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United States Nuclear Regulatory Commission
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Perry Nuclear Power Plant
Docket No. 50-440
Additional Information Regarding a Request for Alternative Examination
Augmented Reactor Vessel Inspection - Inservice Inspection Program

Ladies and Gentlemen:

The Attachment to this letter provides additional information on a request for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the augmented examination requirements of 10 CFR 50.55a(g)(6)(ii)(A)(2). The initial request was submitted by letter dated August 28, 1997 (PY-CEI/NRR-2210L), as supplemented by letter dated September 4, 1997 (PY-CEI/NRR-2211L). Specifically, pursuant to the provisions of 10 CFR 50.55a(a)(3)(i), and consistent with information contained in NRC Information Notice 97-63, "Status of NRC Staff's Review of BWRVIP-05," relief from the examination of the reactor pressure vessel circumferential shell welds for two operating cycles was requested. Relief Request IR-030 has been revised to provide an alternative examination. The Attachment provides the revised relief request.

If you have any questions, or require additional information, please contact Henry L. Hegrat, Manager - Regulatory Affairs at (440) 280-5606.

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Very truly yours,

Lew W Myers

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Add: NRR/DE/EMEB Ltr 1 Encl 1
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Attachment

cc: NRC Resident Inspector
NRC Project Manager
NRC Region III
Authorized Nuclear Inservice Inspector (ANII)
State of Ohio

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST No. IR-030

I. Identification of Components

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

Mark Number	Description
1B13-AA	Lower Head to Number 1 Shell Ring Circumferential Seam
1B13-AB	Number 1 Shell Ring to Number 2 Shell Ring Circumferential Seam
1B13-AC	Number 2 Shell Ring to Number 3 Shell Ring Circumferential Seam
1B13-AD	Number 3 Shell Ring to Number 4 Shell Ring Circumferential Seam

Note: This represents all of PNPP's B1.11 welds.

II. ASME Boiler & Pressure Vessel Code Section XI and 10CFR50.55a(g)(6)(ii)(A)(2) Requirements

In accordance with Table IWB-2500-1 of both the 1983 Edition through Summer 1983 Addenda of Section XI (PNPP'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 first Inspection Interval. The last refueling outage for PNPP's first Inspection Interval is RFO6, which started on September 12, 1997.

III. Relief Requested

Pursuant to 10CFR50.55a(a)(3)(i), and consistent with the information contained in Information Notice 97-63, "Status of NRC Staff's Review of BWRVIP-05," relief is requested from performing volumetric examination of the reactor pressure vessel pressure retaining circumferential shell welds by the end of the first Inspection Interval. Alternatively, the examinations will be performed, following two additional operating cycles, in RFO8.

IV. 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 Boiling Water Reactor (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 has conducted an independent risk-informed assessment of the analysis contained in BWRVIP-05. 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. Therefore, the NRC evaluation appears to support the conclusions of BWRVIP-05.

This independent NRC assessment used the FAVOR code to perform 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 PNPP. In the case of PNPP, there are no circumferential welds within the beltline region. As such, the limiting circumferential welds would be 1B13-AB, which is 6" below the bottom of active fuel, and 1B13-AC, which is 16" above the top of active fuel. Of these, it has been determined that 1B13-AB would be the limiting circumferential weld within the vessel (i.e., relative to RT_{NDT}). For plants with RPVs fabricated by Chicago Bridge & Iron (CB&I), as is PNPP's, the mean end-of-license neutron fluence used in the NRC PFM analysis was 0.19E+19 n/cm^2. However, the highest overall fluence (i.e., peak fluence at the vessel ID surface) anticipated at the end of the requested relief period is only 1.53E+18 n/cm^2 for PNPP. Furthermore, the highest fluence anticipated at the end of the requested relief period for circumferential welds 1B13-AB and 1B13-AC is only 5.8E+17 n/cm^2 and 9.03E+17 n/cm^2 respectively. Thus, the fluence effect on embrittlement is much lower, in fact it is negligible for the circumferential welds, and the NRC analysis described at an August 8, 1997, meeting with the industry is very conservative for PNPP in this regard. Therefore, there is significant conservatism in the already low circumferential weld failure probabilities as related to PNPP.

Other PNPP RPV shell weld information that the NRC staff has requested is included in Table 1 of this relief request. As shown in Table 1, the calculated embrittlement shift in RT_{NDT} (i.e., ΔRT_{NDT}) at the end of the requested relief period for circumferential welds 1B13-AB and 1B13-AC is 13 °F and 21.4 °F respectively. By comparison, using the mean values for fluence and weld chemistry assumed for CB&I reactor vessels in Table 7-5 of Enclosure 1 to the NRC independent assessment report, a ΔRT_{NDT} of 30.16 °F would be derived. Therefore, the calculated ΔRT_{NDT} values for the PNPP circumferential welds are 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 at the end of the requested relief period for circumferential welds 1B13-AB and 1B13-AC is 6 °F and -17.1 °F respectively. For comparison, the highest Upper Bound RT_{NDT} value [i.e., "Inner Surface ($RT_{NDT} + 2\sigma$)°F"] shown within Tables 7-6, 7-7, and 7-8 of Enclosure 1 to the NRC's independent assessment report of BWRVIP-05, would be the RT_{NDT} of 145.1° F shown within Table 7-7 for the B&W fabricated BWR vessels. Again, the calculated Upper Bound RT_{NDT} values for PNPP's circumferential vessel welds are clearly bounded by the limiting RT_{NDT} from Table 7-7 of the NRC independent assessment report, thus providing additional assurance that the PNPP circumferential vessel welds are also bounded by BWRVIP-05.

At the meeting on August 8, 1997, the NRC staff indicated that the potential for, and consequences of, non-design basis events not addressed in the BWRVIP-05 report should be considered. 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 would generally require several operator errors. At the meeting of August 8, 1997, the NRC staff described several types

of events that could be precursors to BWR RPV cold over-pressure transients. These were identified as precursors because no cold over-pressure event has occurred at a U. S. BWR. Also, at the August 8 meeting, the NRC staff identified one actual cold over-pressure event that occurred during shutdown at a non-U. S. BWR. This event apparently included several operational errors that resulted in a maximum RPV pressure of 1150 psi with a temperature range of 79° F to 88° F.

One of the two high pressure make-up systems is the Reactor Core Isolation Cooling (RCIC) system which is steam turbine driven. During reactor cold shutdown conditions, no steam is available for operation of this system. Therefore, it is not plausible for this system to contribute to an overpressurization event while the unit is in cold shutdown. The other high pressure make-up system is the High Pressure Core Spray System (HPCS). Since the reactor vessel level instruments used to close the HPCS injection valve on reactor vessel high water level are calibrated for hot, pressurized conditions, the instruments normally give a false high level signal during mode 4 operations which would prevent the injection valve from opening on a HPCS initiation. Therefore, the possibility of a HPCS initiation causing an overpressurization event while the unit is in cold shutdown is very small.

One precursor event was identified for PNPP in Table C-1 of the NRC's Independent Assessment Report for BWRVIP-05. The event was the inadvertent starting of the High Pressure Core Spray System (HPCS) diesel generator while in Mode 4 (Cold Shutdown) during maintenance of reactor vessel level instrumentation associated with starting of the HPCS system. The HPCS pump and various valves had been placed in a secured status prior to the event; therefore, no actual injection of the high pressure system was possible. There were no other events during cold shutdown conditions which involved a potential inadvertent initiation of the HPCS system. This event did not result in a violation of the pressure-temperature limits since no actual initiation took place.

PNPP classifies all RPV hydrostatic tests as Infrequently Performed Tests or Evolutions (IPTE's). IPTE's receive special management oversight to maintain the plant's level of safety within acceptable limits. Therefore, a challenge to the RPV from a non-design basis cold over-pressure transient is highly unlikely for PNPP during the requested delay and the probability of a cold over-pressure transient is considered to be less than or equal to that used in the NRC analysis described at the August 8 meeting.

The NRC staff has recently transmitted a Request for Additional Information (RAI) regarding the BWRVIP-05 report to the BWR Vessel and Internals Project (BWRVIP). The BWRVIP plans to provide a response to that RAI in the near future that will include additional information on the BWRVIP PFM analysis, comparisons to the NRC staff PFM analysis and additional information regarding beyond design basis cold over-pressure transients. PNPP will work with the BWRVIP to resolve the longer term issues in this area, but PNPP believes BWRVIP-05 and the NRC analysis provide sufficient basis to support approval of this relief request.

It is noteworthy that preservice examinations of PNPP's RPV shell 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 mid-plate segregate or short-length laminar indications, that were almost all below recordable levels, throughout the vessel. Coverage for the axial and circumferential shell weld exams averaged greater than 80%. For the subject circumferential welds, all but 1B13-AA, which was a one sided exam due to the vessel skirt, were "essentially 100% examined". Within those welds, there were only 8 recordable indications and no reportable indications.

In summary, based on the documentation in BWRVIP-05, the risk-informed independent assessment performed by the NRC staff, and the discussion above, a delay of two (2) operating cycles (until the end of the eighth refueling outage) for completion of inspections of the RPV circumferential shell welds is justified.

V. Alternate Examination

Inspections of essentially 100 percent of the longitudinal seam welds in the PNPP Unit No. 1 reactor vessel shell will be performed during RFO6, and inspections of the reactor vessel shell circumferential seam welds will be deferred for two (2) operating cycles. As stated in 10 CFR 50.55a(g)(6)(ii)(A)(2), "essentially 100%" as used in Table IWB-2500-1 means more than 90 percent of the examination volume of each weld, where the reduction in coverage is due to interference by another component or part geometry. The examinations of the longitudinal welds, based on the ultrasonic testing techniques being employed, will result in partial examination of circumferential welds at or near the intersections of these welds.

Table 1[1]
Perry Unit 1 RPV Circumferential Shell Weld Information

Description	Value for 1B13-AB (Shell Ring 1 to 2)	Value for 1B13-AC (Shell Ring 2 to 3)
Neutron fluence at the end of the requested relief period (10 EFPY)[2][3]	5.8 E+17 n/cm^2	9.03 E+17 n/cm^2
Initial (unirradiated) reference temperature	-20 °F	-60 °F
Weld chemistry factor (CF)	41.0	54.0
Weld copper content	0.03 %	0.04 %
Weld nickel content	0.81 %	0.97 %
Increase in reference temperature due to irradiation (ΔRT_{NDT})	13 °F	21.4 °F
Margin term	13 °F	21.4 °F
Mean adjusted reference temperature (Mean ART)	-7 °F	-38.6 °F
Upper bound adjusted reference temperature (ART)	6 °F	-17.1 °F

(1) Table information was supplied by GE Nuclear Energy via GE letter LJT-9716, dated August 27, 1997 and assigned Technical Assignment File Number 81658.
(2) Values in this table were calculated for 10 Effective Full Power Years of operation which is approximately where PNPP will be at the end of the requested relief period (i.e., RFO8).
(3) Fluence values were determined using equation 4-2 of report GE-NE-B13011793-01, "Perry Unit 1 RPV Surveillance Materials Testing and Analysis", which was submitted to the NRC in accordance with 10 CFR 50 Appendix H via letter PY-CEI/NRR-2129L, dated February 18, 1997.