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PG&E Letter DCL-05-033

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
<u>ASME Section XI Inservice Inspection Program Relief Requests</u>

Dear Commissioners and Staff:

Reactor pressure vessel (RPV) and associated pipe weld examinations are required to be performed during the second Inservice Inspection (ISI) interval for Diablo Canyon Power Plant (DCPP) Units 1 and 2. The DCPP Unit 1 second ISI interval ends on December 31, 2005, and the DCPP Unit 2 second ISI interval ends on May 31, 2006. The DCPP Unit 1 RPV weld inspections will be completed during Unit 1 refueling outage thirteen (1R13), currently scheduled for Fall 2005. The DCPP Unit 2 RPV weld inspections will be completed during Unit 2 refueling outage thirteen (2R13), currently scheduled for Spring 2006. The RPV weld examinations for DCPP Units 1 and 2 will be conducted in accordance with the requirements of the 1989 Edition of the ASME Section XI Code, and Section XI, Appendix VIII, 1995 Edition, including the 1996 Addenda.

To facilitate the completion of the RPV weld examinations, Pacific Gas and Electric Company (PG&E) is requesting relief to permit the use of updated examination methodologies when performing the weld examinations. The bases for the requested relief for the RPV weld examinations are provided in relief requests #NDE-NSA, #NDE-DMW, #NDE-PWE, #NDE-SFW, #NDE-ASC, and #NDE-ECT for DCPP Units 1 and 2, provided in Enclosures 1 through 6, respectively.

In addition to the weld examinations discussed above, work will be performed on selected reactor coolant pumps, including replacement of existing bolting with studs and hydraulic nuts. PG&E is requesting relief to permit use of the hydraulic nuts in this application. The bases for the requested relief is detailed in relief request #REP-2, provided in Enclosure 7. Relief request #REP-2 will be used in the second ISI interval during 1R13 and 2R13, and may be carried forward into the third ISI interval. Approval of relief requests #NDE-NSA, #NDE-DMW, #NDE-PWE, #NDE-SFW, #NDE-ASC, #NDE-ECT, and #REP-2 are requested pursuant to the provisions of 10 CFR 50.55a(a)(3)(i).

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Relief request #PRS-3, originally submitted to the NRC in PG&E Letter DCL-96-199, dated November 19,1996, and approved by the NRC on October 15, 1998, adds one new ASME Code Class 1 open-end tail pipe at the newly installed Pressurizer Vacuum Refill Connection. Changes from revision 0 are noted in Enclosure 8 with revision bars. Approval of relief request #PRS-3, R1, is requested pursuant to the provisions of 10 CFR 50.55a(a)(3)(ii).

Enclosure 9 provides notification of PG&E's intent to use Regulatory Guide 1.147 conditionally approved Code Case N-648-1.

PG&E requests approval of the relief requests by October 1, 2005, in support of 1R13.

Sincerely,

James R. Becker
Vice President – Operations and Station Director

mjrm/4557/A0632940

Enclosures
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INSERVICE INSPECTION (ISI) RELIEF REQUEST #NDE-NSA

System/Component for Which Relief is Requested

Reactor vessel nozzle-to-shell welds. Use of ASME Code Case N-613-1 alternative examination volume for Examination Category B-D, Item B3.90, Reactor Vessel Nozzle-to-Vessel Welds subject to Appendix VIII, Supplement 7 examination.

ASME Section XI Code Requirements

1989 Edition without Addenda, Category B-D, Item B3.90, Reactor Vessel Nozzle-to-Shell Welds, Examination Requirements, specifies the examination volume detailed in Figure IWB-2500-7. Figure IWB-2500-7(a) details examination zones in barrel type nozzles joined by full penetration corner welds, as applicable to Diablo Canyon Power Plant Units 1 and 2. The requirement for base metal examination extends a distance of one-half the vessel shell thickness out from the widest part of the weld on each side of the weld.

Code Requirement from Which Relief is Requested

Relief is requested from using the one-half vessel shell thickness dimension to determine the volume of base metal examined on each side of the weld.

Basis for Relief Request

The examination volume for nozzle-to-vessel welds shown in Figure IWB-2500-7(a) extends far beyond the weld into the base material and is unnecessarily large. This extends the examination time and attendant radiation exposure to examination and support personnel significantly and results in no net increase in safety.

The large volume of base metal adjacent to the welds shown in Figure IWB-2500-7(a) was extensively examined during construction, Preservice Inspection, and Inservice Inspection during the first inspection interval, and found to be free of unacceptable flaws. The area of base metal beyond that in Pacific Gas and Electric Company's (PG&E) proposed alternative is not in the high residual stress region associated with the weld. If cracks were to initiate, they would occur in the high stress areas of the weld and heat-affected zones that are contained in the examination volume defined by ASME Code Case N-613-1, and would thus be subject to examination.

The examinations will be performance based in accordance with Section XI, Appendix VIII, Supplement 7, and are demonstrated in accordance with the Electric Power Research Institute Performance Demonstration Initiative Program. Use of these advanced ultrasonic examination techniques by personnel who have demonstrated proficiency will provide assurance that the reactor vessel nozzle-to-shell welds have remained free of service-related flaws, thus enhancing quality and assuring plant safety and reliability.

Proposed Alternative

PG&E proposes to use Code Case N-613-1, Figure 1, which requires 0.5 inch of base metal adjacent to the weld be examined in lieu of the one-half vessel shell thickness dimension shown in Figure IWB-2500-7(a). This alternative assures that the high residual stress region of the weld and heat affected zone is fully examined using procedures, personnel, and equipment that have been qualified by performance demonstration.

The activities included in this relief request are subject to third party review by the Authorized Nuclear Inservice Inspector.

Justification for Granting of Relief

The proposed alternative assures that the high residual stress region of the weld and heat affected zone, as shown in Code Case N-613-1, Figure 1, is fully examined using performance-based ultrasonic examination procedures by personnel and equipment that have been qualified by demonstration and provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative was approved for the V.C. Summer Station by NRC letter dated February 11, 2004 (ML040420386).

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second ISI intervals, during the Unit 1 and Unit 2 thirteenth refueling outages.

This is a new request based on Code Case N-613-1, approved by the ASME Main Committee August 20, 2002, and shown as approved in Draft Regulatory Guide (RG) DG-1125 (proposed Revision 14 of RG 1.147).

INSERVICE INSPECTION (ISI) RELIEF REQUEST #NDE-DMW

System/Component for Which Relief is Requested

Reactor vessel nozzle-to-safe end welds. All eight nozzle-to-safe end welds will be examined in each unit. Use of ASME Code Case N-695 with alternate depth sizing qualification criteria.

ASME Section XI Code Requirements

Examination Category R-A, Item R1.20 (formerly 1989 Edition without Addenda, Category B-F, Item B5.10, Reactor Vessel Nozzle-to-Safe End Butt Welds), specifies volumetric examination. The volumetric examination is to be conducted in accordance with Appendix VIII, Supplement 10, in the 1995 Edition with 1996 Addenda.

Code Requirement from Which Relief is Requested

Relief is requested from using Appendix VIII, Supplement 10, in the 1995 Edition with 1996 Addenda.

Basis for Relief Request

ASME Code Case N-695, "Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1," is shown as acceptable for use in Draft Regulatory Guide (RG) DG-1125 (Proposed Revision 14 of RG 1.147) dated April 2004. To date, although examination vendors have qualified for detection and length sizing on these welds, the examination vendors have not met the established root mean square error (RMSE) requirement for depth sizing. Pacific Gas and Electric Company's (PG&E) contracted examination vendor has demonstrated ability to meet the depth sizing qualification requirement with an RMSE of 0.189 inches instead of the 0.125 inches required by the Code Case.

Addition of the difference in allowable depth sizing tolerance from that actually demonstrated to the flaw depths measured will compensate for the possible variance in the measured depth.

Proposed Alternative

PG&E proposes to use Code Case N-695 with an RMSE of 0.189 inches instead of the 0.125 inches specified for depth sizing in the Code Case. In the event an indication is detected that requires depth sizing, the 0.064-inch difference between the required RMSE and the demonstrated RMSE (0.189 inches - 0.125 inches = 0.064 inches) will be added to the measured through-wall extent for comparison with applicable acceptance criteria. If the examination vendor demonstrates an improved depth sizing RMSE prior to the examination, the excess of that improved RMSE over the 0.125-inch RMSE requirement, if any, will be added to the measured value for comparison with applicable acceptance criteria.

The activities included in this relief request are subject to third party review by the Authorized Nuclear Inservice Inspector.

Justification for Granting of Relief

The proposed alternative assures that the nozzle-to-safe-end welds will be fully examined by procedures, personnel and equipment qualified by demonstration in all aspects except depth sizing. For depth sizing, the proposed addition of the difference between the qualified and demonstrated sizing tolerance to any flaw required to be sized compensates for the potential variation and provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative was approved for the V.C. Summer Station by NRC letter dated February 3, 2004 (ML040340450).

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second ISI intervals, during the Unit 1 and Unit 2 thirteenth refueling outages.

This is a new request based on Code Case N-695, shown as approved in Draft RG DG-1125 (Proposed Revision 14 of RG 1.147), and the examination vendors' best-demonstrated depth sizing performance.

INSERVICE INSPECTION (ISI) RELIEF REQUEST #NDE-PWE

System/Component for Which Relief is Requested

Reactor coolant pipe welds. Eight austenitic stainless steel safe end-to-pipe welds in each Unit. The stainless steel welds join wrought 316 SS to wrought 304 SS material and are similar metal welds. Examinations conducted from the inside surface using the remote mechanized reactor vessel examination tool. Use of ASME Code Case N-696 with alternate Depth Sizing qualification criteria.

ASME Section XI Code Requirements

Examination Category R-A, Item R1.20 (formerly 1989 Edition without Addenda, Category B-J, Item B9.11, Circumferential Welds), specifies volumetric examination. The volumetric examination is to be conducted in accordance with Appendix VIII, Supplement 2, in the 1995 Edition with 1996 Addenda. Proposed Appendix VIII, Supplement 14 (Code Case N-696) defines requirements for combined qualification of Supplements 2, 3 and 10 when the examinations are conducted from the inside surface.

Code Requirement from Which Relief is Requested

Relief is requested from using Appendix VIII, Supplement 2, in the 1995 Edition with 1996 Addenda for austenitic stainless steel reactor coolant pipe welds near the reactor vessel nozzle that are examined with the remote automated reactor vessel examination tool.

Basis for Relief Request

ASME Code Case N-696, "Qualification Requirements for Appendix VIII Piping Examinations Conducted From the Inside Surface, Section XI, Division 1," was passed by the ASME Main Committee on May 21, 2003, but has not yet been addressed in published Regulatory Guides or drafts. Code Case N-696 addresses the combined qualification for Supplement 10 in conjunction with Supplements 2 and 3 when examinations are conducted from the inside surface, commonly referred to as "Supplement 14." To date, although examination vendors have qualified for detection and length sizing on these welds, no examination vendors have met the established root mean squared error (RMSE) for depth sizing. Pacific Gas and Electric Company's (PG&E) contracted examination vendor has demonstrated ability for depth sizing qualification with an RMSE of 0.245 inches instead of the 0.125 inches required by the Code Case.

Addition of the difference in allowable depth sizing tolerance from that actually demonstrated to the estimated flaw depths measured, will compensate for the variance in the depth measured.

Proposed Alternative

PG&E proposes to use Code Case N-696 with an RMSE of 0.245 inches instead of the 0.125 inches specified for depth sizing in the Code Case. In the event an indication is detected that requires depth sizing, the 0.120-inch difference between the required RMSE and the demonstrated RMSE (0.245 inches – 0.125 inches = 0.120 inches) will be added to the estimated through-wall extent for comparison with applicable acceptance criteria. If the examination vendor demonstrates an improved depth sizing RMSE prior to the examination, the excess of that improved RMSE over the 0.125-inch RMSE requirement, if any, will be added to the measured value for comparison with applicable acceptance criteria.

The activities included in this relief request are subject to third party review by the Authorized Nuclear Inservice Inspector.

Justification for Granting of Relief

The proposed alternative assures that the subject reactor coolant piping circumferential welds will be fully examined by procedures, personnel, and equipment qualified by demonstration to Code Case N-696 requirements in all aspects except depth sizing. For depth sizing, the difference between the qualified and demonstrated sizing tolerance will be added to any flaw required to be sized to compensate for the variance in depth sizing as demonstrated by the examination vendor. Use of Code Case N-696 with the stated compensation for depth sizing qualification tolerance provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i). Use of the combined qualification requirements for Supplements 2, 3, and 10 prior to availability of Code Case N-696, and the concept of adding the difference between the required RMSE value and the demonstrated RMSE value to the measured indication depth, were separately approved for the V.C. Summer Station by NRC letter dated February 3, 2004 (ML040340450).

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second ISI intervals, during the Unit 1 and Unit 2 thirteenth refueling outages.

This is a new request based on Code Case N-696, and the examination vendors' best-demonstrated depth sizing performance.

INSERVICE INSPECTION (ISI) RELIEF REQUEST #NDE-SFW

System/Component for Which Relief is Requested

Reactor vessel shell-to-flange weld.

ASME Section XI Code Requirements

1989 Edition without Addenda, Table IWB-2500-1, Category B-A, Item B1.30, requires that the reactor vessel shell-to-flange weld be volumetrically examined once during the interval. Essentially 100 percent of the weld length is to be examined in accordance with Appendix I, I-2100. Paragraph I-2100 requires, "Ultrasonic examination of vessel welds greater than 2 in. thickness shall be conducted in accordance with Article 4 of Section V, as supplemented by this Appendix. Supplements identified in Table I-2000-1 shall be applied." Table I-2000-1 identifies Supplements 2-12, inclusive, as being required.

Additionally, Regulatory Guide 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations" augments the ASME Section V and XI Code requirements.

Code Requirement from Which Relief is Requested

Relief is requested from using the techniques of Section V, Article 4, as supplemented by Section XI, Appendix I, and augmented by Regulatory Guide (RG) 1.150, Revision 1, when performing volumetric examination of the reactor vessel shell-to-flange weld.

Basis for Relief Request

The prescriptive, amplitude-based ultrasonic examination techniques of Section V, Article 4, supplemented by Appendix I, and augmented by RG 1.150, Revision 1, are technically inferior to the performance-based techniques specified in the 1995 Edition with 1996 Addenda of Section XI, Appendix VIII, Supplements 4 and 6, as modified by 10 CFR 50.55a(b)(2)(xv), and demonstrated through the Electric Power Research Institute (EPRI) Performance Demonstration Initiative (PDI) Program. The performance-based techniques of Appendix VIII are required for all other Reactor Vessel Shell Weld examinations, having replaced the Article 4 - Appendix I - RG 1.150, Revision 1, techniques.

The performance-based techniques of Appendix VIII offer several performance enhancements over the prescriptive amplitude-based techniques: (a) increased sensitivity to flaws, (b) demonstrated flaw measurement capability using amplitude-independent sizing techniques, (c) compatibility of the Appendix VIII examination technique with DCPP shell-to-flange weld geometry resulting in good ultrasonic beam coverage. Additionally, radiation exposure to the examiners and support personnel will be reduced because different examination devices will not have to be installed on the reactor vessel inspection robot just to perform the shell-to-flange weld examination.

(a) Increased sensitivity to flaws: The Appendix VIII procedure is more sensitive to flaws because the examination sensitivity level compares to an ASME distance amplitude correction (DAC) level of 5 to 10 percent, the highest practical level for ultrasonic testing. Examinations in accordance with Section V are conducted at 50 percent DAC for the outer 80 percent of wall thickness and 20 percent DAC for the inner 20 percent of wall thickness. The Appendix VIII procedure requires all signals interpreted by the analyst as flaws to be measured and assessed in accordance with the applicable acceptance criteria, regardless of amplitude, recognizing that some flaws can exhibit a low amplitude response depending on orientation. The Section V techniques traditionally have a flaw response cut-off point of 20 percent DAC.

(b) Demonstrated flaw measurement capability using amplitude-independent sizing techniques: The procedure for the proposed shell-to-flange weld examination has been demonstrated in accordance with ASME Section XI, Appendix VIII, Supplements 4 and 6, to the EPRI PDI.

The proposed procedure complies with ASME Code, Section XI, 1995 Edition with 1996 Addenda, as modified by 10 CFR 50.55a. The procedure has been qualified by time-based sizing techniques such as tip diffraction, rather than the amplitude-based ASME Section V techniques that have been proven inaccurate.

(c) Compatibility of the Appendix VIII examination technique with DCPP shell-to-flange weld geometry and previous examination history. The proposed Appendix VIII shell weld examination procedure will use the 45-degree beam angle in four orthogonal directions applied to the weld and volume by various transducer types each covering a specified depth range. The increment size will be 0.5 inches and examination will be conducted to the maximum extent practical. When these examinations are combined with the manual examination performed from the flange seal surface, the coverage is expected to exceed 90 percent.

The previous remote mechanized examination of the shell-to-flange weld was conducted in 1992 for Unit 1, and 1993 for Unit 2. At that time, 45, 60 and 70-degree exam angles were used. Results were acquired and analyzed using an automated ultrasonic exam system with no indications found exceeding the allowable limits of Section XI. There is excellent data archival from the previous exams so that, if necessary, comparison could be made with the Appendix VIII examinations if questions arise concerning indications.

Proposed Alternative

Pacific Gas and Electric Company (PG&E) proposes using qualified personnel and procedures for remote mechanized examination in accordance with the 1995 Edition with 1996 Addenda of the ASME Code, Section XI, Appendix VIII, Supplements 4 and 6, as modified by 10 CFR 50.55a(b)(2)(xv), and demonstrated by the EPRI PDI Program, for the reactor vessel shell-to-flange weld in lieu of volumetric examination in accordance with Article 4 of Section V, as supplemented by Appendix I of Section XI, and augmented by RG 1.150, Revision 1. These examinations will be conducted to the maximum extent practical and are subject to third party review by the Authorized Nuclear Inservice Inspector.

Justification for Granting of Relief

Use of UT procedures and personnel qualified to the 1995 Edition with 1996 Addenda of Section XI of the ASME Code, Appendix VIII, Supplements 4 and 6, as modified by 10 CFR 50.55a(b)(2)(xv) by demonstration through the EPRI PDI Program for the reactor vessel shell-to-flange weld provides an equivalent or better level of quality and safety than the current ASME Code and RG 1.150 requirements in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative was approved for the Surry Station by NRC letter dated December 8, 2004 (ML043510436).

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second ISI intervals, during the Unit 1 and Unit 2 thirteenth refueling outages.

This is a new request based on improved examination technology.

INSERVICE INSPECTION (ISI) RELIEF REQUEST #NDE-ASC

System/Component for Which Relief is Requested

Reactor Vessel Welds. Use of alternative sizing qualification criteria for Examination Category B-A, Items B1.10, Reactor Vessel Shell Welds and B1.20, Reactor Vessel Head Welds, subject to Appendix VIII, Supplement 4, examination. Also applies to Examination Category B-A, Item B1.30, Reactor Vessel Shell-to-Flange Weld when relief has been granted to use Section XI, Appendix VIII, for the reactor vessel shell-to-flange weld examination.

ASME Section XI Code Requirements

1995 Edition with 1996 Addenda, Appendix VIII, Supplement 4, Subparagraph 3.2, requires that (a) no flaw is undersized for depth by more than 0.2 in; (b) flaw lengths estimated by ultrasonics shall be the true length –1/4 in., +1 in.; and (c) three statistical parameters be applied to depth sizing estimates.

Code Requirement from Which Relief is Requested

Relief is requested from using Appendix VIII, Supplement 4, Subparagraph 3.2, in the 1995 Edition with 1996 Addenda.

Basis for Relief Request

Subparagraph 3.2(a) would be implemented together with Subparagraph 3.2(c). The three statistical parameters of Subparagraph 3.2(c) are inappropriate for analyzing the test data from the Supplement 4 performance demonstration. Subparagraph 3.2(c)(1) pertains to the slope of a linear regression line, the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15 percent through-wall. The differences between actual versus true value produce a tight grouping of results which resemble a shot gun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate acceptance criterion.

The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the Code is too lax with respect to evaluating flaw depths within the inner 15 percent of wall thickness.

The third parameter, 3.2(c)(3), pertains to a correlation coefficient, which is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

Therefore, use of Subparagraphs 3.2(a) and 3.2(c) is not appropriate for acceptance of depth-sizing estimates.

Similarly, Subparagraph 3.2(b) is too lax for length estimates in the inner 15 percent of wall thickness.

Proposed Alternative

In lieu of the sizing acceptance criteria in the 1995 Edition with 1996 Addenda of the ASME Code, Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2, PG&E proposes to use Appendix VIII, Supplement 4, Subparagraph 3.2, in the 2001 Edition with 2003 Addenda. This requires sizing performance demonstrations to satisfy the following criteria:

(a) The RMS error of the flaw lengths estimated by ultrasonics, as compared to the true lengths, shall not exceed 0.75 in.
(b) The RMS error of the flaw depths estimated by ultrasonics, as compared to the true depths, shall not exceed 0.15 in.

The activities included in this relief request are subject to third party review by the Authorized Nuclear Inservice Inspector.

Justification for Granting of Relief

10 CFR 50.55a(b)(2)(xv)(C) identifies Section XI, Appendix VIII, Supplement 4, Subparagraph 3.2 in the 1995 Edition with 1996 Addenda as being insufficient to assure the quality and safety of examinations. The alternative proposed by PG&E is the same as recommended in 10 CFR 50.55a(b)(2)(xv)(C) and is required by the 2003 Addenda of ASME Section XI approved and incorporated by reference in 10 CFR 50.55a. The proposed alternative provides an acceptable level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative was approved for the Prairie Island Station by NRC letter dated April 24, 2003 (ML031150063).

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second ISI intervals, during the Unit 1 and Unit 2 thirteenth refueling outages.

This is a new request based on improved examination technology and updated Code and regulatory requirements.

INSERVICE INSPECTION (ISI) RELIEF REQUEST #NDE-ECT

System/Component for Which Relief is Requested

Reactor coolant pipe welds. Nozzle-to-safe end and safe end-to-pipe welds examined from the inner diameter (ID) surface using the remote mechanized reactor vessel examination tool. Examination of the near surface volume of similar and dissimilar metal welds from the ID for axial flaws in the presence of surface roughness. Use of profilometry and eddy current examination methods to supplement ultrasonic examination qualified by performance demonstration.

ASME Section XI Code Requirements

Examination Category R-A, Item R1.20 (formerly 1989 Edition without Addenda, Category B-F, Item B5.10, Reactor Vessel Nozzle-to-Safe End Butt Welds or Category B-J, Item B9.11, Circumferential Welds) specifies volumetric examination. The volumetric examination is to be conducted in accordance with Appendix VIII, Supplement 10, or alternative requirements such as use of Supplement 2 in conjunction with Supplements 2, 3, and 10 (commonly referred to as proposed Supplement 14).

Code Requirement from Which Relief is Requested

Relief is requested from using only the ultrasonic method of Appendix VIII, Supplement 10 (or Supplement 2 in conjunction with Supplement 14), in the 1995 Edition with 1996 Addenda, when performing volumetric examination of the near surface of nozzle-to-safe end or safe end-to-pipe welds in the presence of surface roughness when the examination is conducted from the ID surface.

Basis for Relief Request

The examination vendor for Diablo Canyon Power Plant reactor vessel examinations has qualified for detection of circumferential flaws in accordance with Appendix VIII, Supplements 10 and 14, as demonstrated through the Electric Power Research Institute (EPRI) Performance Demonstration Initiative (PDI) Program, for nozzle-to-safe end and safe end-to-pipe welds examined from the ID surface. The vendor is similarly qualified for detection of axial flaws provided the surface is machined or ground smooth with no exposed root reinforcement or counterbore. Surface roughness may be present that could call into question the ultrasonic qualifications demonstrated for detection of axial flaws in the volume immediately under the surface.

The examination vendor has developed an eddy current technique to augment the ultrasonic examination method and provide increased sensitivity at the near surface. The eddy current technique was first used in the VC Summer reactor vessel primary nozzle examinations of 2000. The procedure was refined after its first use in 2000 by applying it to the VC Summer hot leg dissimilar metal weld section removed from service. The removed section had a number of primary water stress corrosion cracking flaws along with non-relevant indications resulting from metallurgical interface and surface geometry. Using these actual flaws and geometric conditions in the removed section to refine the technique, the vendor developed a reliable flaw-screening criteria which allowed for the successful use of the procedure in the VC Summer 2002 and 2003 examinations.

Since that time, the technique has been successfully blind tested for the Swedish authority SQC Kvalificeringscentrum AB (SQC NDT Qualification Center) under the program, "Qualification of Equipment, Procedure and Personnel for Detection, Characterization and Sizing of Defects in Areas in Nozzle to Safe End Welds at Ringhals Unit 3 and 4," Hakan Soderstrand 7-10-03. The important qualification parameters for Eddy Current in the SQC blind tests were as follows:

-Defect types: fatigue and stress corrosion cracks
-Tilt: +/-10 degrees; Skew: +/-10 degrees
-Detection target size: IDSCC 6mm (0.25 inches) long
-Flaw Location: within 10mm (13/32 inch)
-Length of the planar flaw within a 70% confidence level: +/-9mm (3/8 inch)
-False call rate: less than or equal to 20% for the personnel qualification tests
(Ref. SQC Qualification Report No. 019A/03)

The technique has also been used to supplement examination of portions of the relevant near-surface volumes during the last 10 domestic pressurized water reactor nozzle-to-pipe examinations conducted by the vendor.

Proposed Alternative

Pacific Gas and Electric Company (PG&E) proposes using surface geometry profiling software (profilometry) in conjunction with a focused immersion ultrasonic tranducer positioned to permit accurate profile data across the examination volume to help the examiner confirm locations where the raw data indicates lack of transducer contact due to problematic surface geometry. In addition to profilometry, eddy current examination will be used to supplement ultrasonic examination of the volume immediately under the surface for the nozzle-to-safe end or safe end-to-pipe welds having sufficient surface roughness to call into question the applicability of the ultrasonic examination qualification to detect axial flaws.

Profilometry will be used to determine the surface areas, if any, where roughness may limit the ability of ultrasonic methods to be used effectively as qualified through performance demonstration. The eddy current method will be used in the areas identified by profilometry to assure any axial flaws in the near surface volume that could be missed by ultrasonics due to potential surface roughness are detected.

As a compliment to ultrasonic examinations for rough surface detection coverage, the following eddy current techniques are utilized:

- up to two plus point probes applied circumferentially on the pipe inside surface in scan increments of 0.080 inches circumferentially (for axial flaws) and 0.25 inches axially.

- Automated systems for data collection and analysis

The target flaw size for the eddy current procedure is 0.28 inches long, well within the ASME Code linear flaw acceptance standards of 0.45 inches for austenitic material, and 0.625 inches for ferritic material (defined for the outside surface in the Code Tables).

All eight nozzle-to-safe end welds and all eight safe end-to-pipe welds will be examined. The ultrasonic examinations supplemented by eddy current examinations and profilometry will be conducted to the maximum extent practical and are subject to third party review by the Authorized Nuclear Inservice Inspector.

<u>Justification for Granting of Relief</u>

Use of ultrasonic profilometry and eddy current examination with procedures and personnel qualified through the SQC blind tests to supplement Appendix VIII qualified ultrasonic procedures and personnel for the nozzle-to-safe end and safe end-to-pipe welds provides additional assurance that surface-breaking flaws that may be present would be detected regardless of orientation or potential surface roughness, resulting in an equivalent or better level of quality and safety than that currently qualified to meet ASME Code requirements in accordance with 10 CFR 50.55a(a)(3)(i). The proposed alternative was approved for the V.C. Summer Station by NRC letter dated October 16, 2004 (ML042950444).

<u>Implementation Schedule</u>

This relief request will be implemented during the Unit 1 and Unit 2 second ISI intervals, during the Unit 1 and Unit 2 thirteenth refueling outages.

This is a new request based on alternative examination technology to compensate for potential limitations of the ultrasonic method alone, to detect near surface axial flaws in the presence of surface roughness.

INSERVICE INSPECTION (ISI) RELIEF REQUEST #REP-2

System/Component for Which Relief is Requested

Reactor coolant pump flange replacement bolting (ASME Code Class 1).

ASME Section III Code Requirements

Code Case N-579, "Use of Nonstandard Nuts, Class 1, 2, 3, MC, CS Components and Supports Construction Section III, Division 1," (generically approved in RG 1.84).

Code Requirement from Which Relief is Requested

Relief is requested from use of SA-194 material specified in Code Case N-579 for the nonstandard hydraulic nuts, conformance of thread configuration to ASME B1.1, and from involvement of a "Certificate Holder".

Basis for Relief Request

The reactor coolant pumps were constructed by Westinghouse Electro-Mechanical Division to ASME Section III, winter 1970 Addenda. The reactor coolant pump flanges are located in a high-radiation area inside the containment adjacent to the main coolant loops. The pump casing and main flange (cover) are SA-351 CF8.

The original 4.5-inch diameter SA-193 or SA-540 main flange bolts are being replaced with SA-540 studs. Hydraulic nuts are specified to assure consistent loading around the joint as well as to reduce personnel exposure by shortening time in the area. SA-540 Grade B23 Class 3 material, which is referenced in Section III paragraph NB-2128(a), as an acceptable material type for Class 1 bolting, will be used instead of SA-194 for manufacture of the hydraulic nuts.

Code Case N-579 requires the screw threads of nonstandard nuts be manufactured to meet the requirements for threads in ASME B1.1. While the inside threads of the hydraulic nuts conform to ASME B1.1, the outside threads will have a proprietary thread design developed by the vendor, Nova-Technofast, which minimizes thread deflection between the nut and lock ring, and thereby minimizes loss of pre-load.

Code Case N-579 also assigns responsibilities to a "Certificate Holder" assuming overall responsibility for the components or supports using the nonstandard nuts. In the case of the reactor coolant pumps, Pacific Gas and Electric Company (PG&E) is the entity having overall responsibility in lieu of a "Certificate Holder" and registered professional engineers on the DCPP staff will perform these functions.

Proposed Alternative

SA-540 Grade B23 material meeting the requirements for bolting material in Section III, paragraph NB-2128(a), will be used to fabricate the hydraulic nuts for the reactor coolant pump flanges instead of the SA-194 material specified in Code Case N-579. The hydraulic nuts will incorporate a proprietary outside thread design providing minimized thread deflection to maximize retained load and allow lower preload to be used in contrast to standard threads manufactured in accordance with ASME B1.1. All aspects of this design are approved by the original pump manufacturer. PG&E registered professional engineers acting under the DCPP quality assurance program will assume the responsibilities assigned to the "Certificate Holder" in Code Case N-579.

Justification for Granting of Relief

Use of SA-540 Grade B23 material as referenced by Section III paragraph NB-2128(a), in lieu of the SA-194 material specified in Code Case N-579, will assure adequate strength in the joint. The special thread design of the outside threads of the hydraulic nut minimizes thread deflection and loss of preload. Use of these nonstandard nuts is expected to maintain leak tightness of the joint while reducing radiation exposure to maintenance personnel. These advantages, and performance of "Certificate Holder" responsibilities by PG&E registered professional engineers, provides an equivalent level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(i). Use of the proposed alternative for the Diablo Canyon excess letdown heat exchanger flange (ASME Class 2) was approved by NRC letter dated July 29, 2004 (ML042120199).

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second and third ISI intervals. All bolts on a given pump will be replaced during one refueling outage, however, bolting may not be replaced simultaneously on all pumps. Pumps selected for replacement will be chosen based on maintenance needs and schedules. This is a new request based on approved Code Case N-579.

INSERVICE INSPECTION (ISI) RELIEF REQUEST #PRS-3, R1

Pressure Test Requirement for Which Relief is Requested

Eight ASME Code Class 1 closed end drain line segments, 26 ASME Code Class 1 open end tail pipes, and 4 ASME Code Class 2 open end tail pipes between first and second off manual isolation valves or between first off valve and blind flange or connection. Revision 1 adds one new ASME Code Class 1 open-end tail pipe at the newly installed Pressurizer Vacuum Refill Connection, for a total of 26.

ASME Section XI Code Requirements

1989 Edition, Table IWB-2500-1, Category B-P, Item B15.51; Table IWC-2500-1, Category C-H, Item C7.40 and Code Case N-498-1, requires that piping systems be subject to IWB-5222 tests or IWC-5222 tests at normal operating pressure once each inspection interval, during which visual examination VT-2 is conducted.

Code Requirement from Which Relief is Requested

Relief is requested from performing the IWB-5222 and IWC-5222 tests for certain line segments as described below.

Basis for Relief Request

These line segments between the manual isolation valves (or between the manual isolation valve and blind flange) serve as open or closed end drains, fill, vent or test lines. All of the segments are short, the closed end drains less than 18 inches and the open end segments less than 12 inches on average; and small diameter, being 3/4 inch NPS except for three at 1 inch NPS and four at 2 inch NPS. None of the isolation valves is capable of automatic closure. The line segments are not normally pressurized. Line pressure may exist due to first off valve leakby and thermal effects.

The Code 10-year pressure test (as required by Code Case N-498-1) is impractical, and relief is requested for the following reasons:

a) Using system pressure to test these line segments would require opening the first off manual valve in Mode 3 (hot standby) to pressurize between the two valves or valve and blind flange. However, pressure testing in this manner would result in violation of the Class 1 system requirement for double isolation valve protection.

b) For the closed end drains, costly system modifications would be required to break the system and install test connections with open-ended isolation valves at each location, with the concurrent unnecessary radiation exposure to personnel, in order to permit pressurization during Mode 6. Testing these closed end drain segments without modification would require defueling the reactor, reclosing and repressurizing the primary system, extending the outage critical path by approximately 10 days. Both these options constitute extreme hardships with no compensating increase in safety.

c) For the open-ended line segments, testing in Mode 6 without modification is possible because the lines are provided with test connections and isolation. However, pressurizing each line segment to the nominal reactor coolant system operating pressure would require use of a hydro pump at each of the locations. This would result in unnecessary radiation exposure to plant personnel and increase the risk of contaminated liquid spill. All of these locations are in high radiation areas. Staging the hydro pump, providing access, removing the pipe cap, opening the second off valve, filling and pressurizing the line segment, inspecting, depressurizing and restoring the system, securing the equipment and disposing of the effluent, is estimated by PG&E to require one man-rem at each location.

Proposed Alternative

Each line segment below will be visually inspected once during the 10-year system test, however the line segments will not be pressurized to full system pressure. Pressure may exist due to first off valve leak by and thermal effects. The Class 1 line segments are also observed each refueling outage during the system leakage test and the Class 2 line segments are also observed once each inspection period during the system inservice test. Note: Line numbers given refer to the main line that the subject segment is joined to. The small segments do not have individual line numbers.

Class	Size	Location	Description
1	3/4	line 2527 betwn vlvs 8364A & 283	RCP Lp 1 Seal Inj Drn RCDT
1	3/4	line 2534 betwn vlvs 8364B & 294	RCP Lp 2 Seal Inj Drn RCDT
1	3/4	line 2536 betwn vlvs 8364C & 303	RCP Lp 3 Seal Inj Drn RCDT
1	3/4	line 2541 betwn vlvs 8364D & 308	RCP Lp 4 Seal Inj Drn RCDT
1	3/4	segment between vlvs 513 & 514	Pzr Spray Drn to RCDT
1	2	segment betwn vlvs 8057A & 8058A	RCP Lp 1 Cld Lg Drn RCDT
1	2	segment betwn vlvs 8057B & 8058B	RCP Lp 2 Cld Lg Drn RCDT
1	2	segment betwn vlvs 8057C & 8058C	RCP Lp 3 Cld Lg Drn RCDT
1	3/4	line 109 betwn vlvs 579 &570	Hot Leg Recirc Vent
1	2	line 961 betw vlvs 8057D & 8066, 8058D	Lp 4 Cld Lg Drn (to 3/4")
1	3/4	RVRLIS connection between valve 8070 & blind flange	
1	3/4	line 14 Loop 2 spray line vent between valve 517 & 518	
1	3/4	line 14 Loop 2 spray line drain to RCDT between valve 515 & 516	
1	3/4	line 14 Loop 2 spray line drain to RCDT between valve 519 & 520	
1	3/4	line 13 Loop 1 spray line vent between valve 521 & 522	
1	3/4	line 13 Loop 1 spray line drain between valve 523 & 524	
1	3/4	line 1195 Pressurizer PORV vent betwn valve 8056 & blind flange	
1	3/4	line 1469 Pzr lp seal vent betwn valve 8052 & 8064A,8064B,8064C	
1	3/4	line 1495 RCP 1 seal bypass vent betwn valve 8362A & blind flng	
1	3/4	line 1496 RCP 2 seal bypass vent betwn valve 8362B & blind flng	
1	3/4	line 1497 RCP 3 seal bypass vent betwn valve 8362C & blind flng	
1	3/4	line 1498 RCP 4 seal bypass vent betwn valve 8362D & blind flng	
1	3/4	U2 ln 246 Charging line loop 4 vent between valve 100 & 572	
1	3/4	U2 ln 253 Accumulator inject loop 1 vent between valve 138 & 139	
1	3/4	U2 ln 254 Accumulator inject loop 2 vent between valve 140 & 141	
1	3/4	U2 ln 256 Accumulator inject loop 4 vent between valve 144 & 145	
1	3/4	line 235 Safety inject loop 1 hot leg vent between valve 50 & 51	
1	3/4	U2 ln 236 Safety inject loop 2 hot leg vent between valve 54 & 55	
1	3/4	line 237 Safety inject loop 3 hot leg vent between valve 58 & 59	
1	3/4	U2 ln 238 Safety inject loop 4 hot leg vent between valve 62 & 63	
1	3/4	line 109 Hot leg recirc vent between valve 6 & 935	
1	3/4	line 109 RHR loop 4 vlv 8702 thermal expn drain betw vlv 3, 4 & 7	
1	1	line 730 Pressurizer vacuum refill connection betw isolation vlvs	
2	1	Reactor vessel head vent between valve 8078B & 8078A	
2	1	Reactor vessel head vent between valve 8078C & 8078D	
2	3/4	RVRLIS hot leg instrument connection between valve 617 & 616	
2	3/4	React vessel head vent valve test conn betw test conn & valve 661	

Justification for Granting of Relief

The relief request is justified in accordance with 10 CFR 50.55a(a)(3) because:

a) The proposed alternative provides a reasonable assurance of continued structural integrity. These small, short line segments are normally not pressurized, except for any valve leakby and thermal effects that may cause pressurization. The proposed alternative visual examination will confirm the structural integrity of the line segments. During the 10-year system test, the line segments are expected to remain depressurized. If, however, the line segments pressurize due to valve leakby and thermal effects, the proposed alternate visual examination will essentially be identical to the Code-required VT-2 examination.

b) Compliance with the Code requirements would result in hardship and unusual difficulties without a compensating increase in the level of quality and safety. For the closed end drain line segments, PG&E would have to either (i) pressurize in Mode 3 which would involve an unreviewed safety question by defeating RCS double isolation, resulting in operation in a less conservative manner, (ii) add costly test connections with concurrent increase in potential failure points and unnecessary radiation exposure to plant personnel, or (iii) test with the reactor defueled and reclosed which would significantly increase outage critical path time to repressurize the reactor and would impose an unnecessary thermal cycle on the system.

   For the open-ended line segments, the possibility of testing in Mode 6 exists, however, multiple applications of hydro pumps would be required in high radiation areas with increased personnel exposure and the potential for contaminated liquid spill and increased radwaste generation.

c) The public health and safety is not compromised by this relief because the alternative visual examination provides an acceptable level of quality and safety.

Implementation Schedule

This relief request will be implemented during the Unit 1 and Unit 2 second Inservice Inspection intervals. The alternate visual examination is scheduled at or near the end of the interval, coincident with the 10-year system pressure test.

This request is essentially identical to pressure test reliefs 10, 11A, 11B, and 12 in the first ISI interval, approved in NRC letter dated September 21, 1992, and in NRC letter dated October 15, 1998, for the second ISI interval. Certain vent or drain lines have subsequently been removed and capped, and are not included in this request. Revision 1 adds the newly installed 1" diameter Pressurizer vacuum refill connection between isolation valves to this request.

NOTIFICATION OF INTENT TO USE CODE CASE N-648-1

This letter provides notification that Pacific Gas and Electric Company (PG&E) intends to use ASME Code Case N-648-1 for examination of the Reactor Vessel Primary Nozzle Inside Radius Section during the second Inservice Inspection interval at Diablo Canyon Units 1 and 2. Code Case N-648-1 is conditionally approved in RG 1.147, Revision 13 (reprinted with correction), dated January 2004. In using Code Case N-648-1, PG&E will incorporate all of the conditions stated in RG 1.147, as follows:

"In place of a UT examination, licensees may perform a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria of Table IWB-3512-1 with limiting assumptions on the flaw aspect ratio. The provisions of Table IWB-2500-1, Examination Category B-D, continue to apply except that, in place of examination volumes, the surfaces to be examined are the external surfaces shown in the figures applicable to this table."

The remote VT-1 visual examination system PG&E intends to use will be capable of 8x magnification and will be enhanced by procedural requirements for resolution of a 1-mil-width wire as a basis for sensitivity. All PG&E Inservice Inspection activities, including use of Code Case N-648-1 as described herein, are subject to third party review by an Authorized Nuclear Inservice Inspector.