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July 19, 2010

PG&E Letter DCL-10-077

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20852

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Letter dated June 21, 2010, Request for Additional Information (Set 5) for the Diablo Canyon License Renewal Application

Dear Commissioners and Staff:

By letter dated November 23, 2009, Pacific Gas and Electric Company (PG&E) submitted an application to the U. S. Nuclear Regulatory Commission (NRC) for the renewal of Facility Operating Licenses DPR-80 and DPR-82, for Diablo Canyon Power Plant (DCPP) Units 1 and 2, respectively. The application included the license renewal application (LRA), and Applicant's Environmental Report – Operating License Renewal Stage.

PG&E's response to the request for additional information is included in Enclosure 1. LRA Amendment 3 resulting from the responses is included in Enclosure 2 showing the changed pages with line-in/line-out annotations.

PG&E makes the following commitments in LRA Table A4-1: (1) The Unit 2 gap repair work will be completed prior to the period of extended operation; (2) DCPP plant procedures will be revised to perform concrete inspections per ASME Section XI Subsection IWL within a 5-year interval; and (3) The DCPP work control procedure will be revised to include evaluation of reinforced concrete exposed during excavations.

If you have any questions regarding this response, please contact Mr. Terence L. Grebel, License Renewal Project Manager, at (805) 545-4160.

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NRR

I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 19, 2010.

Sincerely,

James R. Becker
Site Vice President

pns/50323785
Enclosures
cc: Diablo Distribution
cc/enc: Elmo E. Collins, NRC Region IV Regional Administrator
Nathanial B. Ferrer, NRC Project Manager, License Renewal
Kimberly J. Green, NRC Project Manager, License Renewal
Michael S. Peck, NRC Senior Resident Inspector
Alan B. Wang, NRC Project Manager, Office of Nuclear Reactor Regulation

**PG&E Response to NRC Letter dated June 21, 2010
Request for Additional Information (Set 5) for the
Diablo Canyon License Renewal Application**

<u>*RAI B2.1.27-1*</u>

NUREG-1801, "The Generic Aging Lessons Learned (GALL) Report," aging management program (AMP) XI.S1, "ASME Section XI, Subsection IWE," "operating experience" program element states:

> *ASME Section XI, Subsection IWE was incorporated into 10 CFR 50.55a in 1996. Prior to this time, operating experience pertaining to degradation of steel components of containment was gained through the inspections required by 10 CFR Part 50, Appendix J and ad hoc inspections conducted by licensees and the Nuclear Regulatory Commission (NRC). NRC Information Notices (INs) 86- 99, 88-82 and 89-79 described occurrences of corrosion in steel containment shells. NRC Generic Letter (GL) 87-05 addressed the potential for corrosion of boiling water reactor (BWR) Mark I steel drywells in the "sand pocket region." More recently, NRC IN 97-10 identified specific locations where concrete containments are susceptible to liner plate corrosion. The program is to consider the liner plate and containment shell corrosion concerns described in these generic communications. Implementation of the ISI requirements of Subsection IWE, in accordance with 10 CFR 50.55a, is a necessary element of aging management for steel components of steel and concrete containments through the period of extended operation.*

Program element 10 for the Diablo Canyon Nuclear Power Plant (DCPP) ASME Section XI, Subsection IWE AMP does not discuss operating experience related to NRC INs 89-79, 97-10, and 2004-09. In addition, program element 10 for the DCPP ASME Section XI, Subsection IWE AMP does not discuss operating experience related to liner plate corrosion recently identified at other operating plants.

1. *Describe potential effects of steel liner plate corrosion issues discussed in NRC INs 89-79, 97-10, and 2004-09 on the containment liners for DCPP Units 1 and 2.*

2. *Describe the potential effects of steel liner plate corrosion issues that recently occurred at other operating plants on the containment liners for DCPP Units 1 and 2.*

<u>PG&E Response to RAI B2.1.27-1</u>

1. NRC Information Notices 89-79, 97-10, and 2004-09 discuss containment liner corrosion events of differing severities that have occurred in Boiling Water Reactor drywells and suppression pools, Pressurized Water Reactor (PWR) ice condenser liners and PWR reinforced concrete structural liners such as Diablo Canyon Power Plant's (DCPP). The Diablo Canyon ASME Section XI,

Subsection IWE AMP containment liner inspection procedures, NDE VT 3-L, U1 & U2, "VT-3 Visual Examination Of the Containment Liner" and ISI VT GEN-1, U1 & U2, "General Visual Examination of the Containment Liner," specifically addresses inspection of the containment liner for corrosion and degraded liner surfaces. DCPP specific examinations have routinely detected minor surface irregularities. Additional inspection methods have been performed to determine the extent and origin, if possible, of the irregularities. This level of detection demonstrates that conditions and/or surface indications of liner degradation have a high probability of being detected and addressed thus ensuring the containment liner license renewal intended function is maintained. The periodic (40-month) inspection frequency has been specified by ASME Code as being sufficient to detect incipient indications of damage before it becomes widespread.

2. The potential effects of steel liner plate corrosion issues that recently occurred at other operating plants on the containment liners were evaluated for DCPP and the evaluation concluded that the current DCPP containment liner inspections are adequate given the limited occurrences of identified deterioration of operating plants. These inspections include:

    a) A visual examination of the containment liner plate and containment concrete in accordance with 10 CFR 50, Appendix J.

    b) Inspection of the containment coated surfaces to identify any liner plate degradation that would be evidenced by a degradation of the coating.

<u>*RAI B2.1.27-2*</u>

As mentioned in GALL Report AMP XI.S1, "ASME Section XI, Subsection IWE," "operating experience" program element, NRC IN 97-10 identified specific locations where concrete containments are susceptible to liner plate corrosion. The program is to consider the liner plate and containment shell corrosion concerns described in this generic communication.

During its review of plant-specific operating experience, the staff noted that the applicant indicates that gaps were identified in isolated spots along the liner plate and floor interface on 91'EL during the 2R15 outage for Unit 2. The applicant issued notifications documenting the issue. In one of the notifications, the applicant stated that no corrosion was found at the liner/concrete interface and the concrete was in good condition (no cracks or delaminations). However, the applicant recommended that it seal these gaps to prevent any liquid intrusion into the gaps and minimize the potential for corrosion of the carbon steel liner. Explain how the program will effectively manage aging of the carbon steel containment liner during the period of extended operation if permanent remediation by permanently sealing the gap between the liner plate and concrete is not completed.

<u>PG&E Response to RAI B2.1.27-2</u>

During the period of extended operation, PG&E will continue to perform inspections of the interface between the containment liner plate and concrete floor, in accordance with the requirements of AMP XI.S1, "ASME Section XI, Subsection IWE." Any identified areas of degradation will be evaluated as discussed in B2.1.27. AMP XI.S1 will continue to effectively manage aging of the carbon steel containment liner due to any gaps between the liner plate and concrete during the period of extended operation as discussed below.

1. <u>Unit 2</u>: The small gaps between the Unit 2 containment liner plate and concrete floor will be closed by the installation of sealant (caulking). This repair work is currently scheduled for Unit 2 refueling outage 16 (scheduled to start 5/2/2011).

2. <u>Unit 1</u>: PG&E is currently scheduled to perform an inspection of the Unit 1 containment liner plate during Unit 1 refueling outage 16 (scheduled to start 10/4/2010) to determine if similar conditions exist. Any identified degradation will be evaluated and, as appropriate, entered into the corrective action program.

The Unit 2 gap repair work will be completed prior to the period of extended operation. See revised License Renewal Application, Table A4-1, in Enclosure 2

RAI B2.1.28-1

GALL Report AMP XI.S2, "ASME Section XI, Subsection IWL" "acceptance criteria" program element states that ASME Section XI, Subsection IWL, Article IWL-3000 provides acceptance criteria for concrete containments. The GALL Report further states that quantitative acceptance criteria based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R may also be used to augment the qualitative assessment of the responsible engineer.

In its license renewal supporting documentation, the applicant states that its ASME Section XI, Subsection IWL program utilizes a three-tier acceptance process similar to that described in ACI 349.3R-96, and is identified in a plant procedure. According to this procedure, the third tier engineering evaluation criteria include the following thresholds:

a. Popouts and voids, less than or equal to 4 ft. in diameter or equivalent surface area.

b. Scaling less than or equal to 4 in. in depth.

c. Spalled areas less than or equal to 4 in. in depth and 4 ft. in any dimension.

d. For areas not around embedments and penetration outer plates, passive cracks less than or equal to 0.060 in. in maximum width.

e. No limitations on the length, orientation and depth of passive cracks.

f. No evidence of passive deflections.

g. No exposed reinforcing steel.

h. No excessive corrosion on the surface of embedments.

i. Cracks around the embedments and penetration outer plates less than 0.060 in. Any crack equal to or greater than 0.060 in. shall be evaluated on case by case basis.

j. No evidence of seepage or water intrusion into concrete.

The evaluation criteria in ACI 349.3R provides acceptance without further evaluation (tier one), acceptance after review (tier two), and conditions requiring further evaluation. The threshold for tier three engineering evaluations as described above is less stringent than the criteria specified in ACI 349.3R.

Provide a discussion of the basis for the tier three acceptance criteria described in the DCPP implementing procedure. This basis should include reference to any design calculations and industry codes and standards that establish the technical rationale for the tier three acceptance criteria.

PG&E Response to RAI B2.1.28-1

Procedure NDE VT 3C-1 contains a three-tier acceptance criterion which was developed for acceptance of the DCPP containments concrete surface conditions. The first two tiers are based on the requirements of ACI 349.3R-96, Sections 5.1 and 5.2. The third-tier criterion is based on the results of the engineering evaluation performed in PG&E Calculation No. 2305C, Revision 2 (which was provided to NRC staff for their review during an audit) for determining threshold levels (acceptable for continued operability).

All indications exceeding the first-tier criteria are identified, located, measured and recorded for acceptance by responsible professional engineer (RPE) using the second-tier acceptance criteria, or for further evaluation by the RPE.

All indications exceeding the second-tier criteria are evaluated for continued operability by the RPE using the third-tier criteria and/or any supplemental tests or measurements (to fully characterize the identified degraded condition).

NDE VT 3C-1 has been revised to accomplish two objectives:

1. Provide clarification as to exactly when a corrective action document is required to be written (when exceeding the first-tier criteria).

2. Preclude any confusion as to what indications are acceptable under design basis versus acceptability for continued operation by removing the third-tier criteria from the procedure.

RAI B2.1.28-2

LRA Section B2.1.28 states that DCPP operating experience is evaluated and corrective actions are implemented to ensure that the components of the ASME Section XI, Subsection IWL program are maintained.

During its audit, the staff reviewed structural concrete surface examination data for DCPP Units 1 and 2. These data indicate that DCPP Units 1 and 2 containments concrete surface condition at hundreds of locations exceeded the second tier evaluation criteria described in ACI 349.3R. In addition, at more than 10 locations the surface condition exceeded the DCPP inspection criteria for third tier indications. Pacific Gas and Electric Company (PG&E) determined that there is no apparent loss of structural capacity; however, as part of its process, PG&E states that Nuclear Services/Engineering Services/Design Engineering/Civil Engineering (NS/ES/DE/CE) shall assess the results of the examination for acceptance and evaluation.

The applicant is requested to provide the following information:

1. *A summary of the information in the Notifications issued by the responsible engineer for the tier three gross indications that exceeded the threshold limitations for Unit 1 and Unit 2.*

2. *A summary of acceptance and evaluation results for assessments performed by NS/ES/DE/CE for the tier three gross indications that exceeded the threshold limitations for Units 1 and 2.*

3. *Details of remedial and corrective actions that the applicant plans to implement to address aging management of tier two indications and areas of tier three degradation that do not conform to ACI 349.3R guidance during the period of extended operation.*

The staff needs the above information to confirm that the acceptance criteria used and evaluations performed by the DCPP for containment concrete surface degradations are in accordance with the recommendations of element 6 of GALL Report AMP XI.S2, "ASME Section XI, Subsection IWL."

PG&E Response to RAI B2.1.28-2

1. Concrete examinations were performed to meet inservice inspection requirements and evaluate the position of the concrete for the Units 1 and 2 containment structures. The intent of the examinations were to comply with 10 CFR 50.55a, which was amended in 1996, to incorporate requirements of Subsection IWL of ASME Boiler and Pressure Vessel Code, Section XI, which mandated this examination. The first interval inspections of Units 1 and 2 were performed from August 2000 to July 2001, including the October 2000 Unit 1

refueling outage (1R10) and the May 2001 Unit 2 refueling outage (2R10). The second interval inspections for Unit 2 were performed from April 2006 to August 2006, including the 2R13 (April 17, 2006 to May 25, 2006) refueling outage. The examinations consisted of a visual examination of 100 percent of the accessible exterior concrete surface of the containment structure. All tier-three indications for Units 1 and 2 are listed in PG&E Calculation No. 2305C, Revision 2, which was provided to NRC staff during an audit for their review. All indications found during the inservice inspections were acceptable and will not have an adverse effect on the structural integrity of the containment shell for both units. It is judged that the condition of the Units 1 and 2 concrete containments are structurally sound and meet the DCPP design basis requirements.

2. A summary of acceptance and evaluation results for assessments for the tier-three gross indications that exceeded the threshold limitations for Units 1 and 2 are provided in PG&E Calculation No. 2305C, Revision 2.

3. All tier-two indications and areas of third-tier degradation were evaluated, using the guidance of ACI 349.3R-96, as acceptable, and having no adverse effects on the structural integrity of the Unit 1 and 2 containments. In accordance with ACI 349.3R-96, repair or replacement was deemed not necessary, as it was determined that the as-found conditions of the structure do not adversely affect the licensing bases intended function. These indications and areas will continue to be monitored as part of XI.S2 ASME Section XI, Subsection IWL inspections.

RAI B2.1.28-3

GALL Report AMP XI.S2, "ASME Section XI, Subsection IWL" "detection of aging effects" program element states that the frequency and scope of examinations specified in 10 CFR 50.55a and Subsection IWL ensure that aging effects would be detected before they would compromise the design-basis requirements. The frequency of inspection is specified in IWL-2400.

In the LRA, the applicant stated that the DCPP ASME Section XI, Subsection IWL program is in accordance with IWL-2400. Each unit is examined on an alternating 10 year cycle as specified - 4 - in IWL-2421. Visual examinations of 100% of the accessible surfaces on the concrete shells will be completed on 10 year cycles for each unit (1 unit every 5 years). However, the 2001 edition of ASME Section XI, Subsection IWL-2410 states that concrete shall be examined in accordance with IWL-2510 at 1, 3, and 5 years following the completion of the containment Structural Integrity Test CC-6000 and every 5 years thereafter. In addition, the requirements in ASME Section XI, Subsection IWL-2421 only apply to sites with multiple units that have containments with unbonded post-tensioning systems. Therefore, it does not appear that the program is consistent with the recommendations in the GALL Report XI.S2 "detection of aging effects" program element.

1. *What is the basis for selecting the 10-year inspection frequency for each unit?*

2. *What is the impact of the use of 10-year inspection frequency on the DCPP Unit 1 and Unit 2 containment AMP, including detection of aging effects?*

PG&E Response to RAI B2.1.28-3

1. Due to an incorrect interpretation of ASME Section XI paragraphs, IWL-2410 and IWL-2421, the Unit 1 containment concrete inspection per Subsection IWL was not performed in the outage closest to 2005, as required. This issue was entered into the plant corrective action program for resolution. This issue does not apply to Unit 2, as the examinations for Unit 2 were completed as required. The Unit 1 exam is currently scheduled during the 16th refueling outage for Unit 1, which starts in October 2010. Containment in-service inspections were imposed in 10 CFR 50.55a beginning in 1996. At that time, licensees were required to perform the first containment concrete inspections per ASME Section XI Subsection IWL within 5 years. Exams must and will be conducted on a 5-year frequency for both units. The initial inspections were completed in 2000 for Unit 1 and in 2001 for Unit 2. Subsequent inspections were performed in Unit 2 in 2006, and the next Unit 2 inspection is scheduled for May 2011. The examinations for Unit 2 have progressed on the required schedule. Although the Unit 1 inspection was not conducted as required, significant testing of the containment structure has been performed in the surveillance interval, including the integrated leak rate test and the containment structural integrity test. There

were no adverse indications found during the performed tests. Based on the testing performed, it is reasonable to conclude that the containment structural function is adequately maintained.

2. Based on IWL XI.S2 Section XI, Subsection IWL inspection findings and local leak rate testing and integrated leak rate testing results to date, using a 10-year IWL inspection frequency has been adequate to maintain the containment structural safety function. Diablo Canyon Power Plant procedures will be revised to perform concrete inspections per ASME Section XI Subsection IWL within a 5-year interval. See revised License Renewal Application, Table A4-1, in Enclosure 2.

RAI B2.1.29-1

The DCPP ASME Section XI, Subsection IWF states that industry operating experience is evaluated for relevancy to DCPP, and appropriate actions are taken and documented.

It is not clear from the DCPP LRA if NRC IN 2009-04 related to constant supports has been considered in the operating experience.

Explain if the age-related degradation mechanism described in IN 2009-04 has been considered at DCPP.

PG&E Response to RAI B2.1.29-1

PG&E conducted an evaluation of NRC Information Notice (IN) 2009-04, Age Related Constant Support Degradation. A search of the plant equipment database for constant supports at Diablo Canyon Power Plant (DCPP) indicated the following population:

1. Ten constant supports located in Unit 1 on Design Class I safety injection piping inside the pressurizer cubicle; however, none on Unit 2.
2. One constant support on Unit 1, main steam line 585-28 inches (PG&E Design Class I, Code Class E).
3. Thirteen constant supports in each unit, main steam lines (Design Class II, Code Class E) in the turbine building.
4. Twelve vendor-supplied constant supports on cross over legs (MS lines, Design Class II, Code Class E) at 140-ft elevation (turbine building) in each unit.

The following are the differences between the events observed in IN 2009-04 and DCPP:

1. DCPP does not have any constant supports on main steam or feedwater lines inside the containment.
2. The only constant supports inside the containment are in the Unit 1 pressurizer on safety injection lines downstream of pressurizer relief valves 8010A, B, & C. There are no constant supports in the Unit 2 pressurizer. However, variable spring hangers are installed downstream of relief valves 8010A, B, & C.
3. Constant supports in the pressurizer are blocked every refueling outage by maintenance for refurbishment of relief valves 8010A, B, & C. No wear to date of linkages or decrease in support performance has been noticed by maintenance due to vibration.

4. Unit 1 Main steam Lead 4 has one constant support on the Design Class I, Code Class E portion of the piping, close to the 'G' line anchor in the 'GW' area. No significant vibration was noticed. Note: No constant supports in Unit 2 in 'GE' & 'GW' areas.

In response to IN 2009-04, walkdowns and visual inspections of all constant supports outside containment were performed in April 2010 for Units 1 and 2 to check for any wear on constant support linkages due to vibration. All supports appeared to be in good condition with no obvious wear on linkages. Vibration levels at these locations were judged to be extremely small, below ASME OM-3.

The evaluation of IN 2009-04 concluded that constant supports on DCPP piping supports are not experiencing excessive wear due to cyclic loading.

<u>*RAI B2.1.30-1*</u>

GALL Report AMP XI.S4, "10 CFR 50, Appendix J," "detection of aging effects" program element states:

> *A containment LRT program is effective in detecting degradation of containment shells, liners, and components that compromise the containment pressure boundary, including seals and gaskets. While the calculation of leakage rates demonstrates the leak-tightness and structural integrity of the containment, it does not by itself provide information that would indicate that aging degradation has initiated or that the capacity of the containment may have been reduced for other types of loads, such as seismic loading. This would be achieved with the additional implementation of an acceptable containment inservice inspection program as described in XI.S1 and XI.S2.*

In LRA AMP B2.1.30, the applicant stated that visual inspections of containment concrete surfaces outside containment and steel liner plate inside containment are required by 10 CFR 50, Appendix J to be performed prior to any Type A test. In addition, according to LRA AMP B2.1.30, "10 CFR 50, Appendix J," Element 10, Subsection 3.10.2, the most recent Type A test for Unit 1 was performed on March 17, 2009, and the most recent Type A test for Unit 2 was performed on April 4, 2008. However, it is not clear from the LRA how and when the general inspection of the containment concrete surfaces outside containment and steel liner plate inside containment were performed.

Confirm that the DCPP procedures for Type A test comply with the requirements of 10 CFR Part 50 Appendix J, which requires a general visual examination of the accessible interior and exterior surfaces of the containment system for structural deterioration prior to each Type A test. GALL Report AMP XI.S4, "10 CFR Part 50, Appendix J," recommends the use of 10 CFR Part 50 Appendix J for detecting age related degradation of containment.

<u>PG&E Response to RAI B2.1.30-1</u>

License Amendments 197/198, for Units 1 and 2, respectively, changed surveillance requirement SR 45.2.1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE. A visual examination of the interior and exterior surfaces of containment is also required prior to performing a Type A test, during the same outage that the Type A test will be performed. This pre-test visual examination of the interior and exterior surfaces of the containment is performed in accordance with STP M-7W and satisfies the pre-test inspection requirements for Type A tests, detailed in NEI 94-01, Rev. 0.

As discussed in the response to RAI B2.1.28-3, due to an incorrect interpretation of ASME Section XI paragraphs, IWL-2410 and IWL-2421, the Unit 1 containment concrete inspection per Subsection IWL was not performed in the outage closest to 2005, as required. This issue was entered into the Diablo Canyon Power Plant (DCPP) corrective action program for resolution. DCPP procedures will be revised to perform concrete inspections per ASME Section XI Subsection IWL within a 5-year interval. See revised license renewal application, Table A4-1 in Enclosure 2.

DCPP procedures for visual inspection of the containment steel liner plate, comply with the requirements of 10 CFR 50 Appendix J, further ensuring that proper and timely examinations will be conducted prior to each Type A test. Appropriately, the next scheduled inspection for Unit 1 is during the October 2010 refueling outage (1R16). Unit 2 examinations have been performed and progressed on the required schedule.

RAI B2.1.32-1

GALL Report XI.S6, "Structures Monitoring Program," notes that ACI 349.3R-96 provides an acceptable basis for developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes. The plant-specific structures monitoring program is to contain sufficient detail on acceptance criteria to conclude that this program attribute is satisfied.

The "acceptance criteria," program element of the DCPP Structures Monitoring Program references ACI 349.3R-96 as providing an acceptable basis for developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes. The DCPP Structure Monitoring Program uses "Acceptable," "Acceptable with Deficiencies," and "Unacceptable" categories. Although ACI 349.3R is referenced as providing the basis for the acceptance criteria, the staff is unclear what criteria are associated with each of the three acceptance criteria listed in the LRA and how the criteria align with the ACI 349.3R-96 criteria.

Provide the acceptance criteria associated with each of the three categories and indicate how the three categories are comparable to the categories provided in ACI 349.3R-96.

PG&E Response to RAI B2.1.32-1

Diablo Canyon Power Plant (DCPP) Structural Monitoring Program (SMP) implements the requirements of 10 CFR 50.65, with enhancements to satisfy the ten elements specified in NUREG-1801. DCPP's SMP uses "condition classifications," defined in accordance with the guidance provided in NEI 96-03, revision D, to categorize the level of aging affects and utilizes the following acceptance criteria:

> *The responsible engineer shall determine the acceptable/performance criteria for use in the analysis of a structure's condition. In general, the acceptance criteria is based on design bases and licensing bases for the structure, system, or component (e.g., Design Criteria Memoranda, Technical Specifications, Equipment Control Guidelines, FSAR Update, Safety Evaluation Report and its supplements, licensing commitments to the NRC, design drawings, "As Built" drawings, system correspondence files, and industry codes and standards).*

DCPP's current licensing basis does not include a commitment to comply with the requirements of ACI 349.3R-96.

The table below, summarizes DCPP's condition classification categories for concrete structures, with the basis for placing a structure in each category, and compares the categories to the ACI 349.3R-96 evaluation criteria.

Structural Concrete Condition Classifications, Basis for Placing a Structure in Each Category, and Evaluation Criteria		
DCPP "Condition Classification"	DCPP "Basis for Placing in Category"	ACI 349.3-96 "Evaluation Criteria"
Acceptable: Acceptable structures are capable of performing their design functions, including the protection and support of systems or components within the scope of the Maintenance Rule. Acceptable Structures are free of deficiencies or degradations that could lead to possible failure.	A structure is assigned to this category if there are no deficiencies or degradations, or a simple review of the degraded condition indicates that the structure will meet the acceptance criteria.	"Acceptable without further evaluation" (ACI 349.3R, Section 5.1) and/or "Acceptable After Review" (ACI 349.3R, Section 5.2)
Acceptable with Deficiencies: Structures that are acceptable with deficiencies are capable of performing their design functions, including the protection and support of systems or components within the scope of the Maintenance Rule, but are degraded or have deficiencies which could deteriorate to an unacceptable condition, if not analyzed or corrected prior to the next scheduled examination.	A structure is assigned to this category for a range of deficiencies – minor, requiring a review, to major, requiring a detailed evaluation, to assess the current condition of the structure and demonstrate that the structure will meet the acceptance criteria	"Acceptable After Review" (ACI 349.3R, Section 5.2) and/or "Condition Requiring Further Evaluation" (ACI 349.3R, Section 5.3)
Unacceptable: Unacceptable structures are those that are damaged or degraded to an extent that they are not capable of performing their design structural functions, including the protection and support of systems or components within the scope of the Maintenance Rule.	A structure is assigned to this category if a detailed evaluation of the degraded condition indicates that it can no longer meet the acceptance criteria..	"Condition Requiring Further Evaluation" (ACI 349.3R, Section 5.3)

License Renewal Application (LRA) Sections A1.32 and B2.1.32 have been revised to clarify information regarding ACI 349.3R-96. See revised LRA Sections A1.32 and B2.1.32 in Enclosure 2.

RAI B2.1.32-2

The "detection of aging effects" program element of the LRA AMP states that periodic inspections are scheduled such that the accessible areas of both units are inspected over a maximum ten (10) year interval (measured from the date of the baseline or prior routine observation), except water control structures for which all accessible areas of both units are inspected at a frequency of no more than five (5) years.

Industry standards (e.g., ACI 349.3R-96) identified in the GALL Structures Monitoring Program suggest a five-year inspection frequency for structures exposed to natural environment, structures inside primary containment, continuous fluid-exposed structures, and structures retaining fluid or pressure, and a ten-year inspection frequency for below-grade structures and structures in a controlled interior environment.

It is not clear to the staff that all structures, systems, and components (SSCs) at each unit inspected under this AMP are consistent with the industry standards inspection frequency (e.g., as noted in ACI 349.3R-96) or if the SSCs are only inspected at a frequency of ten years.

Please discuss the inspection frequency for each unit and how the frequency meets compliance with industry standards. The staff needs the information to confirm that the inspection frequency criteria in the DCPP Structures Monitoring Program and criteria in ACI 349.3R-96 are aligned.

PG&E Response to RAI B2.1.32-2

The Diablo Canyon Power Plant (DCPP) Structures Monitoring Program (AMP XI.S6) implements the requirements of 10 CFR 50.65. The inspection frequencies, defined in DCPP procedures, are consistent with the guidance of NUMARC 93-01, Revision 2 and Regulatory Guide 1.160, Revision 2. The DCPP procedures consider the guidance of ACI 349.3R-96 and establish frequencies based on the aggressiveness of environmental conditions and physical conditions of the plant structures. These established frequencies provide assurance that any age-related degradation is detected at an early stage of degradation and that appropriate mitigative actions can be implemented.

As described in License Renewal Application (LRA), Section B2.1.31, Structures Monitoring Program, the inspection frequencies for each unit are as follows:

> *Inspections are scheduled such that the accessible areas of both units are inspected over a maximum ten (10) year interval (measured from the date of the baseline inspection or prior routine observation), except water control structures, for which all accessible areas of both units are inspected at a frequency of no more than five years. Inaccessible Area Inspections, for areas that are inaccessible during normal plant operation, will be scheduled for the next*

available time when the area becomes accessible (e.g., outages, curtailments, maintenance activities). In accordance with a plant procedure, the ASW pump bay and traveling screens are currently inspected by divers on a refueling cycle interval.

Also, as part of the plant system health program, the frequency of a periodic inspection may be adjusted, considering data obtained during previous inspections, aggressiveness of the environmental conditions, industry-wide operating experience, industry events, and the physical conditions of the plant structures and structural components.

Note that the inspection frequency for the exterior surface of the containment structures is in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection XWL.

LRA Sections A1.32 and B2.1.32 have been revised to clarify the inspection frequencies. See revised LRA Sections A1.32 and B2.1.32 in Enclosure 2.

RAI B2.1.32-3

In its "operating experience," program element, the applicant noted that pH, sulfates, and chlorides had been monitored monthly at DCPP powerblock locations from August 2008 through July 2009, to obtain data sufficient for making a groundwater aggressiveness determination. The groundwater sample results indicate that the DCPP powerblock groundwater is nonaggressive (i.e., pH > 6.9, chlorides < 215 ppm, and sulfates < 567 ppm). The GALL Report recommends that for plants with non-aggressive groundwater/soil (i.e., pH > 5.5, chlorides < 500 ppm, or sulfates < 1500 ppm) as a minimum they consider: (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations.

1. *Provide locations where groundwater test samples were, or will be taken relative to safety related and important-to-safety embedded concrete walls and foundations, and provide historical results (i.e., pH, chloride content, and sulfate content).*

2. *Due to the high chloride ambient environment, exposure of some plant structures to sea water at DCPP and indications of concrete cracking, spalling, and delaminations, and steel reinforcement corrosion noted in the "operating experience" portion of the AMP for several structures, discuss PG&E's plans, if any, for opportunistic inspections of below-grade structures.*

PG&E Response to RAI B2.1.32-3

1. Groundwater Testing

Diablo Canyon Power Plant (DCPP) has three in-service groundwater test sample wells located inside the protected area in the vicinity of the safety-related power block structures (i.e. containment structures and auxiliary building). The locations of these wells are as follows (see Figure below):

a) Observation Well 01 (OW1) - located inside the well room at the north end of the auxiliary building, with the wellhead at elev. 85 ft. This observation well has two "french-drain" style circuits that collect groundwater and discharge into the well. One french drain encircles Unit 1 Containment foundation at elev. 70 ft and the other is located directly under the Unit 1 Containment reactor cavity at the elev. 51.5 ft.

b) Observation Well 02 (OW2) - located inside the well room at the south end of the auxiliary building, with the wellhead at elev. 85 ft. This observation well has two "french-drain" style circuits that collect groundwater and discharge into the well. One french drain encircles Unit 2 Containment foundation at elev. 70 ft and the other is located directly under the Unit 2 Containment reactor cavity at the elev. 51.5 ft.

c) Dry Well 115 ft (DY1) - located outdoors, in the radiologically controlled area, near the center of the eastern wall of the auxiliary building, with the wellhead at elev. 115 ft. This sample point collects groundwater from both Dry Well 02 and Dry Well 01 (inaccessible due to blockage of the opening by structural elements), which are interconnected by four horizontal "french drains." The purpose of these wells is to relieve any hydrostatic pressure on the outside surface of the eastern wall of the auxiliary building, due to potential ground flow downwards from the slope to the east of the power block structures. Each well includes a gravity drain at elev. 90 ft, which discharges to the transformer yards (elev. 85 ft) at the north and south ends of the auxiliary building.

In support of the license renewal application (LRA), groundwater samples were collected from the three locations, on a monthly basis, from August 2008 through July 2009, and tested for PH, sulfates, and chlorides. This amount of data was considered to be sufficient for making a groundwater aggressiveness determination. A summary of the results of the groundwater sampling is provided in the table below. An evaluation of the data in the table below indicates that the groundwater is non-aggressive in the vicinity of the power block structures.

Figure - DCPP Groundwater Sampling Locations

Table - DCPP Groundwater Sampling Results

Data Collection Date	Collection Location	Chloride (ppm)	Sulfate (ppm)	PH
7/24/2009	DY1 Drywell 115	54.8	45.3	7.5
	OW1 Observation Well 01	188	554	7.4
	OW1 Observation Well 01-R	191	563	
6/30/2009	DY1 Drywell 115	55	46.3	
	OW1 Observation Well 01	194	599	
	DY1 Drywell 115-R	57.1	48.1	
6/18-22/2009	DY1 Drywell 115	55.9	47	
	OW1 Observation Well 01	204	602	7.6
	DY1 Drywell 115-R	57.5	47.8	8.3
5/15/2009	DY1 Drywell 115	57.9	50.7	7.8
	OW1 Observation Well 01	187	565	7.6
	OW1 Observation Well 01-R	191	567	
4/24/2009	DY1 Drywell 115	56	46.9	7.6
	OW1 Observation Well 01	195	525	7.3
	DY1 Drywell 115-R	57	47.7	
3/19/2009	DY1 Drywell 115	47.1	43.5	8.3
	OW1 Observation Well 01	187	456	7.3
2/20/2009	DY1 Drywell 115	55.2	65	8.1
	OW1 Observation Well 01	159	354	7
	DY1 Drywell 115-R	57.1	64.9	
1/23/2009	DY1 Drywell 115	82.3	46.8	7.2
	OW1 Observation Well 01	200	561	6.9
	OW1 Observation Well 01-R	199	561	
12/20/2008	DY1 Drywell 115	91	51.7	7.4
	OW1 Observation Well 01	215	542	6.8
	DY1 Drywell 115-R	97.8	56.9	
12/3/2008	WW2 Water Well 02	87.7	148	
11/21/2008	OW1 Observation Well 01_R	194	564	
	OW1 Observation Well 01	195	567	7.5
	DY1 Drywell 115	131	92.3	8.2
	OW2 Observation Well 02	134	76.9	
10/24/2008	DY1 Drywell 115	139	92.2	7.8
	OW1 Observation Well 01	194	535	7.7
	DY1 Drywell 115-R	133	91.3	
9/19/2008	OW1 Observation Well 01_R	204	550	
	OW1 Observation Well 01	194	528	7.3
	DY1 Drywell 115	132	98.9	
8/22/2008	DY1 Drywell 115	125	96.1	
	OW1 Observation Well 01	195	517	
	DY1 Drywell 115-R	117	89.2	

2. Opportunistic Inspections of Below-Grade Structures

The DCPP work control procedure requires that engineering perform condition assessments of any metallic commodities exposed in excavations, but does not include a similar requirement for reinforced concrete. The DCPP work control procedure will be revised to include evaluation of reinforced concrete exposed during excavations. See revised LRA Table A4-1 in Enclosure 2.

However, an opportunistic inspection of the east wall of the intake structure was performed in 1997, during excavations associated with the installation of auxiliary saltwater piping. The results of this inspection found no evidence of concrete degradation or indication of reinforcing steel corrosion.

RAI B2.1.32-4

NRC IN 2004-05, "Spent Fuel Pool Leakage to Onsite Groundwater," dated March 3, 2004, has identified the potential for leakage of borated water from the spent fuel pool of pressurized water reactors.

A review of operating experience during the AMP audit did not indicate that leakage of the spent fuel pool has occurred at DCPP. During discussions with DCPP personnel it was indicated that Unit 2 has had a persistent minor leak for many years. It is unclear to the staff if leakage of the borated water has resulted in degradation of either the concrete or embedded steel reinforcement that is inaccessible for inspection.

1. *Provide information with regard to how long the leak has been occurring and the size of the leak (volume). Indicate if any chemical analysis has been performed on the leakage for Units 1 & 2 and provide the results of the analysis.*

2. *What was the root cause of the leakage? Include information on the path of the leakage and structures that could be potentially affected by the presence of the borated water.*

3. *Discuss any plans for remedial actions or repairs to address the leakage. In the absence of a commitment to fix the leakage prior to the period of extended operation, explain how the SMP, or other plant-specific program, will address the leakage to ensure that aging effects, especially in inaccessible areas, will be effectively managed during the period of extended operation.*

4. *Provide background information and data, including industry reports cited in the operating experience records, to demonstrate that concrete and embedded steel reinforcement potentially exposed to the borated water have not been degraded. If experimental results will be used as part of the assessment, provide evidence that the test program is representative of the materials and conditions that exist at DCPP Unit 2 spent fuel pool.*

PG&E Response to RAI B2.1.32-4

The Diablo Canyon Power Plant (DCPP) spent fuel pool (SFP) liner leakage monitoring began in 1988. Leak chases are located behind all liner joints (seam welds) for capturing water that potentially leaks through the liner and/or liner seams or plug welds. Any leakage through the liner is collected in the leak chases and is routed via gravity to a leakage monitoring station which has six collection points with isolation valves. The liner leak chases for both units are sampled and evaluated on a weekly basis.

1. Unit 1 SFP Leakage and Investigation History

Presently there are no liner leaks detected on Unit 1. All leak chases have no leakage, except some occasional minor leakage primarily during refueling outages. No liner repairs have been performed on Unit 1 for the life of the plant.

Unit 2 SFP Leakage and Investigation History

The Unit 2 SFP has had persistent minor leakage that varies from 50 to 975 ml per week, with a typical range of 300 to 500 ml per week. There does appear to be very slight increases of leak rate during outages.

Samples are analyzed for the parameters listed below based on frequency and volume of leakage. With small quantities of leakage, there is not always sufficient volume to perform all of the analyses. When this occurs, samples are analyzed based on the following priority from the available quantity: First tritium, then gamma isotopic, then pH, then iron and boron.

a. Tritium: The SFP contains tritium at about 2.5×10^{-1} uCi/ml very consistently. The tritium concentrations in the SFP leakage detection run about 2 to 10 times lower.

b. Gamma Isotopic: The CS-137 concentrations range between 1.01×10^{-6} to 1.95×10^{-5} uCi/ml.

c. pH: There is a consistent trend in the pH samples such that they are basic rather than acidic. The SFP water is normally acidic, about 4.7, due to boric acid. The leakage detection samples are consistently basic with a pH greater than 7. This is most likely due to the water being in contact with the concrete liner.

d. Iron: The results for the iron sampling have consistently been less than 20 parts per billion. These are very small concentrations. This shows that corrosion of the rebars is not occurring. If the rebars inside the concrete were corroding, the iron levels would be much higher.

e. Boron: The SFP boron concentration in both units is normally about 2450 ppm. At this concentration, boron is likely to precipitate with the concrete, if there is only a small volume of water. Beginning in April of 2010, the SFP leakage samples were analyzed by the inductive coupled plasma (ICP) method. This method has given boron results of 900 to 1200 ppm boron. These values further support that boron is precipitating out in the concrete channels of the leakage detection sample pathway.

The percent change in tritium concentration from the SFP to the leak chase is less than the percent change in the boron concentration from the SFP to the leak chase. This reduced boron concentration as compared to tritium is attributed to two factors: 1) boron precipitating in the concrete leak detection channels; and 2) potential dilution by ground water.

The sample from valve SFS-2-56 has been analyzed for iron for the past 10 years. The value is normally 10 ppb iron or less. This would suggest there is very little corrosion of the iron rebar. However, it is possible the iron from corrosion is precipitating in the concrete channel.

2. The evaluations to date have not been able to conclusively identify the root cause of the leakage. During the SFP re-rack in 1988, the liner floor was damaged in a small area and repaired. Records indicate the liner damage was not through-wall.

The path of the leakage is through the liner to the SFP leak chase monitoring location. Structures that could be potentially affected by the presence of the borated water are the SFP concrete and structural steel.

Prior investigations performed by engineering have concluded that long-term leakage is acceptable and will have negligible adverse effect on the concrete and reinforcing steel. Deterioration of concrete from boric acid would result in very slight surface scaling and no cracking of the concrete would occur. Boric acid can cause exposed reinforcing steel to corrode. However, there are no adverse conditions to cause the concrete to crack behind the SFP liner. The type of concrete used for pool structure and the arrangement of rebar would minimize any cracks. Therefore, the uncracked concrete will protect the reinforcing steel from coming into contact with boric acid. Additionally, the weekly release of collected leakage through the leak chase system will prevent any buildup of hydrostatic pressure behind the liner, which could potentially force the water through construction joints. It was also concluded that no SFP leakage reaches the ground water based on DCPP's groundwater sampling program.

3. The amount of leakage being experienced has been evaluated as being acceptable since there is a negligible adverse effect on the concrete and reinforcing steel. Deterioration of concrete from boric acid would result in very slight surface scaling and no cracking of the concrete will occur. Boric acid can cause exposed reinforcing steel to corrode. However, there are no adverse conditions to cause the concrete to crack behind the SFP liner. The type of concrete used for pool structure and the arrangement of rebar would minimize any cracks. Therefore, the uncracked concrete will protect the reinforcing steel from coming into contact with boric acid. Additionally, the weekly release of collected leakage through the leak chase system will prevent any buildup of

hydrostatic pressure behind the liner (which could potentially force the water through construction joints).

PG&E will continue to monitor the Unit 2 SFP leakage and will evaluate new technologies available to detect small SFP leaks.

4. Following operating experience at another nuclear power plant that had identified leak chases being blocked by boric acid deposits, PG&E conducted video inspections of the Units 1 and 2 leak chases in January 2008. The inspection determined that the leak chases were not blocked. A follow up internal video inspection was performed in March 2010 of 2 of the Unit 2 leak chases that were experiencing chronic minor leakage. No changes from the previous video inspections were observed.

Evaluation of SFP Leakage Industry Experience

Main reference document is the following EPRI report:

Repair and Replacement Applications Center: Boric Acid Attack of Concrete and Reinforcing Steel in PWR Fuel Handling Buildings. EPRI, Palo Alto, CA: 2009. 1019168, Final Report, June 2009.

In response to SFP leakage at another nuclear power plant, efforts were made including research of published studies and laboratory testing to determine the effects on reinforced concrete and reinforcing steel after prolonged exposure to boric acid leakage from SFP. The results of these efforts, corroborated by the results from the examination of concrete samples from the fuel handling building of the decommissioned Connecticut Yankee Haddem Neck Nuclear Power Plant, have been documented in the above referenced EPRI report.

Key insights from the report:

1. The reaction between hardened cement paste and an acid solution is controlled by diffusion of the acid into the concrete. The concrete degradation rate decreases over time because the distance through which acid must diffuse to reach unaffected concrete increases. Degradation of concrete by acids follows Fick's Law of Diffusion in which the depth of degradation varies with the square root of time. Therefore, the rate of degradation decreases monotonically. During the laboratory tests, the concrete degradation rate decreased substantially during the 39-month test series. The projected depth of affected cement paste after 70 years of exposure is 1.3 inches (33.02 mm), including adjustment for uncertainty.

2. Reinforcing steel (embedded in concrete) will have negligible corrosion when exposed to SFP leakage over long periods of time (~ 0.004 mm/yr).

3. The wicking effect at the reinforcing steel/concrete interface is minor. Degradation of reinforcing steel at construction joints or cracks with boric acid migration will not spread rapidly and the rebar degradation will remain localized to the vicinity of the construction joint or crack.

Applicability of EPRI report to DCPP

DCPP has similar boric acid concentrations of equal to or greater than 2000 ppm in the SFP. It also has similar concrete mix with type II cement and non-reactive aggregate (granite), which produces a 5000-psi concrete. Therefore, it is reasonable to conclude that (numerical) test results (documented in EPRI report) will also be applicable to DCPP.

RAI B2.1.32-5

The "operating experience" section of LRA Section B2.1.32, "Structures Monitoring Program," states that baseline inspections completed in accordance with 10 CFR 50.65(a), Maintenance Rule, in 1997-2003, and the first periodic follow-up inspection completed in early 2009 concluded that the plant's structures were in good condition and performing well.

During a walkdown of the Unit 1 auxiliary building, the staff noted that there was a crack in the reinforced concrete ceiling adjacent to the spent fuel pool that exhibited evidence of prior leakage in the form of white deposits, potentially indicating either leaching of calcium hydroxide from the concrete or boric acid deposits. The staff is uncertain of the source of the leakage or if this has been documented and will be addressed.

1. *Provide information regarding the occurrence of the crack and the source of the apparent leakage.*

2. *Provide information on any chemical analysis performed on the deposits and analyses conducted to identify the leakage source and path of the leakage.*

3. *If the source of the leakage is the spent fuel pool, identify structures that potentially could be affected by the presence of borated water.*

4. *Discuss any plans for remedial actions or repairs. In the absence of a commitment to repair the crack prior to the period of extended operation, provide information/documentation to demonstrate that the concrete and embedded steel reinforcement have not degraded.*

PG&E Response to RAI B2.1.32-5

1. A visual inspection of the crack using a 10-ft ladder (on elevation 115 ft) determined the crack to vary in width from 0.025 to 0.040 inches. The crack is currently dry and there was no evidence of active leakage. It appears that the white deposits on either side of the crack were previously cleaned several times. One cleaning was conducted prior to the original clear coating (20-plus years ago). There was evidence of the original clear coating covering most of the white deposits. White deposits that were on top of the clear coating have also been cleaned. The inspection identified no corrosion or spalling of the concrete indicating no degradation to the embedded reinforcing bars. The crack down the adjoining wall slab goes down the wall 3 ft and is 0.005 to 0.015 inches wide. This crack was caused by drying shrinkage and is now acting as a control joint. This crack has no characteristics that indicate an overloaded structural flexure, or shear induced crack. The crack down the adjoining wall slab is also a shrinkage crack. Given that this crack is acting as a control joint, no repair of this crack is

required. This crack is most likely active due to temperature variation. The crack does not adversely impact the concrete element.

Upon further investigation of the crack from inside the concrete air duct, it was confirmed that the crack is indeed a shrinkage crack, as there was no evidence of leakage or moisture from the adjoining west wall (spent fuel pool wall). There were signs of previous moisture in the concrete air duct, as seen from evidence of white deposits around the expansion joints adjacent to the east wall (exterior wall) of the auxiliary building. The source of the moisture is most likely rain water entering from the exterior wall and traveling along the expansion joints, and on to the floor of the concrete duct. There were areas of the concrete floor slab inside the air duct, adjacent to the exterior (east) wall, where the paint was peeling. This occurred in the vicinity of the crack in question, making it a high likelihood that this is the moisture that is causing the shrinkage crack and white deposits.

This crack has no characteristics that indicate an overloaded structural flexure or shear induced crack. Dry shrinkage is caused by the loss of moisture from the cement paste during the curing process, which causes the concrete element to shrink. Most shrinkage occurs during the initial curing of the concrete (first 28 days). Dry shrinkage cracking is common in large reinforced concrete structures with limited control joint as found within the auxiliary building. These shrinkage cracks do not affect the structural capacity of the concrete element and are within the range of common shrinkage cracks for this type of structure.

2. Historically, there has been no chemical analysis performed on the deposits. The chemistry engineers performed a scaring of the deposit and collected about 0.1 grams. The sample was characterized with a combination of x-ray diffraction and scanning electron microscopy with associated energy dispersive x-ray spectroscopy. The analysis showed a trace amount of only Co-58. The chemical analysis determined the deposits were mainly calcium carbonate. No Boron was detected from the sample, concluding that the source of the moisture was not from the spent fuel pool.

3. As previously mentioned, it was determined from the inspections that the source of the water is not from the spent fuel pool. Therefore, the exposure of boron to other plant structures is not an issue.

4. The inspection identified no corrosion or delamination / spalling of the concrete indicating no degradation to the embedded reinforcing bars. It has been determined that the crack was caused by drying shrinkage and is now acting as a control joint. The crack characteristics indicated that it was neither an overloaded structural flexure not a shear-induced crack. The subject crack does not adversely impact the performance characteristic of the structure and potential for prorogation is not evident. Given that this crack is acting as a control joint, repair of this crack is not required.

RAI B2.1.33-1

Element 10 of the GALL Report AMP XI.S7, "RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants," states that inspections implemented in accordance with the guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

The "operating experience" program element of LRA Section B2.1.33 states that since 1996, the Intake Structure has been place in Maintenance Rule (MR), Goal Setting (a)(1) status twice. Each occurrence indicated further the adverse impacts of harsh saltwater environment on concrete degradation. With the current refurbishment program and procedural controls in place, the Intake Structure is expected to resume monitoring under MR (a)(2) status by 2010. However, it is not clear to the staff how the adverse impacts are quantified. In addition, it is not clear how the current refurbishment program will be able to manage the aging during the period of extended operation.

Provide the following information:

1. *How the adverse impacts, including delaminations in the concrete at the Intake Structure, are quantified.*

2. *A summary of the evaluations and assessments performed to determine the scope of the refurbishment program.*

3. *Details of the current refurbishment program, and how it will help in aging management during the period of extended operation. How does the current refurbishment program differ from the two previous repairs performed since 1996?*

In absence of a formal commitment to refurbish the Intake Structure, explain how the DCPP AMP XI.S7, "RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants," will adequately manage aging during the period of extended operation.

PG&E Response to RAI B2.1.33-1

1. As discussed in LRA Section B2.1.33, the intake structure is currently monitored and inspected in accordance with Diablo Canyon Power Plant (DCPP) procedures on refueling outage intervals. Areas of concrete degradation are inspected and documented by utilizing concrete experts and technicians from PG&E's Applied Technology Services (ATS) department. Degraded conditions, including delaminations, are documented on drawings of the intake structure. These drawings are updated following each refueling outage and used to assess the conditions against design and licensing basis criteria and for trending purposes. The drawings are part of inspection reports prepared by ATS.

2. A comprehensive qualitative assessment of the intake structure was performed, which included all areas that needed repairs. As a result this assessment, a Maintenance Rule Goal Setting Review was performed and placed the intake structure into Maintenance Rule (a)(1) status in early 2006. A refurbishment plan was developed and documented in the corrective action program based on general assessment of the condition of the structure including review of maintenance rule data files and various inspection reports.

3. The current refurbishment plan includes the following:

   a. Concrete repair/install concrete sacrificial cathodic protection within the traveling screen forebays 1- 2, 1-3, 1-5, 1-6, 2-1, 2-2, 2-3, 2-4, 2-5 and 2-6.
   b. Concrete repair to the seawall refuse sump overflow opening
   c. Concrete repair within circulating water conduit 1-1 and 1-2.
   d. Concrete repair and installation of cathodic protection on seawall
   e. Concrete repairs to the floors and walls of auxiliary saltwater pump vaults 1-1/1-2.
   f. Concrete repair and cathodic protection anode installation top deck
   g. Concrete repair of the intake structure pump deck.

   The refurbishment plan differs from the previous repairs in some of the methods of repairs. For instance, the prior program utilized zinc strips to provide galvanic protection for the rebar in the repaired areas. These had a tendency to passivate and no longer protect the rebar over time. The current program utilizes encapsulated zinc anodes to provide galvanic protection. Vendor documentation and studies has shown this type of anode provides better protection of the rebar and resist passivation than the zinc strips. In addition, embedded galvanic anodes will be installed to provide protection of the sound concrete in the more inaccessible areas of the structure. This is intended to limit the progression of corrosion of the reinforcing steel in areas of sound concrete. This method has been utilized in traveling screen forebays as they are extremely difficult to access.

   DCPP's system and structures health review process ensures that plant structures, systems, and components are adequately maintained to ensure that they will continue to perform their intended functions. The intake structure is reviewed as part of this process. The implementation of this process will ensure that the aging of the intake structure will be adequately managed during the period of extended operation, and maintained in Maintenance Rule (a)(2) status.

RAI B2.1.33-2

Element 10 of the GALL Report AMP XI.S7, "RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants," states that inspections implemented in accordance with the guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

The "operating experience" section of AMP B2.1.33 states that the discharge structure which is currently being monitored and inspected in accordance with DCPP procedures on refueling cycle intervals has had some minor concrete repairs done to the exterior incline wall in early 2002. In addition, during a walkdown, the staff noted delamination of concrete on the top slab of the Discharge Structure. However, PG&E states in LRA Section B2.1.33 that the Discharge Structure is in acceptable condition.

Provide the following information:

1. *How the concrete inside the Discharge Structure is examined and documented during each refueling cycle. Does this examination include any non-destructive examination?*

2. *History and details of the repairs performed in the Discharge Structure, and how these repairs will prevent further degradation during the period of extended operation.*

3. *What plans does PG&E have, if any, to repair or remove the delaminations in the Discharge Structure?*

PG&E Response to RAI B2.1.33-2

1. The inspection is performed and documented per Diablo Canyon Power Plant (DCPP) Procedure TS1.ID4, "Saltwater Systems Aging Management Program" of all the accessible areas of the discharge structure. NDE examinations including hammer sound testing and impact echo testing have been previously performed and will continue to be performed as required in the future.

2. The accessible interior of the discharge structure was partially inspected in 1991 and 1999. These inspections included visual inspection of the concrete surface above the waterline from the vantage point of the weir at STA 3+51. No repairs were determined necessary as a result of these inspections. Based on these inspections, the discharge structure is characterized as structurally sound, with only minor defects (i.e., localized areas of spalling and delaminated concrete) found during these inspections.

   The exterior surface of the discharge structure incline wall was repaired in 2004. This portion of the structure is near the lower exterior and is subjected to harsh wetting and drying, which greatly accelerates corrosion. This repair utilized embedded anodes to protect the reinforcing steel in the sound/unsound portions of the concrete for continued concrete life.

After the repair of the exterior surface of the incline wall, impact echo testing was performed to determine if any delaminations existed on the inside surface of the incline wall. No repairs were determined necessary based on these inspections.

3. Currently, there are no plans to install a cathodic protection system for the discharge structure. The typical repair method has been to remove the delaminated sections of concrete, clean and add/splice rebar (if deemed necessary), and place new sound concrete. To help protect the repairs, Galvashield ® XP+ anodes are used to limit further corrosion in the reinforcing steel.

<u>*RAI B2.1.33-3*</u>

Element 10 of the GALL Report AMP XI.S7, "RG 1.127, Inspection of Water Control Structures Associated with Nuclear Power Plants," states that inspections implemented in accordance with the guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

The "operating experience" section of AMP B2.1.33 states that the discharge circulating water conduits (DCWC) concrete is not visible for detailed inspections due to marine growth found on the interior wall surface. The applicant is developing a schedule to remove marine growth in order to further enhance the monitoring process.

Provide the following information:

1. *When was the DCWC interior concrete surface last inspected in accordance with the requirements ACI 349.3R?*

2. *What is the current frequency for inspection of the DCWC interior concrete surface?*

3. *If marine growth is not removed, explain how the program will effectively manage aging of the DCWC interior concrete during the period of extended operation.*

4. *How is the DCWC interior concrete surface inspected since it is covered with marine growth?*

<u>PG&E Response to RAI B2.1.33-3</u>

1. The Units 1 and 2 circulating water (CW) discharge conduits were last inspected during the May 2002 Unit 1 refueling outage (1R11), and during the May 2001 Unit 2 refueling outage (2R10), respectively. This was an inspection of accessible sections of the CW discharge conduits in accordance with PG&E administrative procedure TS1.ID4 "Saltwater Systems Aging Management Program" and PG&E Plant Engineering Procedure PEP C-17.14, "Concrete Surveillance Program for the Saltwater Systems."

   As discussed in the response to RAI B2.1.32-2, the Structures Monitoring Program implements the requirements of 10 CFR 50.65, and the inspection frequency is consistent with the guidance of NUMARC 93-01, Revision 2 and Regulatory Guide 1.160, Revision 2 in accordance with plant procedures. These plant procedures consider the guidance of ACI 349.3R-96 and establish frequencies based on the aggressiveness of environmental conditions and physical conditions of the plant structures. These established frequencies provide assurance that any age-related degradation is detected at an early stage of degradation and that appropriate mitigative actions can be implemented.

Procedure PEP C-17.14 uses ACI 349.3R as guidance and the inspection criteria is consistent with its requirements.

2. Per TS1.ID4, Section 4.2.2, submerged structures, systems, and components (SSCs) that are not continuously under water should be inspected once every planned refueling outage for that unit in an outage; however discretion on inspection frequency is left up to the system engineer, based on prior material conditions and the significance of the structure so long as it does not exceed the inspection intervals as set forth in the Civil Maintenance Rule Program (currently set at 10 years). License Renewal Application (LRA), Section B2.1.33 notes that in accordance with Regulatory Guide 1.127, an inspection frequency of 5 years is required.

3. Inspections to evaluate the engineering concrete properties of the discharge conduits at DCPP were performed during 2R10 and 1R11. The inspection was conducted in accordance with the inspection program established for submerged and non-submerged areas, TS1.ID4, "Saltwater Systems Aging Management Program;" TS1.NE2, "Structural Monitoring Program;" MA1.NE1, "Maintenance Rule Monitoring Program – Civil Implementation;" and PEP C-17.14, "Concrete Surveillance Program for the Intake Structure." The inspection of the discharge conduits involved a visual inspection of the concrete and sounding for delaminations of sections that were scraped of marine life.

Unit 1 Inspections

The inspection of CW Discharge Conduits 1-1 and 1-2 involved a visual inspection of the concrete and sounding for delaminations of sections that were scraped of marine life. The general condition of the concrete in CW Discharge Conduits 1-1 and 1-2 is structurally sound. Of the total 33,900 square feet of surface area in the discharge conduits, 29,500 square feet of concrete surface was visually inspected. In the area between the gate guides and Sta. -0+40, no spalls or other defects were found during the visual inspection. Two small delaminations were found in CW Discharge Conduit 1-1, with a total area of approximately 12 square feet in size. These two delaminations were located during the 10th refueling outage for Unit 1 and had a corresponding total area of 7 square feet. Two small spalls were located in the area between the end of the common wall and the weir. Their combined area is just over 1 square foot, each spall had exposed reinforcing steel. No repairs are required in either discharge conduit during the 11th refueling outage for Unit 1.

Portions of CW Discharge Conduits 1-1 and 1-2 were visually inspected. CW Discharge Conduit 1-1 was visually inspected from (Sta. -0+40) to the weir (Sta. 3+53). CW Discharge Conduit 1-2 was visually inspected from (Sta. -0+40 to the weir (Sta. 1+89). Approximately 255 square feet of concrete was scraped of marine life and sounded for delaminations. Three areas were scraped of marine life and sounded for delaminations within CW Discharge Conduit 1-1. Area #1 was at (Sta. 2+25) on the common wall, with an area 5.5-ft high by 10-ft wide. Area #2 was at (Sta. 2+40) on the outside wall of CW Discharge Conduit 1-1, with an area 6-ft high by 10-ft wide. Area #3 was at (Sta. 2+85) on the common wall with an area of 5-ft high by 10-ft wide. In CW Discharge Conduit 1-2, two areas were scraped of marine life and sounded for delaminations. Area #1 was on the outside wall at (Sta. 1+35), with an area 5-ft high by 10-ft wide. Area #2 was on the outside wall at (Sta. 1+65), with an area 4-ft high by 10-ft wide. In the area between the end of the common wall and the weir there are 24-ft high walls with the marine growth extending up to a height of 12 ft from the floor. The remaining 12 ft of wall located above the marine growth was sounded for delaminations on the north, south and east walls.

Unit 2 Inspections

The inspection of CW Discharge Conduits 2-1 and 2-2 involved a visual inspection of the concrete and sounding for delaminations of sections that were scraped of marine life. The general condition of the concrete in CW Discharge Conduits 2-1 and 2-2 is structurally sound. Of the total 33,900 square feet surface area in the discharge conduits, 17,400 square feet of concrete surface was visually inspected. One small delamination was found in CW Discharge Conduit 2-1, approximately 1 square foot in size. No repairs are required and no repairs were performed in either discharge conduit during the 2R10 outage.

Portions of CW Discharge Conduits 2-1 and 2-2 were visually inspected. CW Discharge Conduit 2-1 was visually inspected from (Sta. -0+50) to the weir (Sta. 1+89). CW Discharge Conduit 2-2 was visually inspected from (Sta. 2+50 to the weir (Sta. 3+58). Approximately 380 square feet of concrete was scraped of marine life and sounded for delaminations. Two areas were scraped of marine life and sounded for delaminations within CW Discharge Conduit 2-1. Area #1 was at the expansion joint near the "A" line wall (Sta. 0+03) on the common wall with an area 4-ft high by 6-ft wide. The area straddled the expansion joint and the bottom of the area was approximately 3 ft off the ground. Area #2 was at (Sta. 1+35) on the outside wall of CW Discharge Conduit 2-1. An area approximately 6-ft high by 8-ft wide was scraped of marine life and sounded for delaminations. In CW Discharge Conduit 2-2, one area was scraped of marine life and sounded for delaminations. The location was on the common wall at (Sta. 2+50), with an area of approximately 6-ft high by 10-ft wide.

The area between the end of the common wall and the weir consists of an enlarged section where the walls are 24-ft high (conduits are 12-ft high). The marine growth in this zone extends up to approximately 12 ft off the floor. In this area, five locations were scraped of marine life and sounded for delaminations. Three of these areas were on the north wall. Two areas on the south all were inspected. An additional 900 square feet was sounded for delaminations above the bio-foul line on the north, east, and south walls between the end of the common wall the weir.

PG&E is planning on performing the next set of inspections on the discharge conduits during the May 2011 Unit 2 refueling outage (2R16) and during the May 2012 Unit 1 refueling outage (1R17). These inspections will require removal of the marine growth. Subsequent inspections will be performed in accordance with TS1.ID4 requirements. To ensure aging of the discharge conduits will be adequately managed for the period of extended operation, the inspection of the discharge conduits will be performed prior to the period of extended operation in accordance with applicable requirements.

4. Please see response (3). LRA Section 2.4.12 indicates that the discharge structure provides structural support, shelter and protection for the ASW return lines. Sample sections of accessible portions of the discharge conduits were scraped of marine growth prior to inspection. Future inspections will require the removal of marine growth prior to inspection. These sample inspections are considered adequate to demonstrate that the discharge structure is capable of performing its intended license renewal function.

LRA Amendment 3

LRA Section	RAI
Section A1.32	B2.1.32-1, B2.1.32-2
Section B2.1.32	B2.1.32-1, B2.1.32-2
Table A4-1	B2.1.27-2, B2.1.28-3, B2.1.30-1; B2.1.32-3

A1.32 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program manages cracking, loss of material, and change in material properties by monitoring the condition of structures and structural supports that are within the scope of license renewal. The Structures Monitoring Program implements the requirements of 10 CFR 50.65, *the Maintenance Rule*, and is consistent with the guidance of NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 2 and Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Revision 2. Inspection methods~~, inspection frequency~~ and inspector qualifications are in accordance with ~~DCPP procedures that reference~~ ACI 349.3R-96 and ASCE 11-90. *The plant procedures consider the guidance of the American Concrete Institute Committee Report No. ACI 349.3R-96 and establish frequencies based on the aggressiveness of environmental conditions and physical conditions of the plant structures. These established frequencies provide assurance that any age-related degradation is detected at an early stage of degradation and that appropriate mitigative actions can be implemented.*

The Structures Monitoring Program provides inspection guidelines for concrete elements, structural steel, structural features (e.g. caulking, sealants, roofs, etc.), and miscellaneous components such as doors. The Structures Monitoring Program includes all masonry walls and water-control structures within the scope of license renewal and inspects supports for equipment, piping, conduit, cable tray, HVAC, and instrument components.

B2.1.32 Structures Monitoring Program

Program Description

The Structures Monitoring Program (SMP) manages cracking, loss of material, and change in material properties by monitoring the condition of structures and structural supports that are in the scope of license renewal. The SMP implements the requirements of 10 CFR 50.65, the Maintenance Rule, which is consistent with the guidance of NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Revision 2 and Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Revision 2. ~~Inspection methods, inspection frequency and inspector qualifications are in accordance with DCPP procedures which reference ACI 349.3R-96 and ASCE 11-90.~~ The SMP provides inspection guidelines and walkdown checklists for concrete elements, structural steel, masonry walls, structural features (e.g. caulking, sealants, roofs, etc.), structural supports, and miscellaneous components such as doors. The SMP includes all masonry walls and water-control structures within the scope of license renewal. The SMP also inspects supports for equipment, piping, conduit, cable tray, HVAC, and instrument components. The scope of the SMP does not include the inspection of the supports specifically inspected per the requirements of the ASME Section XI ISI Program. Though coatings may have been applied to the external surfaces of structural members, no credit was taken for these coatings in the determination of aging effects for the underlying materials. The SMP evaluates the condition of the coatings as an indication of the condition of the underlying materials.

The following structures are within the scope of License Renewal and are in the scope of the SMP inspections:

Auxiliary Building (includes the control room)
Containment Structure
Turbine Building
Radwaste Storage Facilities
Pipeway Structure
Fuel Handling Building Steel Superstructure
Commodity Supports and Anchorages
Outdoor Tanks and Foundations
Buried Structural Commodities
Electrical Structures and Foundations

The following water control structures are also within scope of license renewal and in scope for SMP:

Intake Structure
Discharge Structure
Circulating Water Conduits

Earth Slopes over the ASW pipes
East and West Breakwaters
Raw Water Reservoirs

The DCPP SMP manages aging by providing measures for monitoring that detect the effects of aging prior to loss of intended function.

~~The aging effects monitored by the DCPP SMP, are consistent with ACI 349.3R-96 and ASCE 11-90.~~

The plant procedures consider the guidance of the American Concrete Institute Committee Report No. ACI 349.3R-96 and establish frequencies based on the aggressiveness of environmental conditions and physical conditions of the plant structures. These established frequencies provide assurance that any age-related degradation is detected at an early stage of degradation and that appropriate mitigative actions can be implemented. The aging effects monitored by the DCPP SMP, are consistent with ACI 349.3R-96 and ASCE 11-90.~~The inspection methods, inspection schedule, and inspector qualifications are specified in the DCPP SMP, which is consistent with ACI 349.3R-96 and ASCE 11-90. Visual inspections are used to determine the condition of SSCs within the scope of the SMP, unless more rigorous inspections are deemed necessary by the design system engineer or civil coordinator.~~

The inspection methods and inspector qualifications are specified in the DCPP SMP, are consistent with ACI 349.3R-96 and ASCE 11-90. Visual inspections are used to determine the condition of SSCs within the scope of the SMP, unless more rigorous inspections are deemed necessary by the design system engineer or civil coordinator.

Inspections are scheduled such that the accessible areas of both units are inspected over a maximum 10-year interval (measured from the date of the baseline or prior routine observation), except water control structures, for which all accessible areas of both units are inspected at a frequency of no more than five years. Inaccessible Area Inspections, for areas that are inaccessible during normal plant operation, will be scheduled for the next available time when the area becomes accessible (e.g., outages, curtailments, maintenance activities). In accordance with a plant procedure, the ASW pump bay and traveling screens are currently inspected by divers on a refueling cycle interval. This procedure will be enhanced to also specifically include inspection of the bar racks, and associated structural components.

The DCPP SMP is consistent with 10 CFR 50.65. Any Civil SSC classified as "acceptable with deficiencies" or "unacceptable" requires consideration for transfer to (a)(1) status. All other civil SSCs are assigned to (a)(2) status. The SMP provides guidance for the determination of performance criteria for SSCs included within the scope of the Maintenance Rule. These guidelines were used to establish the inspection attributes for SSCs monitored by the DCPP SMP. The DCPP SMP uses "Acceptable", "Acceptable with Deficiencies", and "Unacceptable" to classify levels of aging effect for each inspection attribute. The classifications and acceptance criteria are based on

DCPP design bases documents, current licensing bases, and industry standards, such as ACI 349.3R-96 and ASCE 11-90.

NUREG-1801 Consistency

The Structures Monitoring Program is an existing program that, following enhancement, will be consistent with NUREG-1801, Section XI.S6, Structures Monitoring Program.

Exceptions to NUREG-1801

None

Enhancements

Prior to the period of extended operation, the following enhancements will be implemented in the following program elements:

Parameters Monitored or Inspected – Element 3

Plant procedures will be enhanced to monitor groundwater samples every five years for pH, sulfates and chloride concentrations, including consideration for potential seasonal variations.

Plant procedures will be enhanced to specify inspections of bar racks and associated structural components in the intake structure.

Operating Experience

DCPP's SMP is performed in accordance with 10 CFR 50.65(a), Maintenance Rule (10-year intervals). The inspections assess the overall condition of DCPP structures, passive components and Civil Engineering features. The inspection results are used to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Baseline inspections of structures in scope of Maintenance Rule were completed between 1997 and 2003. The first periodic follow up inspection was completed in 2009.

Overall, the baseline inspection report concluded that the plant's structures were in good condition and performing well. Conditions that were noted as having deficiencies were documented and addressed under the Corrective Action Program. Many of the observed conditions were noted for further review during the follow-up periodic inspections. Though the concrete Intake Structure refurbishment program was in progress to repair self-identified structural degradation prior to the Intake being scoped into the Maintenance Rule Program (October 1996), the Intake Structure was conservatively placed into Maintenance Rule (10 CFR 50.65) goal setting status (a)(1). This action was due to the chloride environment that the intake was exposed to and the extent of repairs being required to restore the structure. As a result of an aggressive refurbishment program, the necessary repairs and remediation were performed and the Intake Structure was removed from (a)(1) status in October 1998.

The first periodic follow-up inspection and report was completed in early 2009. The overall condition found the plant structures in good condition. The inspection found no conditions requiring immediate maintenance or repairs. Conditions noted were minor in nature and did not affect the structural integrity of any of the structures inspected. In some cases, corroded steel that was painted as a result of the baseline inspections had corrosion reappear. In such cases, the subject steel was located in damp or wet environments, primarily due to its exposure to the harsh coastal environment. These areas were re-identified in the Corrective Action Program to perform recoating. Some minor concrete cracking and spalling was also identified in the Turbine Building at areas near ventilation louvers. Rainwater leaking through exterior wall louvers has caused embedded reinforcing steel to corrosion and subsequently concrete cracking and spalling. The areas identified are relatively small and do not currently adversely impact the structural integrity of the structural element. However, concrete repairs and/or further examinations will be performed to prevent further degradation to the concrete elements.

The Intake Structure continues to require attention and remediation due to its location in a harsh coastal environment. As a result of a negative trend in concrete degradation, the Intake was placed back into Maintenance Rule goal setting (a)(1) status in December 2005. A repair plan is in place in order to return the Intake Structure to (a)(2) status by 2010.

The ASW pump bay, traveling screens and bar racks are currently inspected by divers on a refueling cycle interval. Any degradation noted during these inspections are entered into the corrective action program, evaluated for impact on the ASW system operability and identified for long term corrective actions as required. Inspections performed to date have not identified any degradation that would impact the ability of the ASW system to perform its intended function.

PH, sulfates, and chlorides were monitored monthly at DCPP powerblock locations from August 2008 through July 2009 to obtain data sufficient for making a groundwater aggressiveness determination. The groundwater sample results show that DCPP powerblock groundwater is non-aggressive (pH>6.9, chlorides<215ppm, and sulfates<567ppm).

The SMP has identified and corrected age-related issues for in-scope structures and structural components. On-going identification of degradation and corrective action prior to loss of intended function provides reasonable assurance that the program is effective for managing the aging effects of structural components.

The DCPP operating experience findings for this program identified no unique plant specific operating experience; therefore DCPP operating experience is consistent with NUREG-1801.

Conclusion

The continued implementation of the Structures Monitoring Program will provide reasonable assurance that aging effects will be managed such that the structures within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Table A4-1 License Renewal Commitments

Item #	Commitment	LRA Section	Implementation Schedule
31	*The Unit 2 gap repair work will be completed prior to the period of extended operation.*	B2.1.27	*Prior to the period of extended operation*
32	*DCPP plant procedures will be revised to perform concrete inspections per ASME Section XI Subsection IWL within a 5-year interval.*	B2.1.28	*Prior to the period of extended operation*
34	*The DCPP work control procedure will be revised to include evaluation of reinforced concrete exposed during excavations.*	B2.1.32	*Prior to the period of extended operation*