April 1, 2002

PG&E Letter DCL-02-033

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Dear Commissioners and Staff:

Enclosed is the Diablo Canyon Power Plant 15-day response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002. NRC Bulletin 2002-01 requested information related to the integrity of the reactor coolant pressure boundary, including the reactor pressure vessel head, and the extent to which inspections have been performed to satisfy applicable regulatory requirements. It also requested information related to the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and how future inspections will ensure continued compliance with applicable regulatory requirements.

If you have questions regarding this response, please contact Mr. Pat Nugent at (805) 545-4720.

Sincerely,

Lawrence F. Womack

tcg
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

	)	Docket No. 50-275
In the Matter of	)	Facility Operating License
PACIFIC GAS AND ELECTRIC COMPANY	)	No. DPR-80
	)	
Diablo Canyon Power Plant	)	Docket No. 50-323
Units 1 and 2	)	Facility Operating License
	)	No. DPR-82

AFFIDAVIT

Lawrence F. Womack, being of lawful age, first being duly sworn upon oath says that he is Vice President – Nuclear Services of Pacific Gas and Electric Company; that he has executed this response to NRC Bulletin 2002-01 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Lawrence F. Womack
Vice President – Nuclear Services

Subscribed and sworn to before me this 1st day of April, 2002.

Notary Public
County of San Luis Obispo
State of California

Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

NRC Requested Information

1. *Within 15 Days of the date of the bulletin, all PWR licensees are required to provide the following:*

   A. *a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant.*

PG&E Response:

Reactor pressure vessel head inspection program:

PG&E's boric acid leakage inspection program (which includes the reactor pressure vessel head area) is implemented by the following procedures: AD4.ID2, "Plant Leakage Evaluation;" Surveillance Test Procedure (STP) R-8C, "Containment Walkdown for Evidence of Boric Acid Leakage;" STP R-8A, "Reactor Coolant System Leakage Test;" and ISI X-CRDM, "Reactor Vessel CRDM Inspection." Procedures AD4.ID2 and STP R-8C reflect PG&E commitments made in response to NRC Generic Letter 88-05.

AD4.ID2 discusses PG&E's commitments with respect to Generic Letter 88-05, and provides a standardized method for reporting and tracking leakage from systems at Diablo Canyon Power Plant (DCPP). PG&E's policy is to minimize boric acid induced corrosion by applying an administrative program that provides for: (1) early detection of boric acid leaks; (2) thorough inspection of the areas surrounding identified boric acid leakage; (3) proper evaluation of areas where leakage has occurred; and (4) prompt action to mitigate the leak, perform repairs, and avoid future damage.

STP R-8C is used to identify boric acid leakage from any source inside containment to prevent boric acid corrosion of Class 1 low alloy/carbon steel reactor coolant pressure boundary (RCPB) components. It is also used to perform examinations of the control rod drive mechanism (CRDM) area above the reactor pressure vessel head insulation if the procedure is being used as a result of an entry into a forced outage and an examination of the CRDM area is warranted (i.e., if leakage is suspected from reactor pressure vessel head components).

ISI X-CRDM provides guidelines for inspection of head penetration canopy seal welds and head penetration tubes for evidence of through wall leakage during refueling outages. ISI X-CRDM has been revised to include direction for performance of reactor pressure vessel head penetration visual inspections required to address NRC Bulletin 2001-01. This procedure has also been revised to inspect the reactor pressure vessel head for boric acid deposits and degradation in response to NRC Bulletin 2002-01.

STP R-8A is the system leakage test required by ASME Section XI for the Class 1 pressure boundary. As a minimum it includes all joints that have been opened and closed since the last performance of the test. This inspection includes the reactor pressure vessel head with the insulation installed.
STP R-8A is performed during the normal heat up and pressurization of the primary system in accordance with Operating Procedure L-1, "Plant Heat up From Cold Shutdown to Hot Standby." A walk down at full operating pressure is conducted by certified examiner(s), and is witnessed or verified by the Authorized Nuclear Inservice Inspector (ANII).

PG&E's inspection program does not currently require a routine 100 percent bare metal reactor pressure vessel head inspection, and PG&E has not performed this inspection in the past.

Maintenance program:

During a normal refueling outage the mirror insulation that covers the portion of the reactor pressure vessel head below the CRDM cooling shroud is removed, and the CRDM cooling ducting is removed from the reactor pressure vessel head. This provides direct visual access to the lower area of the reactor pressure vessel head at the closure flange and to the reactor vessel studs, nuts and washers. It also provides visual access to the portion of the head penetration tubes that are inside the cooling shroud and above the reflective insulation.

<u>NRC Requested Information</u>

1. *Within 15 Days of the date of the bulletin, all PWR licensees are required to provide the following*

   B. *an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head, including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse.*

<u>PG&E Response:</u>

Inspections of the head penetration canopy seal welds which are conducted each refueling outage include a detailed and thorough examination of the area inside the CRDM cooling shroud immediately above the head insulation. Recently identified lower canopy seal weld leaks have resulted in small amounts of boric acid accumulation that typically consist of a film limited to the canopy seal weld and immediate surroundings. Based on thorough inspections of the top of the mirror insulation, PG&E is confident that leakage from sources above the insulation have been identified and corrected during past refueling outages. When boric acid leakage from above the reactor pressure vessel head insulation was observed to flow past the insulation, PG&E performed additional inspections, assessments for head degradation, and cleaning. The insulation is clean and in good condition, and deposits on the insulation are readily identifiable. Except for one incident which is identified in Table 1, no visible leakage on the bare metal of the reactor pressure vessel head below the shroud and around the flange has been observed during normal refueling outage maintenance.

During the DCPP Unit 1 ninth refueling outage in 1999, approximately one half of the reactor pressure vessel head insulation was removed to facilitate head penetration canopy seal weld repairs. Although no formal inspection was performed, no accumulations of boric acid or indications of degradation of the reactor pressure vessel head were identified.

Given that visible boric acid leakage has been promptly identified, evaluated and corrected as noted above, PG&E is confident that there has been no reactor pressure vessel head degradation from boric acid leaks from components above the reactor head.

Pending performance of the 100 percent bare metal inspections during the next outage for each unit, PG&E believes that the industry susceptibility model, as documented in PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), EPRI, Palo Alto, CA: 2001, TP-1006284, provides a reasonable basis to conclude that degradation from head penetration cracking has not occurred at DCPP, as described below in the response to question 1.E.

<u>NRC Requested Information</u>

1. *Within 15 Days of the date of the bulletin, all PWR licensees are required to provide the following*

C. *a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs* *described in 1.A that could have led to degradation and the corrective actions taken to address such conditions.*

PG&E Response:

PG&E has conducted a thorough review of problem reports since commercial operation to identify any boric acid leaks that could have led to degradation of the reactor pressure vessel head. The results of this review for each unit are described below, and are listed in Table 1.

DCPP Unit 1:

During the Unit 1 second refueling outage (1R2) in 1988, boric acid leakage was identified from canopy seal welds on four spare head penetrations. The indications of the leakage were boric acid residue on the canopy seal welds, the head penetration tubes, and the head insulation. Signs of the residue leaking past the insulation were identified. The insulation was removed and inspections identified that the leakage onto the reactor pressure vessel head was minimal. The head was cleaned and inspected. No degradation of the reactor pressure vessel head was identified. The leakage was repaired by removing the threaded, seal welded head adapter, and installing a welded cap. Subsequent inspections of the repaired head penetrations have indicated that the repairs were effective.

Following 1R2, a program was implemented for both units to inspect the lower canopy seal welds during each refueling outage using procedure ISI X-CRDM. Since 1998, several other lower canopy seal weld leaks have been identified. These leaks were small, and boric acid deposits from this leakage were confined to a few square inches of the canopy seal weld and head penetration tube. The leakage did not run or drip onto any other surface, including the insulation and the reactor pressure vessel head. The canopy seal weld leaks were repaired during the outage in which they were identified.

Evidence of leakage from other components has been minor, and is described in Table 1.

DCPP Unit 2:

As a result of the inspections required by the program discussed above, one lower canopy seal weld leak was identified in the Unit 2 eighth refueling outage (2R8). This leak was small, and boric acid deposits from the leakage were confined to a few square inches of the canopy seal weld and head penetration tube. The leakage did not run or drip on any other surface. The canopy seal weld leak was repaired during the 2R8. Subsequent inspections of the repaired canopy seal weld have indicated that the repair was effective.

Reactor pressure vessel head vent valve leakage was identified at the end of the Unit 2 third refueling outage. The leakage was directed away from reactor pressure vessel head components, including the reactor pressure vessel head, into a leak-off container, and directed to the refueling cavity. The leakage did not result in deposits on the reactor pressure vessel head.

During the Unit 2 tenth refueling outage in 2001, a leaking intermediate canopy seal weld leak was identified. Boric acid was deposited on the CRDM housing, an adjacent ventilation component installed to direct ventilation properly, and the CRDM coil stack. Minor amounts of leakage dripped onto the surface of the head insulation. The inspection and evaluation of the leak concluded that the insulation prevented any leakage from reaching the reactor pressure vessel head.

Evidence of leakage from other components has been minor, and is described in Table 1.

Based on the preceding information, PG&E is confident that there has been no leakage from reactor pressure vessel head components that could have led to degradation of either the Unit 1 or Unit 2 reactor pressure vessel head.

<u>NRC Requested Information</u>

1. *Within 15 Days of the date of the bulletin, all PWR licensees are required to provide the following*

   D. *your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s) scope, frequency, qualification requirements, and acceptance criteria.*

<u>PG&E Response:</u>

PG&E, in response to NRC Bulletin 2001-01, committed to perform bare metal effective visual inspections of 100 percent of the penetration tubes during the next scheduled refueling outage for each unit. PG&E will have completed the analysis to support the inspections as qualified visual inspections prior to the Unit 1 eleventh refueling outage scheduled to begin April 28, 2002. In conjunction with the Bulletin 2001-01 inspections, PG&E will also perform inspections in both units to identify any degradation of the reactor pressure vessel head, including thinning, pitting, or other forms of degradation, such as the degradation of the reactor pressure vessel head identified at Davis-Besse. The next refueling outage for Unit 2 is currently scheduled for February 2003.

Method:

The visual inspections under the mirror insulation will be performed using remote examination equipment.

Personnel qualifications:

Personnel performing the remote examination of the bare metal reactor head will be certified at a minimum as VT-2 level II visual examiners in accordance with the requirements of ASME Section XI, 1989 Edition or later approved code editions.

Personnel performing the final evaluation of examination findings will be certified VT-2 level II or III.

Examination system qualification:

The remote examination system will provide visual resolution equivalent to a direct VT-2 visual as specified in the 1992 Edition of ASME Section XI Article IWA-2212 and ASME Section V Article 9 paragraph T-942. The remote examination system and procedure will be demonstrated to resolve a near vision test chart meeting the requirements of ASME Section XI Article IWA table 2210-1 for VT-2 examination.

Acceptance criteria:

Any accumulations of boric acid residue on the reactor pressure vessel head will be investigated to determine the origin of the deposit. Consistent with the ASME Code, discolored surfaces or areas with boric acid buildup will be given particular attention to determine if the surface below the residue is sound, to the extent possible with visual examination equipment. If necessary, supplemental investigation aids such as scrapers/brushes, compressed air and water washing will be applied to suspect areas to assist in the resolution of these areas.

As described in PG&E letter DCL-01-092, "Response to NRC Bulletin 2001-01, 'Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,'" dated August 30, 2001, if head penetration leakage is found in the course of the visual inspections required by NRC Bulletin 2001-01 then the remaining tubes will be examined using appropriate nondestructive examination methods (e.g., volumetric examination). Defects will be repaired or evaluated using qualified ASME Section XI plan or approved alternative.

Boric acid residue whose source is determined to be other than from a penetration tube juncture will be evaluated as noted above. Additional corrective measures regarding the termination of the leak source and the arrest of any corrosive attack of the reactor pressure vessel head will be employed.

Frequency:

The frequency of future inspections beyond those currently scheduled will be based on DCPP inspection results, the Davis-Besse root cause analysis, industry inspection results, and industry initiatives.

In addition to the inspections required by NRC Bulletins 2001-01 and 2002-01, PG&E has committed to perform a volumetric inspection of the DCPP Unit 2 reactor pressure vessel head penetrations as part of the industry response to Generic Letter 97-01. The inspection will be performed during the Unit 2 twelfth refueling outage, currently scheduled to begin in October 2004. This commitment is documented in PG&E letter DCL-00-156, "Revised Schedule for Reactor Vessel Closure Head Penetration Inspection," dated December 12, 2000.

<u>NRC Requested Information</u>

1. *Within 15 Days of the date of the bulletin, all PWR licensees are required to provide the following*

   E. *your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to Item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:*

      (1) *If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.*

      (2) *If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

PG&E Response:

Based on the information contained in the responses to the preceding questions, and for the following reasons, PG&E has concluded it has reasonable assurance that the reactor pressure vessel head, head penetrations and RCPB for DCPP Units 1 and 2 are capable of fulfilling all applicable licensing and design basis requirements. During the next refueling outage for each unit, PG&E will perform an inspection as described in the response to question 1.D. Any leakage, degradation or other conditions adverse to quality will be appropriately addressed as stated in the response to question 1.D. Specific licensing basis requirements are addressed below.

The NRC Bulletin 2002-01 section entitled Applicable Regulatory Requirements cites the following regulatory requirements as providing the basis for the bulletin assessment:

- Appendix A to 10 CFR Part 50, *General Design Criteria for Nuclear Power Plants*
  - Criteria 14 – *Reactor Coolant Pressure Boundary*
  - Criteria 31 – *Fracture Prevention of Reactor Coolant Pressure Boundary, and*
  - Criteria 32 - *Inspection of Reactor Coolant Pressure Boundary*
- Plant Technical Specifications
- 10 CFR 50.55a, Codes and Standards, which incorporates by reference Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, of the *ASME Boiler and Pressure Vessel Code*
- Appendix B of 10 CFR Part 50, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*, Criteria V, IX, and XVI
- NRC Generic Letter 88-05

General Design Criteria (GDC):

The Bulletin states that the applicable GDC include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 31 specifies that the RCPB be designed with sufficient margin to assure that the probability of rapidly propagating fracture is minimized. GDC 32 specifies that components that are part of the RCPB be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity.

As part of the original design and licensing of DCPP, PG&E demonstrated that the design of the RCPB meets these requirements. DCPP complied with these criteria in part by: 1) selecting corrosion resistant Alloy 600 and other austenitic and ferrous materials with extremely high fracture toughness for RCPB materials; and 2) following NRC approved codes and standards for fabrication, erection, and testing of the pressure boundary parts. As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32, respectively, were satisfied during the initial design and licensing, and continue to be satisfied during operation, even though stress corrosion cracking has been identified in other reactor pressure vessel heads in the industry.

DCPP Units 1 and 2 are in the moderate susceptibility range for stress corrosion cracking based upon the MRP-48 susceptibility rankings. To date, industry penetration cracking inspection results have been very consistent with the susceptibility ranking. Plants that are comparable to DCPP have performed both visual inspections and non-visual NDE, and have not identified indications of cracking.

D.C. Cook Unit 2 is a Westinghouse 4-loop plant that is higher in the susceptibility ranking than DCPP Units 1 and 2, and is similar in design to DCPP Units 1 and 2. The reactor pressure vessel head at D.C. Cook Unit 2 was inspected with both visual and non-visual NDE in the spring of 2002. No flaws or leakage were identified in any penetrations.

Millstone Unit 2 is a CE plant that is higher in the susceptibility ranking than DCPP Units 1 and 2. During the current refueling outage, non-visual NDE was performed, and non-through-wall flaws below the weld were identified. While flaws like these would require evaluation or repair, the flaws would have to propagate through the RCPB before degradation of the reactor pressure vessel head could begin.

The recently released draft probable cause summary from Davis-Besse concludes that the head penetration crack was through-wall for approximately two to four cycles. Reactor pressure vessel head degradation occurred over the course of several years. The probable cause summary also notes that indications of boric acid leakage existed for some time. The Davis-Besse estimated corrosion rates were noted as being compatible with the EPRI Boric Acid Corrosion Guidebook. Given this information, PG&E continues to believe that the amount of degradation which could occur from a through-wall leak in a head penetration tube in less than one cycle would not affect the ability of the reactor pressure vessel head to fulfill licensing and design basis requirements.

Based upon the industry susceptibility ranking and their agreement with inspection results to date, PG&E is confident that DCPP Units 1 and 2 are unlikely to have any leakage from head penetration cracking. Even assuming leakage, PG&E expects the results to be typical of plants like Oconee, Crystal River, and others that that have found a small popcorn like deposit on the reactor pressure vessel head near the penetration tube, and not leakage that would cause the degradation observed at Davis-Besse.

Plant Technical Specifications:

The limits for DCPP RCPB leakage are provided in Technical Specification (TS) 3.4.13, and are stated in terms of the amount of leakage (i.e., 1 gallon per minute for unidentified leakage; 10 gpm for identified leakage; and no leakage from a non-isolable fault in the RCPB). Routine surveillance testing is performed to ensure these requirements are met. Based on industry experience, leaks from reactor coolant system Alloy 600 penetrations have been well below the sensitivity of on-line leakage detection systems. If measurable leakage is detected by the on-line leak detection systems, the leak will be evaluated per the TS, and the plant will be shut down if required. Upon detection and identification of a leak, corrective actions will be taken to restore RCPB integrity. PG&E continues to meet the requirements of this TS.

Inspection Requirements (10 CFR 50.55a and ASME Section XI):

The Bulletin describes the requirements for inspection in accordance with the ASME Code, detection of leakage from insulated components, and the acceptance standards if through wall leakage is detected. PG&E has complied with the inspection requirements for insulated components as part of the DCPP inservice inspection program.

Since the head is insulated, and the CRDM nozzles do not represent a bolted connection, the Code permits these inspections to be performed with the insulation left in place. PG&E also complies with the requirements of Generic Letter 88-05 by performing walkdowns during refueling outages and other shutdowns as described in procedure STP R-8C. If conditions are identified in the course of these inspections, corrective actions are performed, including supplemental examinations, repairs and/or evaluations, and inspections for consequential degradation of carbon steel or low alloy steel.

Quality Assurance Requirements (10 CFR.50, Appendix B):

The Bulletin states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements, as required by Appendix B, Criteria V (Instructions, Procedures, and Drawings) and Appendix B, Criteria IX (Control of Special Processes). DCPP programs comply with these standards.

As described above, DCPP has committed to perform visual bare metal inspections of the reactor pressure vessel head. The inspections will be conducted by qualified personnel using qualified procedures, in accordance with Appendix B requirements.

Criterion XVI of Appendix B states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions.

If any cracking, leakage or degradation is detected during the reactor head and head penetration inspections described above, corrective actions will be taken in accordance with the DCPP corrective action program and plant procedures. Any RCPB leakage or degradation would be considered a significant condition adverse to quality and appropriate actions, including performing a cause analysis, will be taken.

In consideration of potential conditions adverse to quality, PG&E has been actively participating in industry organizations (Westinghouse Owners Group and Material Reliability Program) and continues to be aware of industry experience.

NRC Generic Letter 88-05:

As discussed above, PG&E has implemented the inspection and walkdown requirements of Generic Letter 88-05.

Based upon the evaluation and information provided above, and continued compliance with the TS, PG&E believes that DCPP Units 1 and 2 will continue to meet the regulatory requirements described in NRC Bulletin 2002-01, until the inspections are performed in the next refueling outage for each unit.

<u>NRC Requested Information</u>

2. *Within 30 days after plant restart following the next inspection of the reactor pressure vessel head to identify any degradation, all PWR addressees are required to submit to the NRC the following information:*

    A. *the inspection scope (if different than that provided in response to Item 1.D.) and results, including the location, size, and nature of any degradation detected,*

    B. *the corrective actions taken and the root cause of the degradation.*

<u>PG&E Response:</u>

PG&E will submit the information as requested.

NRC Requested Information

3. *Within 60 days of the date of this bulletin, all PWR addressees are required to submit to the NRC the following information related to the remainder of the reactor coolant pressure boundary:*

   A. *the basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.*

PG&E Response:

PG&E will submit the information as requested by May 17, 2002.

Table 1
DCPP Reactor Head History Review for Leakage

Date or Outage	Description of Leak	Inspection, Evaluation or Repair Description
Unit 1		
1R2 - 3/88	Due to industry experience, DCPP inspected the reactor pressure vessel head spare penetration canopy seal welds. Leaks were discovered at L-5, J-5, L-9, and L-11.	This leakage was the first observed canopy seal weld leakage at DCPP. The indications of the leakage were boric acid deposits on the canopy seal welds and head penetration tubes, and head insulation. Signs of the deposits leaking past the insulation were identified. The insulation was removed. Inspections identified that leakage onto the reactor pressure vessel head was minimal. The head area was cleaned and inspected, and no wastage was identified. The repair included removing the threaded, seal welded head adapter and installing a welded cap.
1R3 - 12/89	Reactor head vent valve leakage was detected on 12/8/89 during 1R3.	Leakage occurred for a very short time. The location of the leakage was onto the ventilation ducting and not directly onto the reactor head. A drip collection container and drain tubing was installed on 12/9/89 to route the leakage away from the reactor pressure vessel head and direct flow to the refueling cavity.
1R4 - 3/91	Reactor head vent valve isolation valve RCS-1-604 had a packing leak that was identified during the post-installation hydro test. This was detected during the valve installation and corrected prior to return to service.	The leakage consisted of clean primary water, and did not contain boric acid. Since the leakage was detected during the post-installation hydro test, the duration was brief. No boric acid leaked onto the reactor pressure vessel head.
1R5 - 10/92	Canopy seal weld leak at J-7.	The seal weld leaked with a resulting accumulation of boric acid. There was a relatively small accumulation of boric acid that reached the adjacent insulation. The

Date or Outage	Description of Leak	Inspection, Evaluation or Repair Description
		leakage was minor and did not leak past the reactor pressure vessel head insulation. The leak was repaired by installing a mechanical clamp.
1R7 - 10/95	Canopy seal weld leak at R-11.	Based on the visual inspection of the canopy seal and the surrounding penetrations and insulation, the leak rate was too low to quantify, classify or detect during cycle 7 operation. The leakage was minor and did not leak past the head insulation. The leak was repaired with a weld overlay
1R7 - 11/95	Dry boric acid identified on reactor pressure vessel studs 1, 2, and 54. Borated water was spilled onto the reactor pressure vessel head during valve testing during 1R7. This was identified during reactor reassembly.	The area was cleaned to remove the dry boric acid. There was no damage done to the reactor pressure vessel head or studs.
1R7 - 10/95	Preemptive mechanical clamps installed at spare locations E-7, E-5, E-11, G-7, G-9, G-11 and J-9.	No leakage occurred.
1R8 - 5/97	A deposit on the horizontal surface of conoseal L-1 was identified during the normal operating pressure/normal operating temperature (NOP/NOT) walkdown at the end of 1R8. The conoseal was inspected during 1R9 and no leakage was noted. The deposit was confirmed to be a film of dust as opposed to boric acid.	No leakage occurred.
1R9 - 2/99	Reactor head vent valve isolation valves RCS-1-663 and RCS-1-662 were identified with dry boric acid leaks at the packing.	No leakage onto the reactor pressure vessel head occurred. The boric acid was cleaned and the valves were repacked.
1R9 - 2/99	Canopy seal weld leaks at A-5 and E-15.	The leaks were small resulting in very small deposits of boric acid on the canopy seal and penetration tubes only. No leakage onto the reactor pressure vessel head

Date or Outage	Description of Leak	Inspection, Evaluation or Repair Description
		occurred. The boric acid was cleaned and the leaks were repaired with a weld overlay.
1R10 - 10/00	Reactor head vent valve leakage.	Based on video and photos available, wet leakage from the head vent valve drain trough occurred. The leakage was onto the stud hoist rail and hoists, the ventilation ducting and some traces onto DRPI coil stacks. Inspections were performed of the affected areas. During the course of reactor disassembly the mirror insulation outside of the ventilation shroud was removed. No traces of boric acid on the reactor pressure vessel flange or head were identified. In addition, examinations conducted during ISI exam X-CRDM on 10/10/2000 inside the CRDM shroud did not detect any notable boric acid deposits.
Unit 2		
2R2 - 11/88	Conoseal leakage. There was dry boric acid at 4 of 5 electrical connectors for incore thermocouples.	The dry boric acid was cleaned, the connectors were retightened, and no leakage identified during the NOP/NOT walk down. No leakage onto the reactor pressure vessel head occurred.
2R2 - 11/88	Flange leaks at restricting orifices upstream of the reactor head vent valves.	The bolting was replaced and torqued. The dry boric acid was cleaned. No leakage onto the reactor pressure vessel head occurred.
2R3 - 4/90	Conoseal leakage.	There was no active leakage observed. The deposit was identified as dry boric acid. The dry boric acid was cleaned. No leakage onto the reactor pressure vessel head occurred.
2R3 - 4/90	Reactor head vent valve leakage was identified at the end of 2R3.	No leakage onto the reactor pressure vessel head occurred. The leakage was directed by the leak off container and tubing away from the reactor head and to the refueling cavity.

Date or Outage	Description of Leak	Inspection, Evaluation or Repair Description
2R7 - 4/96	Dry boric acid leak at the top of part length CRDM F-4	No leakage was identified; however dry boric acid was identified. The dry boric acid was cleaned. This condition has been monitored and there has been no additional leakage that could affect the reactor pressure vessel head.
2R8 - 2/98	L-11 canopy seal weld leak.	Boric acid residue was limited to the canopy seal and the head penetration tube. No leakage onto the reactor pressure vessel head occurred. The leak was repaired by installing a mechanical clamp.
2R10 - 5/01	H-10 Intermediate canopy seal leak.	Based on pictures of the reactor pressure vessel head insulation no boric acid leaked onto the reactor pressure vessel head. The rod position indicator coil stack, the ventilation component, and the CRDM coil stack were cleaned. The canopy seal weld area was also cleaned. The leak was repaired with a weld overlay.