UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 12, 2013

Mr. Edward D. Halpin
Senior Vice President and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 – RELIEF REQUEST NO. RVFLNG-INT3 – U1 & U2 - ALTERNATIVE TO ASME CODE, SECTION XI PRESSURE TEST REQUIREMENTS FOR CLASS 1 REACTOR VESSEL FLANGE LEAKOFF LINES (TAC NOS. MF0408 AND MF0409)

Dear Mr. Halpin:

By letters dated December 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12356A059), as supplemented by letter dated April 29, 2013 (ADAMS Accession No. ML13120A273), Pacific Gas and Electric Company (the licensee) requested U.S. Nuclear Regulatory Commission (NRC) review and authorization of relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for system leakage testing of the reactor pressure vessel head flange leakoff lines at Diablo Canyon Power Plant (DCPP), Units 1 and 2, for the third 10-year inservice inspection (ISI) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requested to use an alternative on the basis that complying with the system leakage requirements specified by ASME Code, Section XI, Table IWB-2500-1 and Paragraph IWB-5222 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and is in compliance with the requirements of the ASME Code, Section XI, for which relief was not requested. Therefore, the NRC staff authorizes use of the proposed alternative until the end of the third 10–year ISI interval at DCPP, Units 1 and 2, currently scheduled to end for Unit 1 on May 6, 2015, and for Unit 2 on March 12, 2016.

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

If you have any questions, please contact the Project Manager, Jennie Rankin, at 301-415-1530 or via e-mail at jennivine.rankin@nrc.gov.

Sincerely,

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. RVFLNG-INT3 - U1 & U2

PRESSURE TEST OF REACTOR VESSEL HEAD FLANGE LEAKOFF LINES

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By letters dated December 20, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12356A059), as supplemented by letter dated April 29, 2013 (ADAMS Accession No. ML13120A273), Pacific Gas and Electric Company (the licensee) submitted Relief Request No. RVFLNG-INT3 - U1 & U2 for U.S. Nuclear Regulatory Commission (NRC) review and authorization. The licensee requested relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for system leakage testing of the reactor pressure vessel (RPV) head flange leakoff lines at Diablo Canyon Power Plant (DCPP), Units 1 and 2, for the third 10-year inservice inspection (ISI) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requested to use an alternative on the basis that complying with the system leakage requirements specified by ASME Code, Section XI, Table IWB-2500-1, and Paragraph IWB-5222 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), "Inservice inspection requirements," ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the pre-service 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), "Standards approved for incorporation by reference," 12 months prior to the start of the 120-month inspection interval, subject to the conditions listed therein.

Enclosure

The regulations in 10 CFR 50.55a(a)(3), states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC if (i) The proposed alternatives would provide an acceptable level of quality and safety; or (ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on analysis of the regulatory requirements, the NRC staff concludes that the NRC has the regulatory authority to authorize the licensee's proposed alternative on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(a)(3)(ii).

3.0 TECHNICAL EVALUATION

3.1 Licensee's Request for Alternative

3.1.1 Components for which Relief is Being Requested

RPV head flange seal leakoff piping with a nominal pipe size (NPS) of 3/4 inches. The piping is classified as ASME Code Class 1; Examination Category, B-P; and Item Number, B15.10.

3.1.2 ASME Code Requirements

The Code of record for the DCPP third 10–year ISI interval that is scheduled to end for Unit 1 on May 6, 2015, and for Unit 2 on March 12, 2016, is the 2001 Edition through 2003 Addenda of the ASME Code, Section XI.

By letter dated December 20, 2012, the licensee stated, in part, that

> [ASME Code, Section XI] Table IWB-2500-1, Category B-P, Item B15.10 requires that all Class 1 pressure retaining components be subject to a system leakage test [in accordance with IWB-5220] with a Visual VT-2 examination every refueling outage. Paragraph IWB-5222 specifies that the system leakage test is performed at pressure corresponding to 100 percent rated reactor power. The pressure retaining boundary shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. For the system leakage test performed at or near the end of each inspection interval, the pressure retaining boundary shall extend to all Class 1 pressure retaining components within the system boundary.

3.1.3 Licensee's Reason for Request

By letter dated December 20, 2012, the licensee stated, in part, that

> The RPV Flange Seal Leakoff Piping consists of two NPS 3/4 inch stainless steel pipelines extending from connections at the reactor vessel flange to appropriate isolation valves on the outside of the missile shield wall. The first [leakoff] line

connects at the RPV flange outboard of and separated from normal operating reactor coolant pressure by the inner vessel O-ring seal which is itself compressed between the vessel and head flanges forming a leak-tight seal under normal system conditions. An outer O-ring seal is located outside of the tap in the vessel flange for this first [leakoff] line. Failure of the inner O-ring seal is the only condition under which this first [leakoff] line is pressurized. Therefore, this first [leakoff] line is not expected to be pressurized during the system pressure test following a refueling outage.

The second leakoff line connects at the RPV flange outboard of the outer O-ring seal. Failure of both O-ring seals is the only condition under which this second [leakoff] line is pressurized. Therefore, this second [leakoff] line is also not expected to be pressurized during the system pressure test following a refueling outage.

The inlet to these lines at the RPV flange is in each case a straight-bored hole 3/16 inch diameter through the flange cladding, normal to the plane of the flange, and extending through into a 0.742 inch internal diameter chamber welded under the cladding. The straight bore configuration of this [the 3/16 inch] opening to the [leakoff] piping precludes system pressure testing while the RPV head is removed because there is no provision to plug the orifice for testing using a pressure source on the pipe side, nor to install a pressure source connection to test from the flange side. Plugging or installing a connection would require machining threads in each flange opening with attendant concern over chips that may become a foreign material threat for fuel integrity or in the lines themselves. Additionally, machining would require extensive time in the estimated 20-40 millirem (mRem)/minute (min) field at the vessel flange, an as low as is reasonably achievable (ALARA) concern. After machining, installing and removing the plugs or pressure connections would require [additional] time for installation personnel in the estimated 20-40 mRem/min field, an additional ALARA concern. The licensee noted that in addition, the 3/16 inch plug or pressure connection itself would also introduce foreign materials exclusion issue at the edge of the open reactor vessel.

The [leakoff piping] configuration also precludes system pressure testing by pressurizing the lines externally with the RPV head installed. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips that are housed in recessed cavities in the flange face. If a pressure test of the first (inner) line were to be performed with the [RPV] head on, the inner O-ring would be pressurized in the opposite direction than its design function. The test pressure would result in an inward force on the inner O-ring that would tend to push it into the recessed cavity that houses the retainer clips. Therefore, based on engineering judgment, PG&E concluded that the thin O-ring material would likely be damaged by the inward force.

The second (outer) line cannot be pressurized because there is no O-ring seal outboard of the second line opening in the RPV flange, thus there is no means to retain pressure [unless a plug is installed at the opening].

> Machining the influent holes to accept plugs or pressure connections risks damage to the [piping] system and would subject personnel to significant radiation exposure.
>
> Purposely failing or not installing the inner O-ring in order to perform a pressure test of the outer O-ring would require an additional top head removal cycle to install a new functional O-ring set. The time associated with removing and reinstalling the RPV head to replace the O-rings would subject personnel to significant radiation exposure. Therefore, based on engineering judgment, PG&E concluded that compliance with IWB-5222 system pressure test requirement results in an unnecessary hardship without a compensating increase in the level of quality and safety.

By letter dated April 29, 2013, the licensee stated:

> The material specification of the affected piping is American Society for Testing and Materials (ASTM) A213 Grade TP 316 seamless stainless steel tubing, ¾ inch outside diameter by 0.095 inch wall thickness. Socket weld fittings are ASTM A182 Gr [Grade] F316 or ASTM A403 Gr WP316 stainless steel. Valves RCS-8069A and B are ASTM A182 Gr F316 forged stainless steel. Welds are stainless steel, currently specified as ER-316.
>
> Design pressure of the leakoff lines is 2500 psig [pounds per square inch gauge] at 650°F [degrees Fahrenheit].

The licensee also stated that the subject piping has no degradation history.

3.1.4 Licensee's Proposed Alternative and Basis for Use

As stated in the licensee's submittal dated December 20, 2012:

> In lieu of the requirements of IWB-5222, a VT-2 examination of the accessible areas (i.e., the portion of the O-ring seal leak-off lines between the reactor [biological] shield wall penetration and the isolation valves on the outside of the missile shield wall) will be performed each refueling outage while the refueling cavity is filled, imposing static head of approximately 12 pounds per square inch, gage on the piping during the examination.
>
> The portions of the [leakoff] lines between the reactor vessel flange and the reactor [biological] shield wall penetration are inaccessible when the reactor cavity is flooded. The inaccessible portions between the vessel insulation and the reactor shield wall penetration become accessible for direct examination when the cavity is drained and the excore [neutron detector instrument] manways are removed, as typically done to provide access for examination of the reactor nozzle-to-pipe welds. This portion of the lines will be VT-2 examined each outage in conjunction with the reactor nozzle-to-pipe welds examination or during

excore instrument maintenance although the lines will not be pressurized at that time because the refueling cavity is drained.

The short portion of the [leakoff] lines penetrating the [RPV] insulation and RPV flange are not directly accessible. However, surrounding areas and areas to which leakage would be channeled will be VT-2 examined each outage in conjunction with the above-mentioned examination of the portions between the [RPV] insulation and [reactor] shield wall that are accessible when the excore manways are removed. Again, the lines are not pressurized at this time because the refueling cavity is drained. If any through-wall leakage did occur in the RPV Flange seal leakoff piping, either during cavity flooding or pressurization following [reactor vessel] O-ring failure, it would result in an accumulation of boric acid that would be detected during the VT-2 visual examinations proposed in this request.

The flange seal leakoff piping is essentially a leakage detection/collection system and the inner line would only function as a Class 1 pressure boundary if the inner O-ring fails, or for the outer line if both inner and outer O-rings fail, thereby pressurizing the piping. If the inner O-ring should leak during the operating cycle it will be identified by an increase in temperature of the leak-off line above ambient temperature. This high temperature (greater than 120°F) would actuate an alarm in the Control Room, which would be closely monitored by procedurally controlled operator actions allowing identification of any further compensatory actions required. The leakage would be directed to and collected in the reactor coolant drain tank.

Additionally, a 3/8 inch diameter opening can be compensated by the normal make-up capability of the [reactor coolant] system. The 3/16 inch diameter size of the inlet to each of the leakoff lines is smaller than the [reactor coolant] system makeup capacity, thus this piping could be excluded from the reactor coolant pressure boundary on that basis. The small size of the inlet openings to the lines provides additional assurance that orderly shutdown and cooldown of the reactor coolant system would be achieved even in the event a through-wall flaw was to occur in the leakoff piping.

The proposed alternative examination is based on ASME Code Case N-805 which has not yet been approved by the Nuclear Regulatory Commission (NRC).

3.2 NRC Staff Evaluation

Licensees demonstrate the structural integrity and leak tightness of a piping system by performing system pressure tests in accordance with the ASME Code, Section XI, IWB-5000. As part of the system pressure tests, IWB-5221 and IWB-5222 define the pressure retaining boundary, the required pressure, and the visual examination requirements. The NRC staff evaluates the leakoff pipe system routing, scope of examination, examination method, leakage monitoring, structural integrity of the leakoff lines, and the resulting hardship or unusual difficulty if relief from the ASME Code requirements was not authorized.

The NRC staff concludes that using the normal operating pressure of the RPV to perform the required system pressure test on the leakoff lines in accordance with IWB-5222 would result in a hardship or difficulty as described without a compensating increase in the level of quality and safety.

By letter dated April 29, 2013, in response to the NRC staff's request for additional information (RAI) dated March 15, 2013 (ADAMS Accession No. ML13074A818), concerning identification of the scope of pipe segments that will be examined in a piping and instrumentation diagram (P&ID), the licensee stated, in part, that

> The pipe segments that are required to be examined in accordance with the ASME Code Section XI, Table IWB-2500-1, Examination Category B-P and Item Number B15.10 are identified on the attached P&IDs and extend from the connection below the vessel flange penetrations through 1 inch by ¾ inch reducers. The pipe segments continue down and across the biological shield annulus and through the BS [biological shield wall], then across and down around the reactor coolant loop area, down and through the MB [missile barrier wall], terminating at valves RCS-8069A and B.
>
> T**he pipe segments that are accessible for examination:** The pipe segments accessible for examination with the refueling cavity filled are those portions outside the BS [biological shield wall], i.e., across and down around the reactor coolant loop area, down and through the MS [missile barrier wall] to valves RCS-8069A and B. This portion constitutes the majority of the [leakoff] piping (approximately 90 percent) and is uninsulated except for the portions immediately adjacent to the MS [missile barrier wall] which includes valves RCS-8069A and B. On the P&ID, this portion extends from outside the BS [biological shield wall] through the MS [missile barrier wall] to valves RCS-8069A and B.
>
> **The pipe segments that are inaccessible for examination when the refueling cavity is filled:** The portion that is inaccessible [for examination] when pressurized (i.e., with the refueling cavity filled) extends from the connections at and through the vessel flange down and across the biological shield annulus and through the BS [biological shield wall] penetration. This portion is also uninsulated. On the P&ID, this is shown extending from the reactor vessel to inside the BS [biological shield wall].
>
> **The pipe segments that are accessible for examination when the refueling cavity is empty:** The pipe inside the BS [biological shield] annulus, including the lower portion of the downcomers from the vessel flange, the crossing of the annulus and the pipe inside the penetration sleeve is visible from inside the biological shield annulus when the refueling cavity is empty (i.e., the pipe is depressurized) and the access manways are opened.
>
> **The pipe segments that are inaccessible for examination at any time:** The vertical portion extending from the vessel flange connections down to the open biological shield annulus (approximate length of each line is about 5 1/2 feet)

> passes through the vessel flange and insulation. Then it is tightly fitted between the vessel insulation and the concrete and is not directly accessible at any time.

The NRC staff concludes that the licensee has satisfactorily identified the pipe segments that are accessible and inaccessible for visual examinations during the proposed system pressure (leakage) test.

By letter dated April 29, 2013, in response to the NRC staff's RAI dated March 15, 2013, concerning how the visual examination of the leakoff piping will be conducted, the licensee stated, in part, that

> For the accessible portion of the subject piping from the BS [biological shield wall] to the start of insulation inside the MB [missile barrier wall], the external exposed surfaces of the pipe shall be examined using the direct VT-2 method by certified examiners (VT-2 method is defined in ASME Code Section XI). When the surfaces are accessible for indirect or remote VT-2 visual examination, the pipe and the surrounding areas (including floor areas or equipment surfaces located underneath the piping) shall be examined from the floor elevation for evidence of leakage using the visual examination VT-2 method by certified VT-2 visual examiners.
>
> For the insulated portion on either side of the MB [missile barrier wall], insulation will not be removed. The horizontal surfaces of insulation shall be directly examined at each insulation joint. Surrounding areas (including floor areas or equipment surfaces located underneath the components and on the MB [missile barrier wall]) shall also be examined for evidence of leakage using the visual examination VT-2 method by certified VT-2 visual examiners.
>
> The portion of the pipe extending from the BS [biological shield wall] across the reactor coolant loop area is in a high elevation (approximately 24 feet above the floor), but an unobstructed view is available for the examiner from the floor elevation. The examiner shall perform remote VT-2 examination of this portion of the pipe and will additionally examine the floor and equipment surfaces underneath the pipe.

The NRC staff notes that IWA-5240 does not require insulation to be removed from piping when performing the VT-2 examination. Therefore, the NRC concludes it is acceptable that the licensee does not remove the insulation.

By letter dated April 29, 2013, in response to the NRC staff's RAI dated March 15, 2013, concerning how the operator distinguishes a pipe through-wall leakage from various other leakage sources (e.g., bolted joint [flange] leakage, valve inline leakage, and leakoff through the O-ring), the licensee stated, in part, that

> There are no bolted flanges or other mechanical connections in these lines, except the valves RCS-8069A and B packing glands. The portions of the pipes outside the BS [biological shield wall] are also relatively distant from other potential leakage sources. Accordingly, any leakage detected by certified VT-2

> visual examiners in the portions of piping outside the BS [biological shield wall] (except at the packing glands of valves RCS-8069A and B) shall be considered as through-wall leakage.
>
> The portion of piping from the connection under the reactor flange down and across the biological shield annulus and through the BS [biological shield wall] has no mechanical connections but is subject to potential contact by leakage from the refueling cavity seal. Cavity seal leakage, if present, would be deposited at low temperature and would be associated with dripping from above the pipe (including splash marks) continuing around and down the pipe to areas underneath. This can be distinguished from through wall leakage which would tend to be localized on the pipe surface and have no source from above. Through wall leakage resulting from O-ring failure would be characteristic of localized deposition at high temperature such as a "volcano cone" deposit.

The NRC staff concludes that the licensee will consider any leakage detected outside the biological shield wall (accessible pipe segments) as through-wall leakage from the leakoff piping. This requires the operator to take corrective actions. The NRC staff concludes that this is an adequate measure to monitor the potential leakage from the leakoff line. Although the licensee will be using 12 psig pressure instead of the RCS operating pressure to perform the system leakage test, the NRC staff concludes that the proposed alternative provides reasonable assurance of the structural integrity and leak tightness of the accessible leakoff lines.

By letter dated April 29, 2013, in response to the NRC staff's RAI dated March 15, 2013, concerning the system alignment during normal operation and how the operator is notified of a leak in order to take corrective actions, the licensee stated, in part, that

> The inboard leakoff line (S-22-3/4) is aligned with valve RCS-8069B normally open. Alignment is through valve RCS-8032 past [temperature sensor element,] TE-401 to the reactor coolant drain tank (RCDT).
>
> The outboard leakoff line (S-18-3/4) is isolated with valve RCS-8069A normally closed.
>
> These lines are not normally pressurized. During refueling outages with the refueling cavity flooded, the lines are pressurized due to cavity elevation head only.
>
> TE-401 provides an alarm in the control room when the line temperature exceeds 120°F.
>
> When leakage is detected via high temperature alarm initiated by instrument TE-401, valve RCS-8032 would be closed from the control room, the containment would be entered, and valve RCS-8069B would be closed by manual operator action, isolating the inboard leakoff line. Valve RCS-8069A would then be opened by manual operator action, aligning the outboard leakoff line to valve RCS-8032 and past TE-401 to the RCDT; then valve RCS-8032 would be reopened from the control room. The system would then be in

> configuration to monitor for leakage past the second O-ring seal. If a leakage past the second O-ring seal is detected via high temperature alarm initiated by instrument TE-401, the operator would be referred to Technical Specification 3.4.13 "RCS Operational LEAKAGE" for applicable required action(s), completion time(s) and compliance requirements.

By letter dated April 29, 2013, in response to the NRC staff's RAI dated March 15, 2013, concerning how the licensee determines if the leakoff line itself was leaking through-wall, once the temperature element alarm actuated, the licensee stated, in part, that

> The Reactor O-ring Seal Leakoff system is a closed system. Leakage due to degraded (leaking) leakoff lines (that divert leakoff to the containment) would be detectable by increased airborne radioactive particulate and radiation levels, increased containment humidity, and eventually by increased containment sump levels. This leakage is subject to limits stated for the reactor coolant system in the technical specifications.

The NRC staff concludes that the licensee has appropriate RCS leakage detection systems to detect potential leakage through leakoff pipe wall. Once the leakage is detected, the licensee has procedures to monitor and take corrective actions should the leakage exceed the allowable leakage rate in Technical Specification 3.4.13.

By letter dated April 29, 2013, in response to the NRC staff's RAI dated March 15, 2013, concerning how the structural integrity of those pipe segments that are inaccessible for visual examination could be demonstrated, the licensee stated, in part, that

> The portion of piping from the connection at the reactor flange down and across the biological shield annulus and through the BS [biological shield wall] is inaccessible when the refueling cavity is filled, and is only accessible when the cavity is drained and one or more of the excore instrument access manways are removed. These manways provide access to the reactor vessel nozzles annulus and the portion of the subject piping inside the BS [biological shield wall]. The lower portion of the downcomer from the connection below the reactor flange, the crossing of the biological shield annulus and the BS [biological shield wall] penetration is thus accessible for direct VT-2 examination only when the cavity is drained and the pipe is depressurized.

The licensee stated that it will examine the inaccessible pipe segments (the 5 1/2 feet length of the lines) by checking the bottom of the pipe and adjacent surfaces for evidence of leakage from the pipe above. In addition, the water that enters the pipe when the cavity is flooded is highly borated to refueling concentrations and any through-wall leakage will result in the formation of white crystals that will be detectable long after the piping is depressurized.

If the licensee were not able to detect a through-wall flaw on the leakoff lines by the system leakage test during refueling outages, both units have RCS leakage monitoring systems to detect any potential through-wall leakage from the inaccessible leakoff lines during normal operation. The operator would be notified by the temperature sensor alarm associated with the leakoff lines and alarms from the RCS leakage detection systems (e.g., radioactive particulate

and radiation levels, containment humidity sensors, or containment sump level). The NRC staff concludes that even if the structural integrity of the inaccessible pipe segments cannot be demonstrated by visual examinations, the licensee has measures to monitor the structural integrity of the inaccessible pipe segments.

In summary, the NRC staff concludes that the proposed alternative for the system leakage test will provide reasonable assurance of the structural integrity and leak tightness of the reactor vessel flange O-ring leakoff lines. In addition to the visual examinations of the pressurized leakoff lines to determine through-wall cracks, the licensee has the RCS leakage detection systems, procedures, and technical specification limits to monitor the condition of the leakoff lines.

4.0 CONCLUSION

Based on the above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the reactor vessel flange O-ring leakoff lines, and that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii) and is in compliance with the requirements of the ASME Code, Section XI, for which relief was not requested. Therefore, the NRC staff authorizes use of the proposed alternative until the end of the third 10–year ISI interval at DCPP, Units 1 and 2, currently scheduled to end for Unit 1 on May 6, 2015, and for Unit 2 on March 12, 2016.

All other ASME Code, Section XI requirements for which relief has not been specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: John Tsao

Date: September 12, 2013

If you have any questions, please contact the Project Manager, Jennie Rankin, at 301-415-1530 or via e-mail at jennivine.rankin@nrc.gov.

Sincerely,

/ ***RA*** /

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosure:
As stated

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