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PSEG
Nuclear LLC

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

INSERVICE INSPECTION PROGRAM
RELIEF REQUEST HC-RR-B12
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSES NPF-57
DOCKET NOS. 50-354

Pursuant to 10 CFR 50.55a(a)(3), PSEG Nuclear, LLC (PSEG Nuclear) requests approval of the enclosed relief request. Approval for relief is requested in accordance with the hardship or unusual difficulty provisions of 10CFR50.55a(a)(3)(ii). In this instance, visual examination of the inner radius is limited by physical obstructions. PSEG Nuclear wishes to use the proposed alternative to the required volumetric examination for nozzles in lieu of the existing ASME Section XI Table IWB-2500-1, Examination Category B-D, Item B3.100. Compliance with the proposed alternatives will provide an adequate level of quality and safety for examination of the affected areas.

The attachment to this letter includes the proposed alternative and supporting justification for the relief. Based on the evaluation contained in the attachment, PSEG Nuclear has concluded that the proposed alternative provides an acceptable level of quality and safety. Accordingly, this proposal satisfies the requirements of 10CFR50.55a(a)(3)(ii).

This relief request is applicable to PSEG Nuclear Hope Creek Generating Station. PSEG Nuclear requests that the NRC approve this request by May 2003 in order to support Hope Creek refueling outage RFO11 scheduled to commence April 12, 2003.

Should you have any questions regarding this request, please contact Mr. Howard Berrick at 856-339-1862.

Sincerely,

G. Salamon
Manager – Nuclear Safety and Licensing

Attachment: ISI Relief Request HC-RR-B12

A047

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ASME Code Component Affected

Alternative Exam Requirements for Inner Radius Examination of Class 1 Reactor Pressure Vessel (RPV) Nozzles. [See Table 1 below]

Applicable ASME Code Edition and Addenda

ASME Section XI, 1989 Edition, is the code of record for PSEG Nuclear LLC (PSEG Nuclear) Hope Creek Nuclear Generating Station's Second Ten-Year ISI Program Interval.

Applicable Code Requirement

Conduct volumetric examinations of Hope Creek Nuclear Generating Station RPV Nozzle Inside Radius Sections in accordance with ASME Section XI 1989 Edition IWB-2500-1 requirements for Class 1 Examination Category B-D, Item B3.100, Figures IWB-2500-7 (a) through (d).

Proposed Alternative

Pursuant to 10 CFR 50.55a(a)(3)(ii), PSEG Nuclear LLC proposes to perform an enhanced VT-1 visual examination technique of the surface M-N shown in ASME Section XI Figures IWB-2500-7 (a) through (d) as an alternative to ASME Section XI Table IWB-2500-1, Examination Category B-D, Item B3.100 requiring volumetric examination (Ultrasonic, UT) of the Inner Radius of Class 1 Reactor Vessel Nozzles.

The enhanced remote visual examination will be performed upon the examination surface M-N to achieve essentially 40-60% coverage using 8x magnification video equipment to examine the inner radii. The resolution sensitivity for this remote examination will be established using a 1-mil diameter wire standard similar to that used for other reactor pressure vessel internal examinations intended to detect cracking.

Reactor vessel closure head vent and spray nozzles inner radii will receive direct visual examinations (VT-1) conducted in accordance with ASME XI requirements, while the other remaining aforementioned components will receive enhanced visual examinations using the 1-mil diameter wire standard.

Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-3142.3 or IWB-3142.4.

Basis for Relief

The following Hope Creek RPV Nozzle Inner radius exams listed below contain configurations that impede complete 100 percent visual examination coverage of the nozzle inner radius area surface M-N.

Table 1			
Hope Creek RPV Nozzle Inner Radius Exams			
Summary Number	**Component Identification**	**Estimated Coverage**	**Limitation Configuration**
100195	RPV1-N2A	50%	Thermal Sleeve/Jet Pump Riser 30° Recirculation Inlet Nozzle
100200	RPV1-N2B	50%	Thermal Sleeve/Jet Pump Riser 60° Recirculation Inlet Nozzle
100205	RPV1-N2C	50%	Thermal Sleeve/Jet Pump Riser 90° Recirculation Inlet Nozzle
100210	RPV1-N2D	50%	Thermal Sleeve/Jet Pump Riser 120° Recirculation Inlet Nozzle
100215	RPV1-N2E	50%	Thermal Sleeve/Jet Pump Riser 150° Recirculation Inlet Nozzle
100220	RPV1-N2F	50%	Thermal Sleeve/Jet Pump Riser 210° Recirculation Inlet Nozzle
100225	RPV1-N2G	50%	Thermal Sleeve/Jet Pump Riser 240° Recirculation Inlet Nozzle
100230	RPV1-N2H	50%	Thermal Sleeve/Jet Pump Riser 270° Recirculation Inlet Nozzle
100235	RPV1-N2J	50%	Thermal Sleeve/Jet Pump Riser 300° Recirculation Inlet Nozzle
100240	RPV1-N2K	50%	Thermal Sleeve/Jet Pump Riser 330° Recirculation Inlet Nozzle
100295	RPV1-N5A	40%	Thermal Sleeve/and Sparger 120° Core Spray Inlet Nozzle
100300	RPV1-N5B	40%	Thermal Sleeve/and Sparger 240° Core Spray Inlet Nozzle
100320	RPV1-N8A	60%	Instrumentation Lines 112.5°- Jet Pump Instrumentation Nozzle
100325	RPV1-N8B	60%	Instrumentation Lines 292.5°- Jet Pump Instrumentation Nozzle
100400	RPV1-N17A	50%	Thermal Sleeve/ Collar & Bolt Assembly 45° LPCI Inlet Nozzle
100401	RPV1-N17B	50%	Thermal Sleeve/ Collar & Bolt Assembly 135° LPCI Inlet Nozzle
100402	RPV1-N17C	50%	Thermal Sleeve/ Collar & Bolt Assembly 225° LPCI Inlet Nozzle
100403	RPV1-N17D	50%	Thermal Sleeve/ Collar & Bolt Assembly 315° LPCI Inlet Nozzle

Visual examination of the inner radius region for the subject nozzles is limited because reactor internal piping configuration prevents placement of the camera in all positions necessary to examine surface M-N over the full circumference. The specific nozzle limitations and estimated coverage are indicated in Table 1.

Performance of the volumetric examination results in significant personnel radiation exposure and does not result in a significant increase in the level of plant quality or safety. The redesign and construction of these nozzles and associated thermal sleeves would result in a hardship and unusual difficulty without compensating increase in level of quality or safety.

Performance of the volumetric examination requires the examiner to enter and remain inside the biological shield penetration area around the nozzle for the duration of the ultrasonic examination that takes approximately 1.5 hours. Dose rates for the specified RPV nozzles are in the range of 200 mr/hr to 250 mr/hr with shielding in place. Performance of these examinations results in an estimated personnel exposure of about 3.5 Rem per inspection interval.

Performance of a visual examination using remote cameras essentially eliminates any personnel exposure. The implementation of this relief request should reduce vessel examination time by approximately 18 hours, which translates to significantly reduced personnel radiation exposure and cost savings.

All nozzle forgings were nondestructively examined during fabrication and have been previously examined using ultrasonic techniques specific to the nozzle configuration. No indications of fabrication or service related cracking have been observed as result of these exams.

Nozzle inner radius examinations are the only non-welded areas requiring examination on the RPV. This requirement was deterministically made early in the development of ASME Section XI, and applied to 100 percent of nozzles welded with full penetration welds. Fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region. For all nozzles other than feedwater, there is no significant thermal cycling during operation. Therefore from a risk perspective there is no need to perform volumetric examination on any nozzles other than feedwater or operational Control Rod Drive (CRD) returns. No service related cracking has been discovered in any of the Boiling Water Reactor (BWR) fleet plant nozzles other than feedwater and operational CRD returns. The six feedwater nozzle inner radius sections will continue to be examined in accordance with UT techniques developed and qualified with GE-NE-523-A71-0594-A Revision 1 (the NRC has approved this report under TAC No. MA6787). PSEG Nuclear believes that application of a visual examination alternative for the listed nozzle inner radius regions ensures an acceptable level of quality and safety.

In an internal Nuclear Regulatory Commission (NRC) memorandum (Reference No. 1), the staff indicated that an ultrasonic examination could be replaced by a VT-1 visual examination of the proposed nozzle inspections on the basis that the surveillance is being maintained and a VT-1 visual examination is completed.

Note: For Table IWB-3512-1, the depth of a crack indication is assumed to be one half of the measured length of the crack indication. As previously stated, crack-like surface flaws found exceeding the acceptance criteria of Table IWB-3512-1 are unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-3142.3 or IWB-3142.4.

Based on the above, PSEG Nuclear believes that the proposed alternative to perform an enhanced VT-1, visual examination technique of the surface M-N shown in ASME Section XI Figures IWB-2500-7 (a) through (d) in lieu of ASME Section XI Table IWB-2500-1, Examination Category B-D, Item B3.100 requiring volumetric examination (Ultrasonic - UT) of the Inner Radius of Class 1 Reactor Vessel Nozzles provides reasonable assurance of structural integrity.

Duration of Proposed Alternative

Hope Creek - Second Ten-Year Interval (ASME XI, 1989 Edition)

Precedence

Previous relief has been granted to Detroit Edison Company, Fermi Unit 2, Docket No. 50-341, TAC No. MB2166 and MB2755 dated October 5, 2001].

References

1. NRC internal memorandum from K.R. Wichman (NRC) to W.H. Bateman (NRC) dated May 25, 2000; Subject: `The Third Meeting with the Industry to discuss the elimination of RPV Inner Radius Inspection (ML003718630).

2. Code Case N-648, Alternative Requirements for Inner Radius Examinations of Class 1 Reactor Vessel Nozzles Section XI, Division 1.