PSEG

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Nuclear Business Unit

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

RELIEF REQUEST FOR FIRST TEN YEAR INTERVAL INSERVICE INSPECTION PROGRAM REGARDING AUGMENTED REACTOR VESSEL INSPECTION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Gentlemen:

Public Service Electric and Gas Company (PSE&G) requests approval of the attached Relief Request RR-A4 to the Hope Creek Generating Station First Ten Year Inservice Inspection (ISI) Program. Relief Request RR-A4 requests relief from the reactor pressure vessel (RPV) examination requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI, 1983 Edition, and the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2). Specifically, pursuant to provisions of 10 CFR 50.55a(a)(3)(i), and consistent with information contained in NRC Information Notice 97-63, relief is requested from the examination of the RPV circumferential shell welds for two operating cycles.

NRC Information Notice 97-63 stated that the NRC would consider technically justified requests for relief from BWR licensees scheduled to perform the inspections in Fall 1997 and Spring 1998. PSE&G currently plans to perform the augmented inspections during Hope Creek's Fall 1997 refueling outage.

The details and justification for relief are explained further in the attached Relief Request. PSE&G is requesting NRC review and approval of the Relief Request by September 13, 1997.

Should there be any questions concerning this submittal, please do not hesitate to contact us.

Sincerely,

David R. Powell

David R. Powell
Director -
Licensing & Regulation/Fuels

Attachment: Relief Request No. RR-A4

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HOPE CREEK GENERATING STATION
ISI PROGRAM
RELIEF REQUEST NO. RR-A4

COMPONENT DESCRIPTION

Class 1, Category B-A, Item No. B1.11 reactor pressure vessel pressure retaining circumferential shell welds as identified in the following table:

Reference No.	Description
100010	W-4 Shell to Shell
100015	W-5 Shell to Shell
100020	W-6 Shell to Shell
100025	W-7 Shell to Shell

ASME EXAMINATION REQUIREMENT

In accordance with Table IWB-2500-1 of both the 1983 Edition through Summer 1983 Addenda of Section XI (Hope Creek's current inservice inspection Code of record) and the 1989 Edition of Section XI (as specified in 10CFR50.55a(g)(6)(ii)(A)(2)), all reactor pressure vessel pressure retaining circumferential shell welds are to be volumetrically examined, for essentially 100% of the weld length, by the end of the 1st Inspection Interval. The last refueling outage for Hope Creek's first inspection interval is RFO7.

BASIS FOR RELIEF

The basis for this request for inspection relief is documented in the report "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," that was transmitted to the NRC in September 1995. The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR RPV circumferential shell welds. The BWRVIP-05 report concludes that the probability of failure of the BWR RPV circumferential shell welds is orders of magnitude lower than that of the axial shell welds. The NRC staff conducted an independent risk-informed assessment of the analysis contained in BWRVIP-05. The independent assessment was transmitted to the BWRVIP by letter dated August 14, 1997. This assessment also concluded that the probability of failure of the BWR RPV circumferential welds is orders of magnitude lower than that of the axial shell welds. Additionally, the NRC assessment demonstrated that inspection of BWR RPV circumferential welds does not measurably affect the probability of failure.

This independent NRC assessment included a probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are: the neutron fluence was that estimated to be end-of-license mean fluence; the chemistry values are mean values based on vessel types; and the potential for beyond design basis events is considered.

Although BWRVIP-05 provides the technical basis supporting the relief request, the following information is provided to show the conservatisms of the NRC analysis for Hope Creek. Since Hope Creek's vessel was the only one fabricated by Hitachi and the NRC independent assessment stated in section 2.1.4.1. that it is comparable to a Chicago Bridge & Iron (CB&I) vessel, Hope Creek will use the CB&I as its reference case model. In the case of Hope Creek, the circumferential weld joint between shells 4 and 5 of the RPV would be the limiting circumferential weld within the vessel (i.e. relative to RT_{NDT}). For plants fabricated by or similar to CB&I, the mean end-of-license neutron fluence used in the NRC PFM analysis was 0.19E+19 n/cm^2. However, at Hope Creek, the highest fluence anticipated at the end of the requested relief period is 0.0268E+19 n/cm^2 at the vessel inside surface and 0.0166E+19 n/cm^2 at the welds separating shells 4 and 5. Thus, embrittlement due to fluence effects is expected to be much lower and the NRC analysis as described in the NRC staff independent assessment is conservative for Hope Creek in this regard. Therefore, there is significant conservatism in the already low circumferential weld failure probabilities as related to Hope Creek. Other Hope Creek RPV shell weld information which the NRC staff requested be included in this relief request is provided in attached Table 1.

As noted in Table 1 the bounding ΔRT_{NDT} is 21.8 °F using peak fluence at the end of the requested relief period. The shift in RT_{NDT} will be 25% lower when the peak fluence at the weld separating shells 4 and 5 is used. By comparison, using the mean value for fluence and weld chemistry assumed for CB&I reactor vessels in Table 7-5 of the NRC independent assessment report, a ΔRT_{NDT} of 30.16 °F would be derived. Therefore, the calculated ΔRT_{NDT} value for Hope Creek's reactor vessel is less than, and thus bounded by, the embrittlement shift assumed in the NRC's independent assessment. Furthermore it can be seen in Table 1 that the calculated Upper Bound RT_{NDT} value for the Hope Creek belt line welds is 13.6 °F at the end of the requested relief period. For comparison, the highest Upper Bound RT_{NDT} value [i.e. "Inner Surface (RT_{NDT} +2σ) °F"] shown within Tables 7-6, 7-7, and 7-8 of the NRC's independent assessment report of BWRVIP-05, would be the RT_{NDT} of 145 °F shown in Table 7-7 for the B&W fabricated BWR vessels. Thus, the calculated Upper Bound RT_{NDT} values for the Hope Creek reactor vessel circumferential welds are bounded by the limiting RT_{NDT} from Table 7-7 of the NRC independent assessment report, thus providing additional assurance that the Hope Creek vessel welds are also bounded by BWRVIP-05.

In the August 14, 1997 letter, the NRC stated that the frequency of beyond design basis events not addressed in the BWRVIP-05 report should be assessed in the plant specific relief request. In particular, the NRC staff stated that non-design basis cold over-pressure transients should be considered. It is highly unlikely that a BWR would experience a cold over-pressure transient. In fact, for a BWR to experience such an event, the plant would generally require several operator errors. The NRC staff assessment described several types of events that could be precursors to BWR RPV cold overpressure transients. These were identified as precursors because no cold overpressure event has occurred at a U.S. BWR. The staff assessment also identified one actual cold overpressure event that occurred during shutdown at a foreign BWR.

This event apparently included several operators errors that resulted in a maximum RPV pressure of 1150 psi with a temperature range of 79 °F to 88 °F. The Hope Creek operating procedure for reactor coolant pressure boundary inservice system leakage testing has sufficient procedural guidance to prevent a cold over-pressure event during the leak test performed at the conclusion of each outage. In addition, RPV hydrostatic tests are considered infrequently performed tests and would be conducted with heightened management oversight and control. Additionally, the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) pumps at Hope Creek are steam driven and do not function when the plant is in cold shutdown. Actuation of either of these systems would not lead to a cold over-pressure event. The challenge to the RPV from a non-design basis cold over-pressure transient is highly unlikely at Hope Creek during the requested delay. Therefore, the probability of a cold over-pressure transient is considered to be less than or equal to that used in the NRC analysis described in the NRC's independent assessment and is conservative for Hope Creek.

In addition to the information above , plant specific inspection information provides further bases to conclude vessel failure is highly unlikely. Preservice examinations of the reactor pressure vessel welds were performed in accordance with the 1977 Edition, Summer 1978 Addenda, of Section XI and NRC Regulatory Guide 1.150. The exams found that the RPV shell course plate material contained no reportable indications and of those indications that were recorded most all were attributable to impedance mismatch at the cladding interface or ID cladding geometry.

Inservice examinations of one circumferential and three axial welds were performed in accordance with the 1983 Edition, Summer 1983 Addenda of Section XI and NRC Regulatory Guide 1.150. The results of this examination were as follows:

1. Examination of circumferential weld W-8 had two Section XI recordable indications and three Reg. Guide 1.150 recordable indications. All of these indications were acceptable to the code tables and required no subsequent evaluation.
2. Examination of axial weld W-15-3 had one Section XI recordable indications and two Reg. Guide 1.150 recordable indications. All of these indications were acceptable to the code tables and required no subsequent evaluation.
3. Examination of axial welds W15-1 and W15-2 revealed no recordable indications.

Based on the documentation in BWRVIP-05, the risk-informed independent assessment performed by the NRC staff, and the discussion above, a delay for two refueling cycles in completing the inspection of the RPV circumferential shell welds at Hope Creek provides an acceptable level of quality and safety.

Table 1
Hope Creek RPV Shell Weld Information

Neutron fluence at the end of the requested relief period (1)	0.0268E19 n/cm^2
Initial (unirradiated) reference temperature	-30 °F
Weld chemistry factor (CF)	106
Weld copper content	0.08%
Weld nickel content	0.63%
Increase in reference temperature due to irradiation (ΔRT_{NDT})	21.8 °F
Margin term	21.8 °F
Mean adjusted reference temperature (ART)	-8.2 °F
Upper bound adjusted reference temperature (ART)	13.6 °F

Note: Peak vessel surface fluence at end of cycle 9