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U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
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Washington, D.C. 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 50-219
IPSAR (NUREG-0822) Section 4.12 Design Codes,
Design Criteria and Loading Combinations (SEP Topic III-7B)

References:
1. ASME Code, Section VIII, 1962
2. ASME Code, Section III, 1980
3. GPU Nuclear Calculation No. C-1302-187-5320-007, OC Drywell Penetration Reinforcement
4. GPU Nuclear Calculation No. C-1302-243-5320-042, OC SEP Topic III - 7.B Penetration Fatigue
5. GPU Nuclear Technical Data Report TDR 713 "OCNGS Upper Drywell Shield Wall Thermal Analysis", September 16, 1985

GPU Nuclear letter dated June 4, 1984 submitted results and a description of their evaluation of the subject SEP topic. Subsequently, the NRC Staff and their consultant reviewed the submittal and issued a safety evaluation report on October 29, 1986. In the safety evaluation, the staff concluded that twenty issues out of the twenty-three issues covered by Section 4.12 of NUREG-0822 are considered fully resolved and that the remaining three issues required further evaluation. These issues are:

1) Evaluation of the drywell concrete subject to high temperature and thermal transients,

2) Assessment of differences from current criteria for reinforcement of openings, and

3) Confirmation that cyclic analysis is not required for openings at Oyster Creek in accordance with the code exclusion criteria.

GPU Nuclear has completed evaluations of the three issues. Descriptions and conclusions of the evaluations are provided below:

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1) <u>High Temperature Effect on Drywell</u>

Above elevation 94'-0 in the Oyster Creek Drywell, the shield wall concrete is not insulated from the operating temperatures of the Reactor. ACI 349-80 states that average concrete temperatures under normal operating conditions shall not exceed 150°F. The code does, however, permit higher temperatures provided the resulting reduction in concrete strength is applied to design allowables. An evaluation was therefore conducted to determine the effect of the operating temperatures on the structural integrity of the shield wall concrete.

Based on available information, GPU Nuclear calculated that the temperature at the Reactor Vessel Head could reach 285°F. A finite element, three-dimensional ANSYS computer model was developed to perform a heat transfer calculation and to determine the stresses in the concrete (Ref. 5). This analysis was performed conservatively assuming an upper-bound temperature at the Reactor Vessel Head of 340°F. Based on the heat transfer calculations, average concrete temperatures were determined to reach 180°-280°F. As permitted by ACI 349-80, calculated stresses in the concrete were compared to allowables based on available test data. This analysis shows that even with the reduced allowables, the concrete shield wall is adequate to withstand all design loading conditions. Therefore, the temperatures to which the shield wall is exposed have no impact on the structural integrity of the wall. While cracking may occur, the shield wall retains sufficient capacity to carry all design loads without exceeding allowable capacities.

To ensure that the analysis considered the bounding condition, temperature data from thermocouples installed at the Reactor Vessel Head and on the inside face of the concrete shield wall was reviewed. These thermocouples have shown that the maximum recorded temperature at the Reactor Vessel Head was 294°F. Based on thermocouple data and analytical results, average concrete temperatures have varied from 150°F - 225°F. Therefore, the calculation discussed above and documented in Reference 5 has been determined to be conservative.

Furthermore, tests conducted since ACI 349-80 was issued have demonstrated that strength and deterioration of concrete become significant concerns only at average temperatures in excess of 300°F (Reference: Concrete Technology, D. F. Orchard, Vol. 1, 3rd Edition). Since average temperatures have been measured to be less than 225°F, the operating conditions are not expected to have any impact on the adequacy of the shield wall.

Based on test data, temperature measurements and detailed analytical results, the operating temperatures have been shown to have no impact on the structural integrity of the concrete shield wall. Therefore, the evaluation of the SEP topic associated with high temperature effects on the Drywell concrete wall is complete.

2) Reinforcement of Openings

GPU Nuclear submittal dated June 4, 1984, provided results of a study of the reinforcement existing at Oyster Creek drywell openings. Staff's review concluded that disposition of reinforcement is in accordance with current criteria for some openings but not others. The basis of this conclusion was a Chicago Bridge & Iron (CB&I) Company review of the drywell penetration design performed in 1983 for GPUN which indicated that sixteen penetrations designed per the original design criteria (Ref. 1) did not satisfy the current criteria (Ref. 2). The most common item was the use of partial penetration welds with reinforcing pads. In order to assess the reinforcement and associated details, more detailed reanalysis (Ref. 3) was performed recently to compare the area of reinforcement to the requirements of Reference 2. In the reanalysis, all of the sixteen penetrations were reanalyzed and were found to meet the requirements of Reference 2.

3) Reinforcement of Openings Subject to Cyclic Loadings

Analysis of cyclic loading is required unless the penetrations satisfy the exclusion criteria specified in NE-3221.5(d) of Reference 2. GPU Nuclear's response transmitted by our letter of June 4, 1984 provided our judgement that all openings would satisfy the exclusion rules. The NRC Staff requested an engineering analysis be made. In our subsequent analysis, seventeen penetrations typical of high temperature and pressure fluctuations connected to safety related systems were selected for the analysis. Based on a review of system design conditions data, it was determined that out of seventeen penetrations five will envelope the remaining twelve. These five penetrations selected for the fatigue evaluation are given below:

(1) Isolation Condenser Penetration X-5A/B.
(2) Clean-up Demineralizer Penetration X-10.
(3) Shutdown Cooling Penetration X-7.
(4) Main Steam Penetration X-2A/B.
(5) CRD Scram Discharge Hydraulic Penetration X-3A/B.

All five penetrations are flued heads. A detailed calculation (Reference 4) shows that four penetrations, namely: (a) Clean-up Demineralizer X-10, (b) Shutdown Cooling X-7, (c) Main Steam X-2A/B and (d) CRD Scram Discharge Hydraulic X-3A/B meet the exclusion requirements of Reference 2.

A fatigue analysis (Reference 4) performed for the Isolation Condenser Penetration X-5A/B indicates that it also meets the requirement of Reference 2.

As described above, it is concluded that the three remaining issues meet the requirements of recent criteria. If you have any question concerning the above responses, please contact M. W. Laggart, Manager, BWR Licensing, at (201) 316-7968.

Very truly yours,

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