

JAN 30 2004

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

**INSERVICE INSPECTION PROGRAM
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NOS. 50-354**

**SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RESPONSE -
HOPE CREEK GENERATING STATION RELIEF REQUEST HC-RR-B12 (TAC
NO. MB8407)**

By letter dated April 14, 2003, PSEG Nuclear LLC (PSEG) submitted a request for relief from the required reactor pressure vessel volumetric examination required by the American Society of Mechanical Engineers Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100. The relief was requested pursuant to Title 10 of Code of Federal Regulations Section 50.55e(a)(3)(ii).

The Nuclear Regulatory Commission (NRC) requested additional information be provided in response to their June 26, 2003 letter. By letter dated September 3, 2003 PSEG submitted the requested additional information.

On November 19, 2003 PSEG was contacted by the Hope Creek Project Manager regarding some additional requests for information. These were discussed with the NRC staff on December 18, 2003. The following is being submitted in response to the additional request for information.

Should you have any additional questions, please contact Mr. Michael Mosier at 856-339-5434.

Sincerely,

A handwritten signature in black ink, appearing to read "Steven Mannon".

Steven Mannon
Manager – Nuclear Safety and Licensing

AC07

JAN 30 2004

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REQUEST FOR ADDITIONAL INFORMATION
HOPE CREEK GENERATING STATION

By letter dated April 14, 2003, PSEG Nuclear LLC (PSEG) submitted a request for relief from the required volumetric examination required by the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.100. The relief was requested pursuant to Title 10 of Code of Federal Regulations Section 50.55e(a)(3)(ii).

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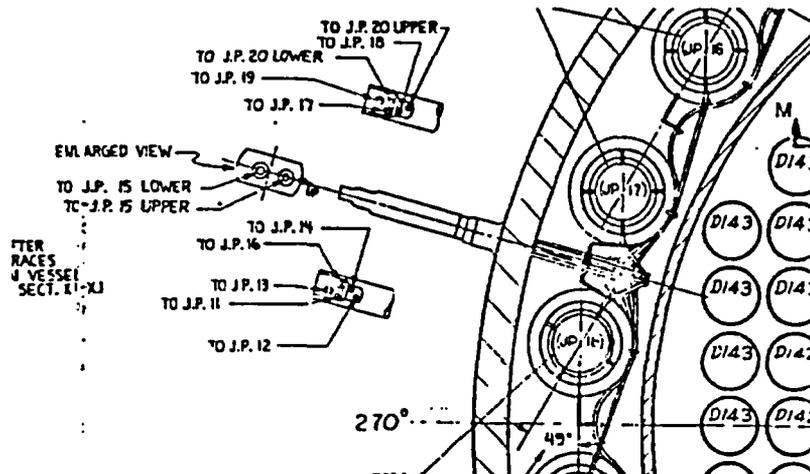
On November 19, 2003 PSEG was contacted by the Hope Creek Project Manager regarding some additional requests for information. These were discussed with the NRC staff on December 18, 2003. The following is being submitted in response to the additional request for information.

- 1. The April 14, 2003 submittal provided a precedence (Fermi Unit 2) for their relief request. The precedence contained a request for relief from UT to enhanced VT-1 with a 100% coverage and another request for relief from UT to enhanced VT-1 with less than 100% coverage. The staff's safety evaluation considered the coverage and sample size from the request with 100% coverage when reviewing the request of less than 100% coverage. Identify other RPV inner nozzle radii that were essentially 100% examined with enhanced VT-1 (referenced previous safety evaluation). What percent of coverage were you able to achieve during the first interval using UT?**

During Hope Creek's first inspection interval the reactor pressure vessel (RPV) inner nozzle radii were essentially 100% examined using manual UT at the cost of significant personnel exposure. Hope Creek performs automated UT of the Feedwater nozzles and expects to continue doing so per the requirements of NUREG-0619.

The remainder of the RPV nozzles inner radii receives a manual UT from the reactor vessel's external shell surface and nozzle bore regions. In PSEG's April 14, 2003 request, it was indicated that dose rates for specified RPV nozzles were in the range of 200 mR/hr to 250 mR/hr with shielding in place. During Hope Creek's RFO11 (Spring 2003), several of the RPV nozzles exhibited significantly elevated dose rates. The nozzles average dose rates ranged from 250 - 300 mR (nozzle N2) to approximately 9.0 R (nozzle N17) on contact. The manual UT exams are performed from within each nozzle's door opening and contribute to higher personnel radiation exposures due to the examiner's close proximity to the

The sketch below demonstrates the N8A and N8B typical reactor vessel jet pump instrumentation nozzle design that retains the twelve (12) jet pump instrument tubes as shown emanating from the nozzle as shown.



3. In the September 3, 2003 response, Question 3 describes a demonstration process of an automated visual examination system but does not address direct visual examinations. Explain the qualification process for direct visual examinations that will be used to demonstrate the 1-mil width sensitivity. Include a discussion on the application of the qualified process.

On page 1 of 4 of relief request HC-RR-B12, the statement: "Reactor vessel closure head vent and spray nozzles inner radii will receive direct visual examinations (VT-1) conducted in accordance with ASME Section XI requirements, while the other remaining aforementioned components will receive enhanced visual examinations using the 1-mil wire diameter wire standard", was an inadvertent error. Direct visual examinations are only intended to be performed upon the reactor vessel closure head's spray and vent penetrations as stated in relief request HC-RR-B11 that was previously approved by the NRC on June 9, 2003 (TAC No. MB8408).