October 26, 2005

Mr. David H. Oatley
Acting Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - APPROVAL OF REQUESTS FOR RELIEF FROM THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) BOILER AND PRESSURE VESSEL CODE (CODE) SECTION XI INSERVICE INSPECTION PROGRAM REQUIREMENTS (TAC NOS. MC6693 AND MC6694)

Dear Mr. Oatley:

By letter dated April 1, 2005, and as supplemented by letter dated July 7, 2005, Pacific Gas and Electric Company (PG&E) submitted Request for Relief Nos. NDE-NSA, NDE-DMW, NDE-PWE, NDE-SFW, NDE-ASC, NDE-ECT, REP-2, and PRS-3, Revision 1, from certain ASME Code Section XI inservice inspection (ISI) requirements for future inspections at Diablo Canyon Power Plant, Units 1 and 2. Subsequently, PG&E's letter dated July 7, 2005, withdrew Request for Relief Nos. NDE-ASC, and REP-2.

The Nuclear Regulatory Commission staff has completed its review of the above Requests for Relief and found them acceptable, as discussed in the enclosed staff's safety evaluation.

Sincerely,

/RA/
Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure: Safety Evaluation

cc w/encl: See next page

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ACCESSION NO.: ML052660331 **NRR-106**

OFFICE	PDIV-2/PM	PDIV-2/LA	EMCB-A/SC	EMCB-B/SC	OGC	PDIV-2/(A) SC
NAME	GShukla:sp	LFeizollahi	MMitchell	TChan	Zorn NLO w/comments	DCollins
DATE	10/3/05	10-03-05	8/29/05	8/18/05	10/12/05	10/26/05

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF FROM THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS

(ASME) SECTION XI INSERVICE INSPECTION PROGRAM

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letter dated April 1, 2005, and as supplemented by letter dated July 7, 2005, Pacific Gas and Electric Company (PG&E, the licensee) submitted Request for Relief Nos. NDE-NSA, NDE-DMW, NDE-PWE, NDE-SFW, NDE-ASC, NDE-ECT, REP-2, and PRS-3, Revision 1, which seek relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI for Diablo Canyon Power Plant (DCPP), Units 1 and 2. Subsequently, PG&E's letter dated July 7, 2005, withdrew Request for Reliefs Nos. NDE-ASC, and REP-2.

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of the ASME Code Class 1, Class 2, and Class 3 components shall be performed in accordance with Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” of the ASME Code and applicable editions and addenda as required by Paragraph 50.55a(g) of Title 10 of the *Code of Federal Regulations* (10 CFR), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Paragraph 50.55a(a)(3) of 10 CFR states in part that alternatives to the requirements of paragraph (g) may be used, when authorized by the Commission, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The components (including

supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval. The applicable ASME Code of record at DCPP, Units 1 and 2, for the second 10-year ISI interval is the 1989 Edition of ASME Section XI with no addenda, for Section XI except Appendix VIII. The applicable ASME Code of record for Appendix VIII is the 1995 Edition of ASME Section XI with the 1996 addenda.

3.0 TECHNICAL EVALUATION

The information provided by the licensee in support of the requests for relief from ASME Code requirements has been evaluated by the staff, as discussed below.

3.1 Request for Relief No. NDE-NSA: Examination Category B-D, Item B3.90, Full Penetration Welded Nozzles in Vessels

ASME Code Requirement: Examination Category B-D, Item B3.90 requires 100 percent volumetric examination, as defined in Figure IWB-2500-7(a), of reactor pressure vessel (RPV) nozzle-to-shell welds during each inspection interval. The requirement for examining adjacent base metal extends a distance of one-half the vessel shell wall thickness from the widest part of the weld, on each side of the weld.

Licensee's Proposed Alternative to the ASME Code: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed to use the alternative contained in ASME Code Case N-613-1, "Ultrasonic Examination of Full Penetration Nozzles in Vessels, Examination Category B-D, Item Nos. B3.10 and B3.90, Reactor Nozzle-to-Vessel Welds, Figures IWB-2500-7(a), (b), and (c)", which defines new base metal examination volumes for RPV nozzle-to-vessel welds. This request for relief applies to both DCPP, Units 1 and 2, nozzle-to-vessel weld examinations.

Licensee's Basis for Relief (as stated):

> 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.

Licensee's Proposed Alternative Examination (as stated):

> 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 [are] fully examined using procedures, personnel, and equipment that have been qualified by performance demonstration.
>
> The activities included in this Request for Relief are subject to third party review by the Authorized Nuclear Inservice Inspector.

Staff's Evaluation

The ASME Code requires 100 percent volumetric examination of RPV nozzle-to-shell welds during each inspection interval. The requirement includes examining the weld and adjacent base metal volume that extends a distance of one-half the vessel shell wall thickness from the widest part of the weld, on both the nozzle and shell sides of the weld. The licensee proposed to use Code Case N-613-1, which requires only 0.5 inch of base metal adjacent to the weld be examined in lieu of the one-half vessel shell thickness dimension required by the ASME Code. This Code Case has not been approved for general use in Regulatory Guide 1.147, Revision 13, June 2003.

RPV nozzle-to-vessel welds were required to be fabricated under a rigorous quality assurance program and radiographic examinations were applied to provide a basis for acceptance. Subsequent preservice and inservice ultrasonic examinations have been conducted on these welds and adjacent base material in accordance with the ASME Section XI ISI program requirements. All examinations conducted on these nozzles have shown the extended ASME base metal volume to be free of unacceptable flaws.

Service-induced cracks, should they initiate, are expected to occur in the most highly stressed region of these nozzle-to-vessel weldments, which is the weld and heat affected zone (HAZ). The HAZ in these thick-walled ferritic structures only extends a few millimeters from the fusion line of the weld into the adjacent base metal. As indicated by the licensee, these areas are contained within the reduced examination volume shown in Figure 1 of Code Case N-613-1.

The Staff concludes that, should significant degradation be occurring in the subject nozzle-to-vessel welds, there is reasonable assurance that it will be detected by the examinations proposed by the licensee. For this reason, the licensee’s proposal provides an acceptable level of quality and safety, therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the use of Code Case N-613-1 is authorized for the second ISI

interval at Diablo Canyon, Units 1 and 2, or until such time as Code Case N-613-1 is referenced in Regulatory Guide 1.147. At that time, if the licensee intends to continue to use this Code Case, the licensee must follow all provisions of Code Case N-613-1 with limitations or conditions specified in Regulatory Guide 1.147, if any. As an alternative to the ASME Code, the licensee must use the Code Case in its entirety.

3.2 Request for Relief No. NDE-DMW: Flaw-Sizing Error Limitations for Pressure Retaining Welds in Piping Examined from the Inside Surface of Pressurized-Water Reactors (PWR) Subject to Appendix VIII, Supplement 10, Qualification Requirements for Dissimilar Metal Piping Welds

ASME Code Requirement: Performance demonstration requirements for qualifying procedures, personnel and equipment to inspect dissimilar metal piping welds are listed in the 1995 Edition/1996 Addenda of ASME Section XI, Appendix VIII, Supplement 10. The acceptance criteria for flaw depth-sizing state that depth-sizing is acceptable, if the Root Mean Square (RMS) error is less than 0.125 inches.

Licensee's Proposed Alternative to the ASME Code: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative to the depth-sizing acceptance criteria outlined in the 1995 Edition/1996 Addenda of the ASME Code, Section XI, Appendix VIII, Supplement 10. This proposal is to use an RMS error of 0.189 inches in lieu of the 0.125 inches required by the ASME Code.

PG&E proposes to use Code Case N-695 with an RMSE [root mean square error] 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.

Licensee's Basis for Relief (as stated):

> 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).
>
> The activities included in this Request for Relief are subject to third party review by the Authorized Nuclear Inservice Inspector.

Staff's Evaluation

The *Code of Federal Regulations*, specifically 10 CFR 50.55a(g)(6)(ii)(C)(2), requires implementation of Appendix VIII, Supplement 10, as listed in the 1995 Edition, 1996 Addenda of the ASME Code, Section XI for qualification purposes. The current flaw depth-sizing requirements stated in Section XI, Appendix VIII, specify an acceptance criteria of 0.125 RMS error for successful qualification. The licensee proposes to use an RMS error value of 0.189 inch in lieu of the ASME Code-required value of 0.125 inches imposed by Appendix VIII, Supplement 10 and included in paragraph 3.2(b) of the Electric Power Research Institute (EPRI) Performance Demonstration Initiative (PDI) alternative. The proposed alternative applies to through-wall sizing of flaws identified during examinations of dissimilar metal welds from the inside surface of the component.

Supplement 10 requires that examination procedures, equipment, and personnel used for examination of dissimilar metal piping welds have a maximum error that is less than or equal to an RMS error value of 0.125 inches. The nuclear industry is in the process of qualifying personnel in accordance with Supplement 10 requirements, as implemented through the PDI program. However, no personnel that have attempted qualification to the requirements of Supplement 10 have achieved an RMS error value of 0.125 inches.

The licensee has stated that its vendor has only been able to achieve an RMS error value of 0.189 inches. As a result, the licensee is proposing to use a depth-sizing criterion of 0.189 inches as an acceptable RMS error value. The licensee also proposes to add the difference (0.064 inches) between the ASME Code-required RMS error (0.125 inches) and the demonstrated accuracy (0.189 inches) to the measurements acquired from flaw sizing.

The licensee's vendor has successfully completed the performance demonstration requirements for detection of all flaws as specified in Supplement 10, therefore the ability to detect inservice flaws complies with current 10 CFR 50.55a requirements. In addition, the licensee will supplement the ultrasonic examination with an eddy current inspection technique to enhance the ability to detect surface-breaking flaws. These examination capabilities provide reasonable assurance that significant degradation, should it occur, will be detected.

The licensee's alternative would only be used should a flaw be detected and the vendor is required to perform depth-sizing during the second interval inspections. Currently, no vendor has been able to comply with the ASME Code-required RMS error of 0.125 inches for examinations applied remotely from the inside surface of reactor pressure vessel (RPV) nozzle safe end welds. The performance of the vendor (with an RMS error of 0.189 inches), represents the current achievable state-of-practice for through-wall sizing from the inside surface of RPV nozzle safe-end dissimilar metal welds. For this reason, the Staff concludes that compliance with the ASME Code-required RMS error value is currently not feasible. Further, the licensee’s approach to add the difference between the ASME Code-required RMS error and the achieved performance demonstrated accuracy, to depth-sizing measurements for any flaws that are detected, in conjunction with acceptance standards specified in IWB-3500 of the ASME Code, provides an acceptable level of quality and safety.

Based upon the evaluation above, pursuant to 10 CFR 50.55a(a)(3)(i), Request for Relief No. NDE-DMW, is authorized for the second inspection interval. If an NDE sizing technique is demonstrated to be successful in meeting the RMS error requirements of Supplement 10, the licensee should implement the ASME Code requirement.

3.3 Request for Relief No. NDE-SFW: Examination Category B-A, Item B1.30, Pressure Retaining Welds in Reactor Vessel, Reactor Vessel Shell-to-Flange Weld

ASME Code Requirement: The 1989 Edition of ASME Code, Section XI, Appendix I, Subparagraph I-2110, requires that ultrasonic testing (UT) of reactor vessel shell-to-flange welds be conducted in accordance with Article 4 of ASME Code, Section V, supplemented by the requirements of Table I-2000-1. In addition, Regulatory Guide (RG) 1.150, Revision 1, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," serves as guidance for the UT examination of reactor pressure vessel (RPV) welds.

Licensee's Proposed Alternative to the ASME Code: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed using procedures and personnel qualified for remote mechanized examination in accordance with the 1995 Edition, 1996 Addenda of the ASME Code, Appendix VIII, Supplements, 4 and 6, in lieu of the techniques required by ASME, Section V, Article 4, when performing volumetric examination of the reactor vessel shell-to-flange weld.

Licensee's Basis for Relief (as stated):

> 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.

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.

Staff's Evaluation

The ASME Code requires that ultrasonic examination of shell- and head-to-flange welds in vessels of greater than 2 inches in thickness be conducted in accordance with Article 4 of ASME Section V, as supplemented by requirements in Table I-2000-1. ASME Section V, Article 4, provides a prescriptive-based process for qualifying UT procedures and performing examinations. The licensee proposed to use procedures and personnel qualified in accordance with performance-based criteria listed in the 1995 Edition, 1996 Addenda, of the ASME Code, Appendix VIII, Supplements 4 and 6, as implemented by the industry's Performance Demonstration Initiative (PDI) program. These performance-based methods are currently required by 10 CFR 50.55a for examination of RPV shell welds when scanning from the vessel shell surface. As for scanning from the flange surface, the licensee will follow the requirements of the current Code of record for the second ISI interval.

When prescriptive-based UT procedures that comply with the requirements of Article 4 of ASME Section V have been used in round robin tests containing real flaws in mockups of reactor vessels, and the results are statistically analyzed according to the screening criteria of ASME Code, Section XI, Appendix VIII, the procedures are less effective than examinations that utilize Appendix VIII, Supplements 4 and 6 qualified procedures. Performance-based ultrasonics is generally applied with higher sensitivity, which increases the probability of detecting a flaw when compared to prescriptive-based Section V, Article 4 requirements. Also, flaw sizing is more accurately determined with the time-based tip diffraction criteria used by performance-based ultrasonics than with the less accurate amplitude criteria for prescriptive-based Section V, Article 4 requirements. Procedures, equipment, and personnel qualified through the PDI program have demonstrated their skill level to detect flaws common to nuclear power plants and have shown high probability of detection levels. This has resulted in an increased reliability of inspections for weld configurations subject to the requirements of Appendix VIII.

The licensee proposed an alternative to use ultrasonic procedures and personnel qualified in the 1995 Edition, including the 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). The Staff concludes that the proposed alternative with PDI-qualified procedures and personnel applied from the vessel shell surface will provide superior examination results to those that would be realized from the prescriptive requirements described in ASME Section V, as augmented by RG 1.150 recommendations. The licensee's proposed alternative provides reasonable assurance that, if significant degradation is occurring on the RPV shell-to-flange weld, it will be detected, thus the licensee's proposed alternative provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's alternative is authorized for the second interval at DCPP, Units 1 and 2.

3.4 Request for Relief No. NDE-ASC: Examination Category B-A, Items B1.10, B1.20, and B1.30, Pressure Retaining Welds in Reactor Vessel, Shell, Head and Shell-to-Flange Welds

In response to an NRC Request for Additional Information (RAI), the licensee elected to withdraw request for relief No. NDE-ASC in a letter dated July 7, 2005, and therefore, must meet the requirements of 10 CFR 50.55a for the second intervals at DCPP, Units 1 and 2.

3.5 Request for Relief No. REP-2: Replacement of Nonstandard Bolts for the Reactor Coolant Pump Flange

In response to an NRC RAI, the licensee elected to withdraw request for relief No. REP 2 in a letter dated July 7, 2005, and therefore, must meet the requirements of 10 CFR 50.55a for the second intervals at DCPP, Units 1 and 2.

3.6 Request for Relief No. PRS-3: Revision 1, Examination Categories B-P and C-H, All Pressure Retaining Components

ASME Code Requirement: Examination Category B-P, Item B15.51, and Examination Category C-H, Item C7.40, require that a system hydrostatic test, as defined in Paragraphs IWB-5220 and IWC-5222, respectively, be performed on Class 1 and Class 2 piping systems following each reactor refueling outage prior to plant start-up. In addition, Paragraph IWB-5221 (a) requires the system hydrostatic test be conducted at a pressure not less than the pressure corresponding with 100 percent rated reactor power.

Licensee's Alternative to ASME Code: In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the requirement to perform a system hydrostatic test on a total of 8 Class 1 closed end drain line segments, 26 Class 1 open end small-bore pipes, and 4 Class 2 open end small-bore pipes between first and second manual isolation valves. The original request for Relief No. PRS-3 was approved by the NRC for the second 10-year interval in an SER dated October 15, 1998. This revision deletes certain vent and drain lines that have been capped subsequent to this approval, and adds one additional Class 1 small-bore pipe that has been newly installed on the pressurizer refill connection. As an alternative, the licensee proposes to visually inspect each pipe segment once during the 10-year system hydrostatic test, however the line segments will not be pressurized to full system pressure. The Class 1 line segments will be observed each refueling outage during the system leakage test and the Class 2 line segments will be observed once each inspection period during the system inservice test.

Licensee's Basis for Relief (as stated):

> 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 [nominal pipe size] except for three at 1 inch NPS and four at 2 inch NPS. None of the isolation valves are 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 [ASME] Code 10-year pressure test (as required by [ASME] 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.

Staff's Evaluation

The ASME Code requires that a system leakage test be performed at the end of each refueling outage, and when performed at or near the end of the interval, the test must include all Class 1 components within the reactor coolant system (RCS) boundary. In a letter dated October 15, 1998, the NRC-evaluated Request for Relief No. PRS-3 for DCPP, Units 1 and 2, which included the same segments, except that some are now either removed or capped, and only one new line has been added. In the initial evaluation of PRS-3 the NRC determined that the alternative visual examination proposed by the licensee should detect any leakage and that the licensee's alternative provides reasonable assurance of the continued operational readiness of the subject segments. A review of the new piping segment referenced in PRS-3, Revision 1, shows that this piping segment is the same in terms of diameter, function, material, and length as those piping segments for which an alterative has been authorized. For the licensee to conduct a pressure test on this pipe segment would require that the licensee modify the pressure boundary to the existing piping to accommodate fittings, replace valves, or other appurtenances needed to support this activity. Another option would be for the licensee to use a stand-alone hydrostatic pump connected to the subject piping between the first and second isolation valves to obtain a pressure equivalent to that during

normal RCS operation. Again, this may require modifications to the piping pressure boundary, and could potentially inject water into the primary system if pump pressure slightly exceeds normal RCS pressure. Either of these methods would result in a significant hardship for the licensee.

The Staff concludes that performance of hydrostatic tests on the subject piping segments, including the new pressurizer vacuum refill connection segment, in accordance with ASME Code requirements would require significant plant modifications and subject the licensee to an undue burden with no compensating increase in quality or safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee's revised alternative remain authorized for the second ISI interval at DCPP, Units 1 and 2.

3.7 Request for Relief No. NDE-PWE: Examination Category R-A, Item R1.20 welds (formerly 1989 Edition with no Addenda, Category B-J, Item B9.11, Circumferential Welds)

Components for Which Relief Is Requested

This request for relief is applicable only to eight austenitic stainless steel safe end-to-pipe welds in the primary reactor coolant system examined from the inside surface at each of the DCPP units.

Code Requirement

Examination Category R-A, Item R1.20 (formerly 1989 Edition with no 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 of the Code with 1996 Addenda. 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.

Performance demonstration requirements for qualifying procedures, personnel, and equipment to inspect austenitic stainless steel and dissimilar metal piping welds are listed in the 1995 Edition with 1996 Addenda of ASME Section XI, Appendix VIII, Supplements 2 and 10. Licensees may (1) elect to use the requirements of Supplement 2 as listed, (2) seek NRC approval for new ASME code cases currently being reviewed by Code Committees, or (3) propose an alternative to Code requirements. Paragraph 3.3(c) of the Electric Power Research Institute (EPRI) Performance Demonstration Initiatives (PDI) alternative states that personnel, equipment, and procedures are qualified for depth-sizing when the flaw depths estimated by ultrasonic examination, as compared to true flaw depths, do not exceed 0.125-inch root-mean-squared error (RMSE).

Code Requirement from which Relief Is Requested

Relief is requested from using the depth-sizing criterion of 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.

Licensee's Proposed Alternative:

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposes using the vendor achieved through-wall sizing value of 0.245-inch RMSE in the examination of Supplement 2 piping welds, in lieu of the 0.125-inch RMSE required by Paragraph 3.3(c) of the EPRI PDI alternative (proposed Supplement 14) for flaw depth-sizing qualification.

Licensee's Basis for Relief: (as stated)

> 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 square error (RMSE) for depth sizing. Pacific Gas and Electric Company's (PG&E) contracted examination vendor has demonstrated [the] 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.

Staff’s Evaluation

Supplement 10 of Appendix VIII to ASME Section XI requires that examination procedures, equipment, and personnel meet specific criteria for flaw depth-sizing accuracy. The Code specifies that the maximum error of flaw depth measurements, as compared to the true flaw depths, must be less than or equal to 0.125-inch RMSE.

In a letter dated July 7, 2005, in response to the NRC staff’s questions, the licensee stated that the RMSE is 0.367-inch for Supplement 2 qualification, 0.189-inch for Supplement 10 qualification, and is 0.245-inch when Supplement 2 and 10 are combined (Supplement 14). The licensee further stated that no vendor has successfully complied with the code-required RMSE values of 0.125 inches for qualification tests. The alternative Supplement 14 qualification was performed to Revision 2, which is the latest Supplement 14 as published in the 2004 Edition of the ASME Code, Section XI.

The nuclear industry is in the process of qualifying personnel in accordance with Supplements 2 and 10 requirements, as implemented through the PDI program. However, personnel have been unsuccessful in achieving the Code-required RMSE value for flaw depth-sizing demonstrations performed from the inside surface of a pipe

weld. At this time, achieving an RMSE value of 0.125 inches is impractical since no vendor has been able to comply with the Code-required value. The performance of the vendor with an RMSE of 0.245 inches, represents the current achievable state of practice for through-wall sizing from the inside surface of the reactor vessel nozzle. As a result, the licensee is proposing to use a depth-sizing criterion of 0.245 inches to size any detected flaw during the examination of the subject safe end-to-pipe welds. The licensee also proposes to add the difference (0.120 inches) between the Code-required RMSE (0.125 inches) and the demonstrated accuracy (0.245 inches) to the measurements acquired from flaw sizing.

The Code requirement of 0.125-inch RMSE for acceptable flaw depth-sizing error is for examinations of nozzle-to-safe end dissimilar metal welds and safe end-to-piping similar metal welds from the inside surfaces of the components using automated equipment. The demonstrated accuracy for these welds during performance qualifications was 0.245-inches. The licensee's practice of adding the difference between the Code-required RMSE and the demonstrated accuracy to the measurements acquired from sizing any detected flaws, in addition to the use of the acceptance standards specified in Section IWB-3500 of the Code, provides acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the remainder of the second 10-year ISI interval at DCPP, Units 1 and 2.

3.8 Request for Relief No. NDE-ECT: Examination Category R-A, Item R1.20 reactor coolant pipe welds (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)

Components for Which Relief Is Requested

This request for relief is applicable only to DCPP's reactor coolant pipe welds. Specifically, it is applicable to the nozzle-to-safe end and safe end-to-pipe welds examined from the inside diameter (ID) surface using the remote mechanized reactor vessel examination tool.

Code Requirement

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 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.

Licensee's Proposed Alternative

During the qualification process, ultrasonic examiners concluded that transducer contact could not be maintained in certain areas of the specimen during scanning for axial defects. In the procedure performance summary issued by the PDI, a limitation is noted for the detection of axial flaws in Supplement 10 field weld configurations.

The licensee proposes using surface geometry profiling software (profilometry) in conjunction with a focused immersion ultrasonic transducer 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 detection of any axial flaws near the surface.

Licensee's Basis for Relief: (as stated)

> 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 V[.]C[.] Summer reactor vessel primary nozzle examinations of 2000. The procedure was refined after its first use in 2000 by applying it to the V[.]C[.] 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 V[.]C[.] 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," at Hakan Soderstrand 7-10-03 [in 2003]. 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.

Staff's Evaluation

Supplement 10 of Appendix VIII to the ASME Code, Section XI requires that examination procedures, equipment, and personnel meet specific criteria as specified in Table VIII-S10-1 of Appendix VIII for both detection and false calls. However, a limitation was noted for detection of axial flaws where transducer contact could not be maintained in certain areas of the specimen.

As an alternative, the licensee will use a surface geometry profiling technique to identify locations that lack transducer contact. An eddy current technique will be used to supplement ultrasonic procedure at these locations. The eddy current probe will provide data from areas with irregular surface conditions. The licensee’s proposed alternative will use surface geometry profiling software in conjunction with a focused immersion ultrasonic transducer 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 rough surface geometry. In addition to profilometry, the licensee will use eddy current examination to supplement ultrasonic examination of the volume immediately under the surface of the nozzle-to-safe end or safe end-to-pipe welds with sufficient surface roughness which may challenge the applicability of the ultrasonic examination qualification to detect axial flaws. The technique proposed by the licensee has been demonstrated and blind tested by the industry in recent years. The staff finds that the licensee’s proposed alternative, which combines UT and eddy current techniques in examining axial flaws, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the proposed alternative for the remainder of the second 10-year ISI interval at DCPP, Units 1 and 2.

4.0 CONCLUSION

The DCPP, Units 1 and 2, Request for Relief Nos. NDE-NSA, NDE-DMW, NDE-PWE, NDE-SFW, NDE-ECT, and PRS-3, Revision 1, to the ASME Code requirements have been reviewed by the NRC staff, and based on the above evaluation, the staff has concluded that the

licensee's requests for relief provide an acceptable level of quality and safety, and, therefore, are acceptable, as discussed below.

For Request for Relief Nos. NDE-NSA, NDE-DMW, and NDE-SFW, the NRC staff concluded that the licensee’s proposed alternatives provide acceptable levels of quality and safety. Therefore, the licensee’s proposed alternatives are authorized pursuant to 10 CFR 50.55a (a)(3)(i) for the second 10-year ISI interval. For Request for Relief NDE-NSA, ASME Code Case N-613-1 is authorized for use until such time as the code case is referenced in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." At that time, if the licensee intends to continue to use this code case, the licensee must follow all provisions of ASME Code Case N-613-1 with limitations or conditions specified in RG 1.147, if any.

For Request for Relief Nos. NDE-ASC, and REP-2, the licensee elected to withdraw these requests for relief in a letter dated July 7, 2005, and therefore, must meet the requirements of 10 CFR 50.55a for the second intervals at DCPP, Units 1 and 2.

For Request for Relief No. PRS-3, Revision 1, the NRC staff concluded that the ASME Code-required 10-year system hydrostatic test is a significant hardship without a compensating increase in quality and safety. Furthermore, the licensee’s proposed alternative provides a reasonable assurance of leak tightness and structural integrity of the subject components. Therefore, the licensee’s proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year ISI interval.

For Request for Relief No. NDE-PWE, the NRC staff has reviewed the information provided and concludes that the licensee's practice of adding the difference between the Code-required RMSE and the demonstrated accuracy to the measurements acquired from sizing any detected flaws, in addition to the use of the acceptance standards specified in Section IWB-3500 of the Code, provides acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the proposed alternative for the remainder of the second 10-year ISI interval at DCPP, Units 1 and 2.

For Request for Relief No. NDE-ECT, the NRC staff has reviewed the information provided and finds that the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the proposed alternative for the remainder of the second 10-year ISI interval at DCPP, Units 1 and 2.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in these Requests for Relief remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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