N-498-1. This Code Case requires a four (4) hour hold time for insulated piping. The test procedure requires that: 1) test pressure is in the range of 1020 and 1065 psig and, 2) temperature is in the range of 220 and 240 degrees F. The lower temperature bound is based on OCGS's current Technical Specification limit associated with reactor pressure vessel nil-ductility transition temperature, which is 218 degrees F. Given this narrow temperature range for which the test was to be performed, and the relatively short time between shutdown from full power operations and test performance (approximately 19 days), the reactor decay heat load made it impractical to conduct the test sequence (four (4) hour hold time and an approximate one (1) hour inspection time) in a continuous fashion without exceeding the procedurally established maximum temperature of 240 degrees F. The procedurally established maximum temperature of 240 degrees F ensures that the Technical Specification limit of 250 degrees is not exceeded, and ensures adequate time to depressurize and place shutdown cooling into service.

The reactor recirculation pumps were used to increase temperature during the initial test pressurization. Once the test pressure was reached, the temperature continued to increase up to the limit set in the test procedure. Oyster Creek Generating Station (OCGS) was not designed with a high pressure decay heat removal system; therefore, temperature increased such that the temperature limits could not be maintained.

The test procedure requires depressurization to less than 100 psig prior to exceeding 250 degrees F so that the shutdown cooling system can be placed into service to reduce temperature. The cumulative time at test pressure for this portion of the test was 118 minutes (approximately two (2) hours).

Reactor coolant temperature was reduced to the low end of the test band, the shutdown cooling system was secured, and pressure was raised to the test pressure. Based on the heatup rate during this second pressurization (approximately 5.76 degrees per hour), it was determined that the 250 degree limit would be exceeded, and the continuous four (4) hour time and pressurized inspection could not achieved without exceeding the maximum temperature. The VT-2 inspections commenced after a second hold time of approximately 139 minutes (two (2) hours, 19 minutes).

Based on estimated calculation, in order to achieve a continuous hold time of four (4) hours, and the additional one (1) hour for inspection, it would be necessary to extend the outage an additional ten (10) days.

Pursuant to 10 CFR 50.55a(g)(5)(iii), relief is requested on the basis that extending the outage in order for decay heat to be at a level which would not result in exceeding the procedurally established 240 degree F limit is impractical.

In order to avoid the additional days necessary for decay heat to dissipate, major hardware modifications in the form of additional decay heat removal systems would be necessary to remove this heat. Addition of new nuclear systems to cool down primary containment water for the purpose of removing decay heat is impractical considering the cost of the additional nuclear safety systems. As discussed in "NRR Office Instruction" LIC-102, dated July 18, 2002, requiring major plant or hardware modifications is a basis for impracticality.

5. BURDEN CAUSED BY COMPLIANCE

In order to avoid the additional days necessary for decay heat to dissipate, major hardware modifications in the form of additional decay heat removal systems would be necessary. These modifications would cause significant financial burden. Additionally, the time required to reduce decay heat load would significantly extend the refueling outage, with no increase in safety gained. This would cause a significant loss of generation capability.

Existing Technical Specifications require temperature to be maintained within a narrow band between a lower limit of approximately 218 degrees F and an upper limit of 250 degrees F. Given the short duration of this outage and the inability to remove decay heat at the test pressure, this band is insufficient to allow compliance with the code specification for maintaining test pressure for a four (4) hour period prior to inspection. Therefore, it is not only impracticable, but prohibited to comply with the code specification.

6. PROPOSED ALTERNATE AND BASIS FOR USE

The test performed resulted in the following:

1) The pressure was held for a "cumulative" time of greater that four (4) hours,

2) The test temperature was maintained continually for a minimum of four hours,

3) The minimum pressure reached during the test was approximately 45 psig.

Based on the above, the test is deemed to have satisfactorily met the intent of the Code requirements.

The basis for the hold time for insulated systems is to allow any leakage to travel through insulation. The reactor pressure vessel pressure was decreased to allow the temperature to be lowered. Since the time at the lower pressure was minimized, it is judged that any insulation that became wet from leakage remained wet until the inspection. In addition, there was no time during the total test sequence during which the primary system pressure was reduced to atmospheric, nor was the test temperature allowed to go below 220 degrees F. The minimum pressure reached during the test was approximately 45 psig; therefore, leakage was always in the outward direction (from the primary system to the drywell) during the test.

For the systems applicable to this request for relief, there were no repairs, replacements or modifications made to the insulated piping during the outage. The piping material for these systems is composed of carbon steel, stainless steel, and inconel. Ultrasonic and liquid penetrate examinations were performed on the Isolation Condenser, Core Spray, Shutdown Cooling, Reactor Water Cleanup, Reactor Vessel Nozzles, Reactor Recirculation, and Main Steam in accordance with the Inservice Inspection Program for this outage with no recordable flaws identified.

The insulation utilized on insulated systems is a Nukon blanket type insulation. Additionally, a mirror insulation is used on portions of the reactor vessel. There are no relief requests for reduced inspections for these systems.

7. DURATION FOR PROPOSED ALTERNATIVE

The Third Ten-Year Interval Inservice Inspection (ISI) Program concluded on October 14, 2002. However, a 90-day extension was approved to complete required testing for this interval.