

MAY 09 2003



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United States Nuclear Regulatory Commission
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**INSERVICE INSPECTION PROGRAM
REVISION TO RELIEF REQUEST HC-RR-B11
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSES NPF-57
DOCKET NOS. 50-354**

On April 14, 2003, PSEG Nuclear requested relief (HC-RR-B11) from the requirements of ASME Section XI 1989 Edition, Table IWB-2500-1, Examination Category B-D, Item B3.100, in order to implement the alternative requirements of 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 that required volumetric examination (Ultrasonic, UT) of the Inner Radius of Class 1 Reactor Vessel Nozzles. The request for relief is for the second 10-year inservice inspection (ISI) interval.

During a teleconference on May 5, 2003, PSEG provided a response to the NRC Staff's question concerning the relief request. As a result of this telecon, PSEG is submitting a revision to relief request HC-RR-B11 for the NRC Staff's action. The relief request contained in this submittal replaces the relief request submitted on April 14, 2003 in its entirety.

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 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).

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Should you have any questions regarding this request, please contact Mr. Brian Thomas at 856-339-2022.

Sincerely,



G. Salamon
Manager – Nuclear Safety and Licensing

Attachment:
ISI Relief Request HC-RR-B11

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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 (EVT-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 (EVT-1) 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.

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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

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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].

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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.

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-- Alternative Provides Acceptable Level of Quality and Safety --