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

**INSERVICE INSPECTION PROGRAM
RELIEF REQUEST HC-RR-B11
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSES NPF-57
DOCKET NOS. 50-354**

Pursuant to 10CFR50.55a(a)(3), PSEG Nuclear, LLC (PSEG Nuclear) requests approval of the enclosed relief request. Approval for relief is requested in accordance with the alternative examination provisions of 10CFR50.55a(a)(3)(i). PSEG Nuclear proposes to use an alternative to the required volumetric examination for nozzles where plant configuration is such that visual examination of the inner radius may be performed on essentially 100 percent of the inner radius 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 10 CFR 50.55a(a)(3)(i).

This relief request is applicable to PSEG Nuclear Hope Creek Generating Station. PSEG Nuclear requests that the NRC approve this request by April 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

A047

Attachment:
ISI Relief Request HC-RR-B11

C Mr. H. Miller, Administrator
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ASME Code Component Affected

Alternative Exam Requirements for Inner Radius Examination of Class 1 Reactor Pressure Vessel 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 Requirements

Conduct ultrasonic examinations of Hope Creek Nuclear Generating Station Reactor Pressure Vessel (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)(i), approval is requested to use the proposed alternative to the required volumetric examination for nozzles where plant configuration is such that visual examination of the inner radius may be performed on essentially 100 percent of the inner radius 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.

PSEG Nuclear 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 100% 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.

Examinations proposed would be performed during the following refueling outages RFO11 (Spring 2003), RFO12 (Fall 2004), and RFO13 (Spring 2005).

Basis for Relief

The following Hope Creek RPV Nozzle Inner radius exams listed below do not contain configurations that would impede visual examination of the nozzle inner radius area surface M-N.

Table 1

Hope Creek RPV Nozzle Inner Radius Exams

Summary Number	Examination Area Identification	Configuration
100408	RPV1-NIAIR	0°- Recirculation Outlet Nozzle
100409	RPV1-NIBIR	180°- Recirculation Outlet Nozzle
100460	RPV1-N3AIR	72°-Main Steam Nozzle
100465	RPV1-N3BAIR	108°-Main Steam Nozzle
100470	RPV1-N3CAIR	252°-Main Steam Nozzle
100475	RPV1-N3DAIR	288°-Main Steam Nozzle
100520	RPV1-N6AIR	Spray Head Nozzle
100525	RPV1-N6BIR	Spare Spray Head Nozzle
100530	RPV1-N7IR	Head Vent Nozzle
100330	RPV1-N9A	Capped CRD Hydraulic Return Nozzle

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 CRD returns.

No service related cracking has been discovered in any of the BWR (boiling water reactor) 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.

According to the 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.

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.

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.

Duration of Proposed Alternative

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

Precedence

Previous relief has been granted to Detroit Edison, Fermi Unit 2 [NRC Safety Evaluation Reports 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.