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United States Nuclear Regulatory Commission
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Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
Submittal of Inservice Examination Relief Requests

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a(a)(3)(i and ii), relief requests for the Perry Nuclear Power Plant In-service Examination Program are being proposed. Attachment 1 contains the identification of the affected components, the applicable code requirements, the description of the proposed relief request, the basis for the relief and the proposed alternate testing. Relief Request IR-023 addresses testing of snubbers, Relief Request IR-029 addresses examination of welds for piping four inches and larger, Relief Request IR-030 addresses examination of reactor pressure vessel (RPV) shell welds, Relief Request IR-041 addresses the Alternate Requirements for Containment (IWE/IWL) Repair/Replacements, Relief Request IR-044 addresses examination of RPV closure nuts and Relief Request IR-045 addresses RPV shell to flange and head to flange weld examinations. Review and approval of the relief requests are requested by October 31, 2000, in order to plan for the RF08 refueling outage.

There are no regulatory commitments contained in this letter. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

For John K. Wood
Attachments

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

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Table of Contents

Subject	Relief Request Number	Page Numbers
Testing of Snubbers	Relief Request IR-023 Revision 2	pages 2 through 5
Examination of Welds for Piping Four Inches and Larger	Relief Request IR-029 Revision 2	pages 6 through 10
Examination of Reactor Pressure Vessel (RPV) Shell Welds	Relief Request IR-030 Revision 1	pages 11 through 18
Alternate Requirements for Containment (IWE/IWL) Repair/Replacement	Relief Request IR-041 Revision 0	pages 19 through 20
Examination of RPV Closure Nuts	Relief Request IR-044 Revision 0	page 21
RPV SHELL to FLANGE and HEAD to FLANGE WELD EXAMINATIONS	Relief Request IR-045 Revision 0	pages 22 through 24

Sheet 1 of 4

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-023, Rev. 2

I. Identification of Components

All safety-related hydraulic and mechanical snubbers.

II. ASME B&PV Section XI Requirements

ASME XI, 1989 Edition, Subarticle IWF-5000, Inservice Inspection Requirements for Snubbers, states that preservice examinations and tests of snubbers (IWF-5200), inservice examinations and tests of snubbers (IWF-5300), and examination and tests of snubber repairs and replacements (IWF-5400) shall be in accordance with the first Addenda to ASME/ANSI OM-1987, Part 4. Furthermore, 10 CFR 50.55a(b)(2)(viii) specifies that the ASME/ANSI OM Part 4 Edition and Addenda to be used shall be the OMa-1988 Addenda to the OM-1987 Edition.

III. Relief Request

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested from performing preservice examinations and tests of snubbers, inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements in accordance with ASME/ANSI OM Part 4 of the OMa-1988 Addenda to the OM-1987 Edition.

IV. Basis for Relief

Perry performs preservice and inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements in accordance with the technical requirements within section 6.4.1 of the Operational Requirements Manual (ORM). The requirements were relocated from Technical Specification 4.7.4 upon Perry's incorporation of Improved Technical Specifications. The basic technical requirements within ORM 6.4.1 for examination and testing of snubbers are essentially the same as those within ASME/ANSI OM Part 4 of the OMa-1988 Addenda to the OM-1987 Edition. Additionally, the ORM provides requirements for the inspection of snubbers following transient events and a snubber service life replacement program.

Revision 0 of this relief request, which requested similar relief for snubbers for Perry's first 10-year inspection interval, was approved by the NRC (reference TAC No. M61303, dated 9/7/90).

Revision 1 of this relief request, which was resubmitted for the start of Perry's second 10-year inspection interval, was approved by the NRC (reference TAC No. MA3437, dated 11/22/99).

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-023, Rev. 2

Revision 2 updates Perry's second ORM testing plan option to be consistent with the "37 Testing Sample Plan" of the OMa-4 Code, 1995 Edition, up to the 1996 Addenda, which is the Code of reference approved in the recent changes to 10CFR50.55a(b)(3)(v) (see 64 FR 51388; September 22, 1999). Upon approval of this relief request, the proposed changes to the ORM will be implemented. Also, prior to use of the revised test plan, the Nuclear Regulatory Commission will be notified in writing pursuant to 10 CFR 50.4, as required by ORM Section 6.4.1.1.e.

V. Alternate Testing

Preservice and inservice examinations and tests of snubbers, and examinations and tests of snubber repairs and replacements in accordance with the technical requirements within section 6.4.1 of the Operational Requirements Manual (ORM). The functional testing requirements therein, or, in the case of the second sample plan, proposed to be therein, are as follows:

At least once per refueling interval, a representative sample of snubbers shall be tested using one of the following 3 plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The snubber functional test period may start ninety days prior to a scheduled refueling outage and shall be completed prior to the end of the scheduled refueling outage. The Nuclear Regulatory Commission shall be notified in writing pursuant to 10 CFR 50.4 of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented.

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

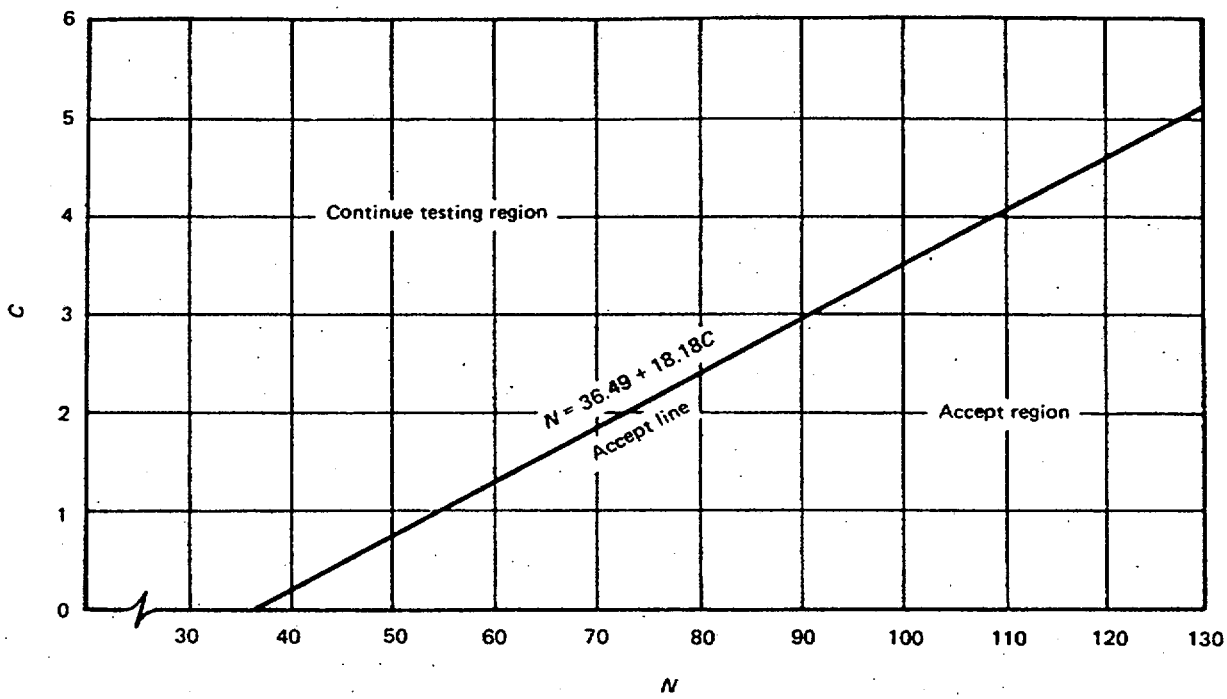
2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 6.4.1-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 6.4.1-1. If at any time the point plotted falls on or below the "Accept" line, testing of snubbers of that type may be

terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of the equipment failure are retested; or

3) An initial representative sample of 55 snubbers of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls on or below the "Accept" line or all the snubbers of that type have been tested. The representative sample selected for the function test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional but shall not be included in the sample plan, and failure of this functional test shall not be the sole cause for increasing the sample size under the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

Sheet 4 of 4

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-023, Rev. 2



SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST
Figure 6.4.1.-1

Sheet 1 of 5

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-029, Rev. 2

I. Identification of Components

Class 1, Category B-J (Item numbers in attached table), piping welds 4 inches NPS and greater.

II. ASME B&PV Section XI Requirements

ASME XI, 1989 Edition, Table IWB-2500-1, Category B-J requires 100% surface and volumetric examination of 25% of the circumferential butt welds and their intersecting longitudinal welds. In accordance with Category B-J Note 1 of the Table, the examinations are to include all terminal ends, joints where the seismic and operational load stress levels exceed primary plus secondary stress intensity range of $2.4S_m$ (i.e., "high stress" location) or a cumulative usage factor U of 0.4 (i.e., "high fatigue" location), all dissimilar metal welds, and additional welds (if necessary) such that the total number of circumferential butt welds equals 25%.

III. Relief Requested

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested from selecting the welds in accordance with Note 1 when structural interferences make such selection impractical. The relief is requested for Perry's second 10-year inspection interval.

IV. Basis for Relief

The welds identified in the attached table are "high stress" welds, but examination is impractical as they are in radiation areas and are encased in jet impingement shields. The jet shields are elbow or tee-shaped structural steel enclosures around Reactor Recirculation (RR) System and Main Steam (MS) System piping welds. The smallest of the RR jet shields weighs over 1600 lbs. and is assembled with 48 bolts. Each of the MS jet shields weighs over 2240 lbs. and is assembled with 180 bolts. The bolting for all of the jet shields is high strength, one time use, bolting that must be torqued to 10,000-16,000 ft/lbs. Disassembly, for inspection, and re-assembly of these jet shields would be a labor-intensive effort with over 100 man-hours each. General area dose rates for the MS jet shield locations range from 20-50 mr/hr and contact dose rates for the RR piping beneath the RR jet shields range from 200-400 mr/hr. Therefore, removal of any of the jet shields would require significant dose expenditure.

The structural integrity of the piping pressure boundary was demonstrated during construction by meeting the requirements of the ASME Code Section III, and additionally by meeting the requirements of ASME Section XI during preservice inspections. The subject RR and MS welds were examined (prior to installation of the jet shields) in accordance

with the appropriate Code requirements, weld techniques and welders were qualified in accordance with Code requirements, and materials were purchased and traced in accordance with the appropriate Code and NRC requirements and guidelines. There were no reportable indications during preservice inspection. Additionally, the MS jet shields were removed in the first inspection interval (at considerable dose and monetary cost), the welds received inservice examinations, and they were found to be free of reportable indications.

The pressure boundary passed the required preservice hydrostatic and first interval inservice pressure tests, and has operated for a total of about 3,220 equivalent full power days from November 1987 through the end of 1999 without leakage indication attributable to the subject welds.

Complete examinations meeting the requirements of the ASME Code Section XI have been performed on similar "high stress" welds within the RR and MS Systems where jet shields are not present or are easily removed, with satisfactory results. These welds are subject to the same operating and environmental conditions as the obstructed welds.

Other RR and MS welds of the same size and configuration, but that are not "high stress" welds, will be examined in place of the obstructed welds. In accordance with ASME Research White Paper, "Risk-Based Alternative Selection Process for Inservice Inspection of LWR Nuclear Power Plant Components," (Library of Congress Catalog Number 94-71660) a recent industry survey, which included 50 nuclear units representing 733 cumulative years of operation, found that there is no apparent relationship between the type of welds selected for inspection (i.e., high design stress/fatigue welds versus low stress/fatigue welds) and the detection of flaws.

Design, procurement and operational provisions against nil ductile failure of the subject welds remain as described in the Perry USAR.

Revision 1 of this relief request, which requested the same relief for welds of the same Examination Category for Perry's second 10-year inspection interval, was approved by the NRC (reference TAC No. MA3437, dated 11/22/99).

Revision 2 adds two additional welds to the scope of the relief. In preparation for an upcoming refueling outage, it was identified that the two welds are also within jet shields.

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-029, Rev. 2

In summary, because of the dose burden, acceptable initial condition, successful Code hydrotest and operating experience without related leakage indications, the satisfactory examination of identical welds, the substitution of welds of similar size and configuration, protection against brittle failure, and the previous approval of the same relief it is concluded that the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

V. Alternate Examination

Welds of the same size and similar configuration, but that are not "high stress" welds, will be examined in place of the obstructed welds to maintain the 25% selection requirement.

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-029, Rev. 2

ITEM NO.	COMPONENT I.D.	SYS./DWG. NO.	DESCRIPTION	NATURE OF OBSTRUCTION	% COMPLETE
B9.11	1B33-0029 (1B33-0028)	RR/602-101	16" PIPE TO 12" x 16" SWEEPOLET (16" PIPE TO 12" x 16" SWEEPOLET)	JET IMPINGEMENT SHIELD RBRAD8	0%
B9.11	1B33-0032 (1B33-0090)	RR/602-101	16" PIPE TO 12" x 16" SWEEPOLET (16" PIPE TO 12" x 16" SWEEPOLET)	JET IMPINGEMENT SHIELD RBRAD7	0%
B9.11 B9.12	1B33-0043 1B33-0043-U (1B33-0046 1B33-0046-D)	RR/602-101	12" PIPE TO NOZZLE SAFE-END, CRC; AND UPSTREAM LONGSEAM (12" PIPE TO 12" x 24" REDUCER, CRC; AND DOWNSTREAM LONGSEAM)	JET IMPINGEMENT SHIELD RBRAD4	0%
B9.11 B9.12	1B21-0002 1B21-0002U 1B21-0002D (1B21-0005 1B21-0005U 1B21-0005D)	MS/605-103	26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBSC1	0%
B9.11 B9.12	1B21-0039 1B21-0039U 1B21-0039D (1B21-0042 1B21-0042U 1B21-0042D)	MS/605-104	26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBSD1	0%
B9.11 B9.12	1B21-0040 1B21-0040U 1B21-0040D (1B21-0043 1B21-0043U 1B21-0043D)	MS/605-104	26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBSD1	0%

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST #IR-029, Rev. 2

ITEM NO.	COMPONENT I.D.	SYS./DWG. NO.	DESCRIPTION	NATURE OF OBSTRUCTION	% COMPLETE
B9.11 B9.12	1B21-0079 1B21-0079U 1B21-0079D (1B21-0082 1B21-0082U 1B21-0082D)	MS/605-102	26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBSB1	0%
B9.11 B9.12	1B21-0118 1B21-0118U 1B21-0118D (1B21-0121 1B21-0121U 1B21-0121D)	MS/605-101	26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBSA1	0%
B9.11 B9.12	1B21-0119 1B21-0119U 1B21-0119D (1B21-0122 1B21-0122U 1B21-0122D)	MS/605-101	26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (26″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBSA1	0%
B9.11	1B33-0091 (1B33-0092)	RR/602-103	16" PIPE TO 12" x 16" SWEEPOLET (16" PIPE TO 16" x 24" CROSS)	JET IMPINGEMENT SHIELD RBRBD8	0%
B9.11 B9.12	1B33-0099 1B33-0099U 1B33-0099D (1B33-0119 1B33-0119U 1B33-0119D)	RR/602-103	12″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS (12″ PIPE TO ELBOW AND UPSTREAM AND DOWNSTREAM LONGSEAMS)	JET IMPINGEMENT SHIELD RBRBD5	0%

Note: Welds in parentheticals are the non "high stress" welds to be substituted for the obstructed "high stress" welds.

RR = REACTOR RECIRCULATION SYSTEM
MS = MAIN STEAM SYSTEM
CRC = CORROSION RESISTANT CLAD WELD

Sheet 1 of 8

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

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 the 1989 Edition with no Addenda of Section XI, and the Augmented Reactor Vessel examination requirements of 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 each inspection interval.

III. Relief Requested

Pursuant to 10CFR50.55a(a)(3)(i) and 10CFR50.55a(g)(6)(ii)(A)(5), permanent relief is requested from the inservice inspection requirements of 10CFR50.55a(g) for the cirumferential reactor pressure vessel welds. The relief is being requested consistent with the guidance contained in Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds."

IV. Basis for Relief

Background Information:

Consistent with the guidance contained in Information Notice 97-63, "Status of NRC Staff's Review of BWRVIP-05," PNPP requested 2 cycle relief from performing volumetric examination of the reactor pressure vessel pressure retaining circumferential shell welds (Rev 0 of this relief request). By letter dated September 18, 1997 (TAC No. M99474), the NRC approved the requested relief.

Sheet 2 of 8

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

Technical Basis:

The technical bases for PNPP's request is documented in the BWRVIP-05 report, which was transmitted to the NRC in September 1995, and the NRC Safety Evaluation Report, "Evaluation by the Office of Nuclear Reactor Regulation Related to the Review of the Topical Report by the Boiling Water Reactor Vessel and Internals Project: 'BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,' (BWRVIP-05)," dated July 28, 1999 [Reference: NRC TAC No. M93925].

The NRC evaluation of the BWRVIP-05 report used the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate reactor pressure vessel failure probabilities. Three key assumptions in the PFM analysis are: 1) the neutron fluence was that estimated to be end-of-license (EOL) mean fluence, 2) the chemistry values are mean values based on vessel types, and 3) the potential for beyond design basis events is considered. Although the BWRVIP-05 report provides the technical basis for eliminating inspection of BWR reactor pressure vessel circumferential shell welds, the following information is provided to demonstrate the conservatism of the NRC analysis relative to PNPP's reactor pressure vessel.

Reactor Vessel Integrity

The BWRVIP-05 report concludes that the probability of failure of the BWR reactor pressure vessel circumferential shell welds is orders of magnitude lower than that of the axial shell welds. The BWRVIP-05 report indicates that, for a typical BWR reactor pressure vessel, the failure probability for axial welds is 2.7×10^{-7}, and the failure probability for circumferential welds is 2.2×10^{-41} for 40 years of plant operation (i.e., until end-of-license or 32 EFPY).

ΔRT_{NDT} is a measure of the amount of irradiation embrittlement at peak ID fluence (i.e., within the reactor pressure vessel beltline region). In the case of PNPP, there are no circumferential welds located 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 reactor pressure vessels fabricated by Chicago Bridge & Iron (CB&I), the mean end-of-license neutron fluence used in the NRC PFM analysis was 0.51E+19 n/cm^2. The highest fluence values anticipated at the end-of-license for welds 1B13-AB and 1B13-AC are 0.19E+19 n/cm^2 and 0.29E+19 n/cm^2 respectively. Thus, the fluence effect on embrittlement is lower for PNPP's limiting circumferential welds and significant conservatism exists in the already low circumferential weld failure probabilities as related to PNPP.

As shown in Table 1, the calculated embrittlement shift in RT_{NDT} (i.e., ΔRT_{NDT}) at the end-of-license (i.e., 32 EFPY)is 22.8 °F for weld 1B13-AB and 35.7 °F for weld 1B13-AC. By comparison, Table 2.6-4 of the NRC's Final Safety Evaluation of the BWRVIP-05 Report indicates an embrittlement shift of 109.5 °F for CB&I fabricated vessels. Therefore, the calculated ΔRT_{NDT} values for PNPP's limiting circumferential welds are less than, and thus bounded by, the embrittlement shift assumed in the NRC's Final Safety Evaluation of the BWRVIP-05 report.

Furthermore, as seen in the attached Table 1, the calculated Upper Bound RT_{NDT} values for PNPP's limiting circumferencial welds at end-of-license are 25.6 °F for 1B13-AB and 11.4 °F for 1B13-AC. For comparison, the calculated Upper Bound RT_{NDT} for circumferential welds in CB&I fabricated vessels utilizing the information in Table 2.6-4 of the NRC's Final Safety Evaluation of the BWRVIP-05 report is 100.5 °F. Thus, the calculated Upper Bound RT_{NDT} values for PNPP's limiting circumferential welds are clearly bounded by the calculated limiting RT_{NDT} for CB&I vessels and additional assurance is provided that the welds are also bounded by the BWRVIP-05 report and NRC plant specific analyses.

Beyond Design Basis Considerations:

At a 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, 1997 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.

Review of Potential High Pressure Injection Sources

One of PNPP's two high pressure make-up systems is the Reactor Core Isolation Cooling (RCIC) system. This system is driven by a steam turbine. 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.

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

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 for PNPP was identified within 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.

During normal cold shutdown conditions, reactor pressure vessel level and pressure are controlled with the Control Rod Drive (CRD) and Reactor Water Cleanup (RWCU) systems using a "feed and bleed" process. The reactor is not taken solid during these times, and plant procedures require opening of the head vent valves after the reactor has been cooled to less than 190°F. If either of these systems were to fail, the Supervising Operator (SO) would adjust the other system to control level. Under these conditions, the CRD system typically injects water into the reactor at a rate of <60 gpm. This slow injection rate allows the operator sufficient time to react to unanticipated level changes and, thus, significantly reduces the possibility of an event that would result in a violation of the pressure-temperature limits.

The Standby Liquid Control (SLC) system is another high pressure water source to the reactor pressure vessel. However, there are no automatic starts associated with this system. SLC injection requires an Operator to manually start the pump from the Control Room or from the local test station. Additionally, the injection rate of the SLC pump is approximately 41 gpm, which would give the Operator ample time to control reactor pressure in the case of an inadvertent injection.

Pressure testing of the RPV is classified as an "Infrequently Performed Test or Evolution" which ensures that these tests receive special management oversight and procedural controls to maintain the plant's level of safety within acceptable limits. PNPP practice is to

monitor and maintain a large temperature margin at the initial stages of pressurization while raising the vessel temperature to the hydrostatic test temperature (approximately 140°F). During pressurization and performance of an RPV pressure test, level and pressure are controlled with the CRD and RWCU systems using a "feed and bleed" procedure. Increase in pressure is limited to 50 psi per minute. When at test pressure (1025 to 1050 psig), if the temperature nears the limits of the temperature verse pressure curve, the procedure requires that the pressure be reduced to less than 700 psig. If any other termination criteria is met then the test is placed on hold after the plant is restored to a safe condition as directed by the procedure. These practices minimize the likelihood of exceeding the pressure-temperature limits during performance of the test.

Procedural Controls/Operator Training to Prevent RPV Cold Over-Pressurization

Operating procedural restrictions, operator training, and work control processes at PNPP provide appropriate controls to minimize the potential for RPV cold over-pressurization events.

During normal cold shutdown conditions, reactor water level, pressure, and temperature are maintained within established bands in accordance with operating procedures. Operations policies require that Supervising Operators (SOs) frequently monitor for indications and alarms to detect abnormalities as early as possible. These policies also require that the Unit Supervisor (US) be notified immediately of any changes or abnormalities in indications. Furthermore, procedures require that changes which could affect reactor level, pressure, or temperature only be performed under the knowledge and direction of the US. Therefore, any deviations in reactor water level or temperature from a specified band will be promptly identified and corrected. Finally, the status of plant conditions, and ongoing activities which could affect critical plant parameters, and contingency planning are discussed by Operators at each shift turnover. This ensures that on-coming Operators are cognizant of any activities which could adversely affect reactor level, pressure, or temperature.

A review of industry operating experience indicates that inadequate work management is a potential contributor to a cold over-pressurization event. At PNPP, work performed during outages is scheduled by the Work Management Section. Dedicated Senior Reactor Operators provide oversight of outage schedule development to avoid conditions which could adversely impact reactor water level, pressure, or temperature. From the outage schedule, a plan of the day (POD) is developed listing the work activities to be performed. These PODs are reviewed and approved by Management, and a copy is maintained in the

Control Room. Changes to the PODs require Management review and approval. Additionally, the detailed outage schedule receives a thorough shutdown risk assessment review to ensure defense-in-depth is maintained.

During outages, work is coordinated through the Work Support Center, which provides an additional level of Operations oversight. In the Control Room, the US is required, by procedure, to maintain cognizance of any activity which could potentially affect reactor level or decay heat removal during refueling outages. The SO is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Pre-job briefings are conducted for work activities that have the potential of affecting critical reactor parameters. These briefings are attended by the cognizant individuals involved in the work activity. Expected plant responses and contingency actions to address unexpected conditions or responses that may be encountered are included in the briefing discussion.

The plant procedure for unit shutdown limits reactor pressure to ≤15 psig while flooding up to cold shutdown water level and requires frequent monitoring of reactor pressure to ensure that this limit is not exceeded. Additionally, this procedure requires opening of the head vent valves after the reactor has been cooled to less than 190°F and either the main steam isolation valves are closed or condenser vacuum is broken. Procedural controls for reactor temperature, level, and pressure are an integral part of Operator training. Specifically, Operators are trained in methods of controlling water level within specified limits, as well as responding to abnormal water level conditions outside the established limits. Additionally, SOs receive training on brittle fracture limits and compliance with the Technical Specification pressure-temperature limits curves. Plant-specific procedures have been developed to provide guidance to the Operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

PNPP RPV Weld Examination History:

Preservice examinations of PNPP's RPV shell axial and circumferential 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. There were no reportable indications. 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.

During RFO6, which occurred in the fall of 1997, all of PNPP's axial shell welds were examined utilizing the General Electric automated GERIS-2000 sytstem. Coverage was essentially 100% and, again, there were only a few recordable and no reportable indications. In examining the axial welds, an average incidental coverage of 2-3% of the circumferential welds was also achieved at or near the intersection of the welds.

Summary:

The BWRVIP-05 report provides the technical basis for eliminating inspection of BWR reactor pressure vessel circumferential shell welds. The BWRVIP-05 report concludes that the probability of failure of the BWR reactor pressure vessel circumferential shell welds is orders of magnitude lower than that of the axial shell welds. Based on an assessment of the materials in PNPP's limiting circumferential welds, the conditional probability of reactor pressure vessel failure should be less than that estimated in the NRC's analysis. Based on Operator training and established procedures that have been implemented, the low probability of cold over-pressure transients will limit the frequency of cold over-pressure events to the amounts specified in the NRC's June 30, 1998, safety evaluation.

V. Alternate Examination

Examinations of essentially 100% of PNPP's axial seam welds will be performed in accordance with 10CFR50.55a(g) during the remaining inspection intervals of PNPP's current license term. The examinations of the axial welds will result in partial coverage, approximately 2-3%, of the circumferential welds at or near the intersection of these welds.

Sheet 8 of 8

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

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

Description	**Values for 1B13-AB (Shell Ring 1 to 2)**	**Values for 1B13-AC (Shell Ring 2 to 3)**	**USNRC Limiting Plant Specific Analysis Parameters, SER Table 2.6-4**
Neutron fluence, n/cm^2, at the end of the requested relief period (32 EFPY)[2]	.19 E+19	.29 E+19	.51 E+19
Initial (unirradiated) reference temperature (RT_{NDT})	-20 °F	-60 °F	-65 °F
Weld chemistry factor (CF)	41.0	54.0	134.9[3]
Weld copper content	0.03 %	0.04 %	0.10 %
Weld nickel content	0.81 %	0.97 %	0.99 %
Increase in reference temperature due to irradiation (ΔRT_{NDT})	22.8 °F	35.7 °F	109.5 °F
Margin term	22.8 °F	35.7 °F	56 °F[4]
Mean adjusted reference temperature (Mean ART)	2.8 °F	-24.3 °F	44.5 °F
Upper bound adjusted reference temperature (ART)	25.6 °F	11.4 °F	100.5 °F[4]

(1) Table information developed from GE Nuclear Energy letter LJT-9716, dated August 27, 1997 (assigned Technical Assignment File Number 81658) and report GE-NE-B1301793-01, "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.

(2) The fluence values for welds AB and AC were calculated using the peak ID fluence value reported in GE-NE-B1301793-01 and multiplying by the factors of 0.38 and 0.59, respectively. The factors were conservatively obtained from comparing the elevation of the welds with respect to the core and the relative flux distribution shown in Figure 4-3 of the referenced report.

(3) This value erroneously given as 109.5 °F in SER Table 2.6-4. The correct value, derived from Table 1 of 10CFR50.61, is 134.9 °F.

(4) Values not actually given in SER Table 2.6-4, but calculated from values therein.

Sheet 1 of 2

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST No. IR-041

Identification of Components

All Class MC pressure retaining components.

II. ASME B&PV Section XI Requirements

ASME Section XI, 1992 Edition, 1992 Addenda, Subsections IWE and IWL

III. Relief Requested

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested from the documentation requirements of subarticles IWA-4140, "Repair/Replacement Program and Plan", and IWA-4910, "Reports and Records", of the 1992 Edition, 1992 Addenda of ASME Section XI for repairs, replacements and modifications to Class MC (IWE) and CC (IWL)components.

IV. Basis for Relief

In the Federal Register dated August 8, 1996, (61 FR 41303), the NRC amended its regulations to incorporate the 1992 Edition, 1992 Addenda of Section XI when performing containment examinations. In addition to the requirements of Subsection IWE, this also imposes the requirements of the 1992 Edition, 1992 Addenda of Subsection IWA-4000 for containment repairs/ replacements. As clarified in NRC Information Notice 97-29, "Containment Inspection Rule," the IWE and IWL requirements for repair/replacement became effective September 9, 1996. The staff had considered relief from the repair/replacement requirements up to September 9, 1997 as reasonable.

Perry Nuclear Power Plant (PNPP) holds a National Board Nuclear Repair (NR) Certificate of Authorization for repairs/replacements to Class 1, 2, 3, and MC components. This Certificate of Authorization is based on PNPP's repair/replacement program which is written to the requirements of the 1989 Edition with no Addenda of ASME Section XI for PNPP's second 10-year inspection interval. PNPP's repair/replacement program does not currently incorporate the documentation requirements within Subsection IWA-4000, 1992 Edition, 1992 Addenda for containment repairs/replacements. Because PNPP considers the likelihood of a containment repair/replacement becoming necessary as being very remote, addressing the repair/replacement documentation requirements imposed by the NRC's amended containment examination regulations has been deferred until NRC review and approval of this relief is completed. It is understood that if a containment repair/replacement becomes necessary, PNPP would have to address all the repair/replacement requirements of the 1992 Edition, 1992 Addenda of Subsection IWA-4000, or gain relief.

In accordance with 10 CFR 50.55a(g)(4)(ii), PNPP's ISI and Repair/ Replacement programs were recently updated to the 1989 Edition of Section XI for our second 10-year inspection interval. To simplify the process, PNPP updated the repair/replacement documentation requirements for all components within the scope of Section XI to the requirements of the 1989 Edition with no Addenda. The requirements are given in subarticles IWA-4800, "Records" (for Repairs), and IWA-7520, "Reports and Records" (for Replacements). They are essentially the same as those from the 1983 Edition, Summer 1984 Addenda, which was the reference Code for PNPP's

previous inspection interval, and they did not require the generation of any new procedures or record types. However, the documentation requirements in the 1992 Edition, 1992 Addenda are different. They are given in subarticles IWA-4140, "Repair/Replacement Program and Plan" (which describes activities that are to be documented in a formal Repair/Replacement Plan) and IWA-4910, "Reports and Records". There was a major restructuring of the repair and replacement rules by the 1992 Edition, 1992 Addenda, with repairs and replacements all consolidated under IWA-4000. Most of the changes in the 1992 Edition, 1992 Addenda were clarifications of the previous requirements, not new requirements. An exception was a new requirement within subarticle IWA-4140 to have a formal Repair/Replacement Plan for each repair/replacement activity. Repair/Replacement Plans increase the documentation requirements associated with repair/replacement activities. Subarticle IWA-4910 added a list of "as applicable" repair/replacement record requirements, rather than simply providing reference to IWA-6000, "Records and Reports", as IWA-4800 did in the 1989 Edition. If applied to the IWE and IWL components, these requirements would require the development of new procedures and record types. As Repair/Replacement Plans are not required by subarticles IWA-4800 or IWA-7520 of the 1989 Edition, such plans are not required for IWB, IWC, IWD, and IWF examinations. Therefore, for consistency PNPP proposes to use the documentation requirements of IWA-4800 and IWA-7520 of the 1989 Edition for all components that are within the scope of Section XI. In this manner, the documentation requirements for Class MC(IWE) and CC(IWL) components will be no different than those for Class 1 components. It will eliminate the need to create new Repair/Replacement Plan records and the administrative burden associated with the processing and maintenance of those records. The creation of additional records increases the man-hours necessary to develop and maintain PNPP's repair/replacement program without providing a commensurate increase in the quality of containment vessel repairs/replacements.

Based on the above discussion, relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii). Compliance with the repair and replacement documentation requirements of the 1992 Edition, 1992 Addenda of ASME Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety.

V. Alternative Examination

For Class MC(IWE) and CC(IWL) components, PNPP's repair/replacement program will comply with all the requirements of the 1992 Edition, 1992 Addenda of ASME Section XI except the documentation requirements of subarticles IWA-4140 and IWA-4910. In their place the documentation requirements of subarticles IWA-4800 and IWA-7520 of the 1989 Edition will be used for repairs and replacements to the containment vessel.

Sheet 1 of 1

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST No. IR-044
Code Case N-627

I. Identification of Components

Reactor Pressure Vessel Closure Nuts (72 Total).

II. ASME Boiler & Pressure Vessel Code Section XI Requirements

ASME XI, 1989 Edition, Table IWB-2500-1, Examination Category B-G-1, Item B6.10, requires a surface examination of RPV closure nuts.

III. Relief Requested

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested from performing the Code-required surface exams. Alternative examinations are proposed in accordance with Code Case N-627.

IV. Basis for Relief

Use of Code Case N-627 will allow VT-1 visual examinations of the RPV closure nuts as an alternative to performing surface examination. Within the industry, there have been no failures of RPV closure nuts. Furthermore, ASME Subcommittee XI determined that, for the intended purpose of the RPV closure nut exams, a VT-1 examination could replace the surface examination. This was incorporated into the 1989 Addenda of ASME XI and is unchanged through the current Edition and Addenda. Within 10CFR50.55a, the NRC has endorsed ASME Section XI through the 1995 Edition with the 1996 Addenda (refer to 64 FR 51395) without any limitations on the use of the Table IWB-2500-1, Category B-G-1 requirements.

V. Alternate Examination

The alternative requirements of Code Case N-627 will be used. That is, VT-1 visual examinations will be used in lieu of the surface examination required by Table IWB-2500-1, Examination Category B-G-1, Item B6.10.

Sheet 1 of 3

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST No. IR-045
Code Case N-623

I. Identification of Components

Reactor Pressure Vessel Shell to Flange and Head to Flange Welds (PNPP Mark numbers 1B13-AE and 1B13-AG).

II. ASME Boiler & Pressure Vessel Code Section XI Requirements

ASME XI, 1989 Edition, Table IWB-2500-1, Examination Category B-A, Items B1.30 and B1.40, allow deferral of examinations to the end of the inspection interval only if partial examinations are conducted from the flange face in the first inspection period.

III. Relief Requested

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested from having to perform partial examinations from the flange face in order to defer the RPV shell-to-flange and head-to-flange welds to the end of the inspection interval. The alternative scheduling is proposed in accordance with the provisions of Code Case N-623.

IV. Basis for Relief

Background:

The 1975 Edition of ASME Section XI only allowed RPV shell welds to be deferred to the end of the interval. Since that time, the industry has built an extensive experience base with the examination of RPV shell, head-to-shell, flange-to-shell, and nozzle-to-shell welds with no unacceptable examination results. Typically, RPV weld exams, many of which are performed with automated UT equipment, are the most costly ISI exams from both a financial and a dose perspective. As such, and considering the failure free industry examination experience, ASME Subcommittee XI has taken progressive steps to allow deferral of the different welds. In 1978, ASME XI was changed to allow deferral of bottom head welds. In 1988, it was changed to allow deferral of bottom and top head welds. And, in 1993, Code Case N521 (which is endorsed by the NRC in Regulatory Guide 1.147) was issued to allow deferral of nozzle-to-shell welds. All of these steps were made to allow licensees to perform the exams in the most efficient manner (i.e., least expense and dose), which is all at one time at the end of the interval. As the partial flange-to-shell and head-to-flange welds are typically manual exams with less impact than the other RPV weld exams, Subcommittee XI did not take steps to allow their deferral until the issuance of Code Case N-623 in 1999.

Hardship:

Elimination of the flange weld partial examination as a condition for deferral will allow the flange welds to be deferred to the end of the interval and examined only once. This would coincide with the requirements for all the remaining RPV shell and head welds.

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST No. IR-045
Code Case N-623

In the case of the shell-to-flange weld, the partial exams are performed from the flange surface following disassembly or prior to re-assembly, while the reactor cavity is drained down. Thus, in the typical outage where critical path is through the refuel floor, these exams become a critical path activity. Duration to complete the partial shell-to-flange weld exams is approximately 4 hours for 2 NDE technicians. Dose rates at the RPV flange with the cavity drained range from 50-300 mr/hr. Therefore, elimination of the shell-to-flange weld partial exams would save approximately 4 hours of critical path time and a cumulative dose of up to 2.4 REM.

Technical Justification:

Recently, in order to support development of Boiling Water Reactor Vessel and Internals Project report 05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," BWR plants were surveyed to provide RPV shell weld inspection results. Of the 34 BWRs, 16, including PNPP, responded. The survey results for shell-to-flange and head-to-flange welds are shown in Tables A-5 and A-6, respectively, of the report. Only 6 indications exceeding the acceptance criteria of IWB-3500 were identified. All six were characterized as subsurface indications and met the acceptance criteria of IWB-3600. As can be seen, the only indications that are being found are subsurface fabrication defects that are benign. Subsequent to the issuance of the BWRVIP-05 report, PNPP conducted its initial exams of the shell-to-flange weld from the shell surface at the end of its first inspection interval (i.e., during RFO6). Similar to the results of PNPP's preservice exams and partial exams completed in RFO2, the RFO6 exams found no reportable indications.

The shell-to-flange weld exams that were performed during RFO6 were performed utilizing GE's automated GERIS-2000 system. The exams covered essentially 100% of the Code-required volume. If partial exams of the shell-to-flange weld were to be performed in accordance with the scheduling requirements of Table IWB-2500-1, they would need to be performed in PNPP's upcoming refueling outage (RFO8). The value of performing manual partial exams of the shell-to-flange weld following the recent completion of comprehensive automated exams, especially in light of the industry experience to date, is extremely negligible.

Summary:

Because of the acceptable preservice and first inspection interval exams, the industry experience to date, and the comprehensive exams that are performed at the end of the interval, it is concluded that the specified requirements would result in hardship without a compensating increase in the level of quality or safety.

Sheet 3 of 3

Perry Nuclear Power Plant Unit 1
RELIEF REQUEST No. IR-045
Code Case N-623

V. Alternate Examination

The alternative requirements of Code Case N-623 will be used. The Code Case states that inspection of shell-to-flange and head-to-flange welds of a reactor vessel may be deferred to the end of the interval without conducting partial exams from the flange face provided the following conditions are met:

(a) No welded repair/replacement activities have ever been performed on the shell-to-flange or head-to-flange weld.

(b) Neither the shell-to-flange weld nor head-to-flange weld contains identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420(b).

(c) The vessel is not in the first inspection interval.

PNPP meets all the above conditions.