UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 1, 1995

MEMORANDUM TO: Philip F. McKee, Project Director
Project Directorate I-3,
Division of Reactor Projects I/II

FROM: Richard H. Wessman, Chief
Mechanical Engineering Branch
Division of Engineering, NRR

Jack R. Strosnider, Chief
Materials and Chemical Engineering Branch
Division of Engineering, NRR

SUBJECT: PROPOSED RESPONSE TO NUCLEAR INFORMATION & RESOURCE SERVICE LETTER OF OCTOBER 16, 1995, REGARDING THE ADEQUACY OF THE OYSTER CREEK CORE SHROUD BRACKETS (TAC NO. M93857)

On October 19, 1995, the staff of Project Directorate I-3 forwarded a Work Request to the Mechanical Engineering Branch (EMEB) requesting that the EMEB staff review a letter from the Nuclear Information and Resource Service (NIRS), dated October 16, 1995. The content of this letter provided the NRC with four questions regarding the structural integrity of the Oyster Creek Nuclear Generation Station (OCNGS) core shroud modification (repair). The staff of Project Directorate I-3 also requested that EMEB, with support from the Reactor Systems Branch (SXRB) and the Materials and Chemical Engineering Branch (EMCB), provide a response to the letter from NIRS.

The EMEB, EMCB and SXRB staffs have reviewed the letter of October 16, 1995, from NIRS, the content of applicable licensing documentation from General Public Utilities (GPU, the licensee for OCNGS), and the content of applicable NRC SERs and letters that have been issued in regard to the structural integrity of the OGNGS reactor internals, and has determined that GPU has taken appropriate measures to provide adequate assurance that the structural integrity of the OCNGS core shroud brackets will be maintained during subsequent operating cycles, and to ensure the safety of the OCNGS reactor.

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The following attachment provides the staff's proposed reply to Mr. Gunter, Director of the Nuclear Watchdog Group, who is representing NIRS in this matter.

Docket No.: 50-219

Attachment: As stated

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NAME	JMedoff:jm:efh		JRRajan		JRStronsnider		RHWessman	
DATE	10/30/95:jb		10/30/95		11/2/95		11/1/95	

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cc: B. W. Sheron
G. C. Lainas
A. W. Dromerick

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Mr. Paul Gunter
Director - Reactor Watchdog Project
Nuclear Information and Resource Service
1424 16th Street, N.W.
Washington, DC 20036

SUBJECT: RESPONSE TO YOUR LETTER OF OCTOBER 16, 1995, TO THE NRC REGARDING THE SAFETY EVALUATION OF THE CORE SHROUD REPAIR AT OYSTER CREEK NUCLEAR GENERATION STATION

Dear Mr. Gunter:

This letter is to acknowledge receipt of your correspondence of October 16, 1995 regarding the operational safety of the Oyster Creek Nuclear Generation Station (OC). In your correspondence of October 16, 1995, you stated that NIRS had specific concerns in regard to the design of the OC core shroud modification (repair) design. Specifically, these concerns focused "on the Shroud Restraint Lower Hook depicted in Figure 4 of the NRC Safety Evaluation dated November 25, 1994." More specifically, you contended the following points:

1. "That the lower hooks for the restraint assemblies are dependent on the originally installed gussets, which in turn are composed of plates of Type 304 stainless steel and Inconel 600."

2. "That the anchor system utilized for the core shroud is dependent upon a <u>susceptible material</u> which has been subject to long-term exposure by the same harsh operational environment that incubated cracking of the core shroud."

3. "That both the NRC and the licensee ignored information contained within NUREG/CR-5754 which warns that it is '<u>essential</u> to control or eliminate stressors'" and that "to the contrary, GPUN incorporated 'a susceptible material' in the fabrication of the anchor system for the core shroud fix, namely, Type 304 stainless steel and Inconel 600 which have been subjected to aging conditions for 25 years during the reactor's operation."

In your correspondence of October 16, 1995, you requested "a response from the NRC regarding the Safety Evaluation [of November 25, 1995] and proposed inspections as they pertain to the alleged incorporation of a susceptible material into the Shroud Restraint Assembly." More specifically, you requested that the NRC provide responses to the following questions:

1. "Has the NRC evaluated or reviewed the licensee's evaluation of the effects of stress corrosion cracking (SCC), fatigue, erosion, embrittlement, and creep of the plate material and its Heat Affected Zones (HAZ) which fabricates the gussets? Is that documentation available for public review?"

ATTACHMENT

2. "Given that a susceptible material has already been incorporated into the Shroud Restraint Assemblies, what has the licensee done to mitigate the other two conditions necessary to incubate SCC as described in NUREG/CR-5754?"

3. "Where has the NRC or the licensee evaluated the consequences of the failure of one or more Type 304 stainless steel gussets used to anchor the Shroud Restraint Assemblies?"

4. "Does the NRC plan to require frequent enhanced inspections of the tie rod anchor plates in question? If so what does the NRC consider to constitute frequent and enhanced inspections for age related deterioration.? If not, why not?"

The staff's responses to your questions are provided below. These responses are given in a specifc "Question" and "Response" format.

Question 1: Has the NRC evaluated or reviewed the licensee's evaluation of the effects of stress corrosion cracking (SCC), fatigue, erosion, embrittlement, and creep of the plate material and its Heat Affected Zones (HAZ) which fabricates the gussets? Is that documentation available for public review?

Response: The NRC has reviewed the licensee's evaluation of the effects of IGSCC, fatigue, erosion, embrittlement, and creep on the gusset (bracket) materials and welds. Of these age-related degradation mechanisms, only IGSCC has the potential for being of concern with respect to the OC core shroud brackets. These brackets at OC consist of lugs welded to the shroud support cone (fabricated from ASTM B186, Inconel 600, condition A1 or A2 descaled material), clevises welded to the core shroud (fabricated from ASTM A240 Type 304 stainless steel), and pins joining the lugs and clevises together (fabricated from ASTM A276 Type 304L stainless steel). The Type 304 materials for the clevises and Type 304L materials for the pins were all given solution annealing treatments as part of the manufacturing process in order to improve the corrosion resistance of the fabricated pieces. The clevises and pins have maximum carbon contents of 0.08% and 0.03% respectively. The lower carbon content of the pins should make them less susceptible to IGSCC than the clevis components. The Inconel 600 materials were also heat treated per ASTM specification B162. The materials for the brackets were selected for compatibility with each other, the reactor environment, and with the materials at the welded points of attachment. However, the potential for the occurrence of IGSCC in the welds that join the clevises to the core shroud support or the lugs to the shroud support cone cannot absolutely be precluded. It should be emphasized, however, that the brackets were originally added to the OC core shroud design to provide added structural support for the H7 and H8 shroud welds. These gussets were installed into the OC plant design prior to initial operation, prior to the staff's issuance of NUREG/CR-5754, and prior to the licensee's submittal of the OC core shroud repair design.
GPU has undertaken an active examination effort to determine whether or not any IGSCC degradation of these brackets had occurred. GPU performed enhanced VT-1 examinations of the 30 shroud brackets during the October/November 1994

refueling outage (RFO) for the OC reactor; these additional examinations were included as part of GPU's inspections for the OC core shroud. Of the VT-1 examinations performed on the 30 brackets, only the examination of the No. 2 bracket revealed a relevant, but minor indication. This indication was evaluated and determined to be acceptable for further service. These inspection results are documented in a letter from Mr. R. W. Keaten, Vice President, GPU Nuclear Corporation, to the NRC, dated November 3, 1994. The staff's assessment of the licensee's core shroud inspection results were included as part of its evaluation of GPU's response to Generic Letter (GL) 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors." This evaluation is provided in the staff's Safety Evaluation Report (SER) to Mr. R. W. Keaten, Vice President and Director of Technical Functions, GPU Nuclear Corporation, dated February 23, 1995. These documents are available for review at the NRC's Public Document Room, Gelman Building, 2120 L Street, NW, Washington, DC. Further discussion of the relevance of these inspections to the adequacy of the OC core shroud repair design is provided in our response to your question regarding the consequences of a gusset failure (third question).

Question 2: Given that a susceptible material has already been incorporated into the Shroud Restraint Assemblies, what has the licensee done to mitigate the other two conditions necessary to incubate SCC as described in NUREG/CR-5754?

Response: GPU has maintained better control of chemistry conditions in the OC reactor coolant since Cycle No. 12, commencing May 1989. Specifically, GPU installed a hydrogen water chemistry system at OC after Cycle No. 12. The implementation of hydrogen water chemistry at OC has enabled GPU to lower the oxygen content of the OC reactor coolant, and therefore lower the operating conductivity of the OC reactor coolant. The improvements in the reactor coolant dissolved oxygen content and operating reactor coolant conductivity have been shown to serve to mitigate the environmental conditions that might contribute to IGSCC of the core shroud brackets and other vessel internal components at OC.

Question 3: Where has NRC or the licensee evaluated the consequences of the failure of one or more Type 304 stainless steel gussets used to anchor the Shroud Restraint Assemblies?

Response: As stated in the NRC's "Safety Evaluation Regarding the Oyster Creek Core Shroud Repair," dated November 25, 1994, the gusset assemblies and their welds at the H_7 and H_8 locations were inspected by the licensee and found satisfactory. Hence, the use of these assemblies for the lower attachment point is acceptable. These inspection results are documented in a letter dated November 3, 1994, from Mr. R. W. Keaton to the USNRC Document Control Desk.

For all practical purposes, failure of a gusset assembly used for tie rod attachment during normal operations results in failure of the associated tie rod assembly (i.e., tie rod assembly becomes detached from its attachment point). GPU analyzed the consequences of tie rod assembly becoming detached during normal operations and concluded that the impact on the shroud or other tie rod assemblies due to one tie rod assembly failure is of minimal safety significance. The potential exists that a failure of one tie rod assembly could slightly increase the postulated shroud crack leakage. GPU conservatively modeled the crack to provide 0.001 inch leakage path per weld. GPU estimated that the total leakage from all welds, H_1 through H_{6B}, having postulated 360° through-wall cracks was approximately 60 gpm. The staff reviewed GPU's analysis and concluded that the slight increase in shroud crack leakage due to the failure of one tie rod assembly is bounded by this analysis and is acceptable. Additionally, the staff concluded that failure of more than one tie rod assembly (or gusset) is not likely based on the inspection results of the gussets.

If individual components should somehow break off the tie-rod assembly, they will fall into the "V" shaped section at the bottom conical support plate or if small enough, could be transported into the recirculation loop and its pump. The consequences of a loose tie-rod component are no different than those from other loose parts from the reactor internals within the recirculation system. Also, the NRC has determined that there is no safety concern requiring monitoring for loose parts in the reactor system for Oyster Creek.

GPU's analysis is documented in "GPU Nuclear's Safety Evaluation, SE-403037-001, Rev. 0, for the Reactor Vessel Core Shroud 15R Outage Enhancement," dated November 9, 1994. The documents discussed above are available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

Question 4: Does the NRC plan to require frequent enhanced inspections of the tie rod anchor plates in question? If so what does the NRC consider to constitute frequent and enhanced inspections for age related deterioration? If not, why not?

Response: The NRC has requested that licensees commit to augmented inspections of repair assemblies as a contingent condition of receiving any approval of core shroud repair designs. GPU has committed to performing augmented inspections of a percentage of the shroud brackets (gussets) during subsequent operating RFOs. These inspections will be included as part of GPU's augmented inspection program for the OC tie rod-assemblies, and will serve the purpose of determining whether or not any additional or further degradation is likely to occur in the OC core shroud brackets during subsequent operating cycles. This augmented inspection program has been submitted to the staff for review. The staff is still awaiting the BWRVIP's generic guidelines for performing augmented inspections of repair assemblies and for performing re-inspections of core shrouds. The staff will review the augmented inspection program submitted by GPU pending a review of the awaited guidelines from the BWRVIP. What the NRC considers acceptable in terms of proposed augmented inspection methods and frequencies will be based on the results of the staff's review of the BWRVIP's augmented inspection proposals.

Please note that the NRC will continue to take regulatory action on a plant-specific or generic basis as may be appropriate when age related degradation issues are identified. I hope that this information answers your questions to the staff.

Sincerely,

Phillip F. McKee, Project Director
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation