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PG&E Letter DCL-13-046

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

10 CFR 50.55a

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Response to NRC Request for Additional Information Regarding Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Pressure Test Requirements for Class 1 Reactor Vessel Flange Leakoff Lines

Reference: 1. PG&E Letter DCL-12-129, "Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Pressure Test Requirements for Class 1 Reactor Vessel Flange Leakoff Lines," dated December 20, 2012

In Reference 1, Pacific Gas and Electric Company (PG&E) submitted a request for approval to use an alternative to the ASME Section XI Code pressure test requirements for inservice inspection of Class 1 reactor vessel flange leakoff lines (Category B-P, Item B15.10) for Diablo Canyon Power Plant Units 1 and 2.

The NRC Staff provided a request for additional information (RAI) via e-mail, dated March 15, 2013. The Enclosure to this letter provides PG&E's response to the NRC RAI.

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter.

If you have any questions or require additional information, please contact Mr. Tom Baldwin at (805) 545-4720.

Sincerely,

Barry S. Allen

RNTT/4231/SAPN 50524575-5
Enclosure
cc: Diablo Distribution
cc/encl: Thomas R. Hipschman, NRC Senior Resident Inspector
Arthur T. Howell, III, NRC Region IV
James T. Policksoski, NRR Project Manager
State of California, Department of Occupational Safety and Health,
Pressure Vessel Unit

PG&E Response to NRC Request for Additional Information Regarding Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Pressure Test Requirements for Class 1 Reactor Vessel Flange Leakoff Lines

NRC Question 1:

Provide the following in regards to piping and configuration:

(1) Submit a piping and instrumentation diagram (P&ID) for the entire reactor vessel head flange leakoff system.

(2) Identify on the P&ID the pipe segments that are required to be examined in accordance with the ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P and Item Number B15.10.

(3) Identify on the P&ID the pipe segments that are accessible and inaccessible for inspection.

PG&E Response:

The following is provided in regard to piping and configuration:

(1) A P&ID for the reactor vessel head flange leakoff system for Unit 1 is presented in Attachment 1 of the enclosure. A P&ID for Unit 2 is presented in Attachment 2 of the enclosure. The leakoff lines are shown at coordinates 34-D. The portions subject to examination extend from the reactor vessel to and including valves RCS-8069A and B.

- (MB) refers to the 3 foot thick missile barrier wall, often referred to as the crane wall. This wall stands nominally 33 feet outboard of the biological shield wall.
- (IMB) refers to the area inside of (or enclosed by) the MB.
- (OMB) refers to the area outside of the MB.
- (BS) refers to the biological shield wall, often referred to as the reactor shield wall. This wall is approximately 4 feet thick at the subject penetration.
- (IBS) refers to the area inside of (or enclosed by) the BS.
- (OBS) refers to the area outside of the BS.

(2) The pipe segments that are required to be examined in accordance with the ASME Code Section XI, Table IWB-2500-1, Examination Category B-P and Item Number B15.10 are identified on the attached P&IDs and extend from the connection below the vessel flange penetrations through 1 inch by ¾ inch reducers. The pipe segments continue down and across the biological shield annulus and through the BS, then across and down around the reactor coolant loop area, down and through the MB, terminating at valves RCS-8069A and B. On the P&ID, these portions commence at the reactor vessel and end at valves RCS-8069A and B.

(3) **The pipe segments that are accessible for examination:** The pipe segments accessible for examination with the refueling cavity filled are those portions outside the BS, i.e., across and down around the reactor coolant loop area, down and through the MB to valves RCS-8069A and B. This portion constitutes the majority of the piping (approximately 90 percent) and is uninsulated except for the portions immediately adjacent to the MB which includes valves RCS-8069A and B. On the P&ID, this portion extends from outside the BS through the MB to valves RCS-8069A and B.

The pipe segments that are inaccessible for examination when the refueling cavity is filled: The portion that is inaccessible when pressurized (i.e., with the refueling cavity filled) extends from the connections at and through the vessel flange down and across the biological shield annulus and through the BS penetration. This portion is also uninsulated. On the P&ID, this is shown extending from the reactor vessel to inside the BS.

The pipe segments that are accessible for examination when the refueling cavity is empty: The pipe inside the BS annulus, including the lower portion of the downcomers from the vessel flange, the crossing of the annulus and the pipe inside the penetration sleeve is visible from inside the biological shield annulus when the refueling cavity is empty (i.e., the pipe is depressurized) and the access manways are opened.

The pipe segments that are inaccessible for examination at any time: The vertical portion extending from the vessel flange connections down to the open biological shield annulus (approximate length of each line is about 5 1/2 feet) passes through the vessel flange and insulation. Then it is tightly fitted between the vessel insulation and the concrete and is not directly accessible at any time.

NRC Question 2:

Provide the following in regards to leakage visual examination:

(1) For the accessible portion of the subject piping, describe exactly how the visual examination will be performed to identify the pipe through-wall leakage.
(2) If the pipe is insulated, discuss whether the insulation will be removed. If insulation will not be removed, discuss how the pipe would be examined to identify the potential pipe through-wall leakage.
(3) If the subject pipe is located in a high elevation or far away location from the examiner, describe how the potential pipe through-wall leakage can be identified.

(4) Discuss how the operator distinguishes a pipe through-wall leakage from various other leakage sources (e.g., bolted joint [flange] leakage, valve in-line leakage, and leakoff through the O-ring, etc.).

PG&E Response:

The following is provided in regard to leakage visual examination:

(1) For the accessible portion of the subject piping from the BS to the start of insulation inside the MB, the external exposed surfaces of the pipe shall be examined using the direct VT-2 method by certified VT-2 examiners (VT-2 method is defined in ASME Code Section XI). When the surfaces are accessible for indirect or remote VT-2 visual examination, the pipe and the surrounding areas (including floor areas or equipment surfaces located underneath the piping) shall be examined from the floor elevation for evidence of leakage using the visual examination VT-2 method by certified VT-2 visual examiners.

(2) For the insulated portion on either side of the MB, insulation will not be removed. The horizontal surfaces of insulation shall be directly examined at each insulation joint. Surrounding areas (including floor areas or equipment surfaces located underneath the components and on the MB) shall also be examined for evidence of leakage using the visual examination VT-2 method by certified VT-2 examiners.

(3) The portion of the pipe extending from the BS across the reactor coolant loop area is in a high elevation (approximately 24 feet above the floor), but an unobstructed view is available for the examiner from the floor elevation. The examiner shall perform remote VT-2 examination of this portion of the pipe and will additionally examine the floor and equipment surfaces underneath the pipe as discussed in (1) above.

(4) There are no bolted flanges or other mechanical connections in these lines, except the valves RCS-8069A and B packing glands. The portions of the pipes outside the BS are also relatively distant from other potential leakage sources. Accordingly, any leakage detected by certified VT-2 visual examiners in the portions of piping outside the BS (except at the packing glands of valves RCS-8069A and B) shall be considered as through-wall leakage.

The portion of piping from the connection under the reactor flange down and across the biological shield annulus and through the BS has no mechanical connections but is subject to potential contact by leakage from the refueling cavity seal. Cavity seal leakage, if present, would be deposited at low temperature and would be associated with dripping from above the pipe (including splash marks) continuing around and down the pipe to areas

underneath. This can be distinguished from through wall leakage which would tend to be localized on the pipe surface and have no source from above. Through wall leakage resulting from O-ring failure would be characteristic of localized deposition at high temperature such as a "volcano cone" deposit.

NRC Question 3:

Provide the following:

(1) For the pipe segments that are inaccessible for inspection, discuss how the structural integrity of the inaccessible pipe segments will be demonstrated.
(2) Discuss the degradation history of the subject piping (e.g., wall thinning, cracking, etc.).
(3) Provide the material specification of the affected piping and associated welds.
(4) Provide the design pressure and wall thickness of the leakoff lines.

PG&E Response:

The following is provided in regards to structural integrity:

(1) The portion of piping from the connection at the reactor flange down and across the biological shield annulus and through the BS is inaccessible when the refueling cavity is filled, and is only accessible when the cavity is drained and one or more of the excore instrument access manways are removed. These manways provide access to the reactor vessel nozzles annulus and the portion of the subject piping inside the BS. The lower portion of the downcomer from the connection below the reactor flange, the crossing of the biological shield annulus and the BS penetration is thus accessible for direct VT-2 examination only when the cavity is drained and the pipe is depressurized. This portion will be directly examined when one or more excore manways are removed during refueling outages, such as in conjunction with examination of the reactor nozzle to safe end dissimilar metal welds or when servicing the excore instrumentation. The upper portion of the downcomer from the opening at the reactor flange, down to the open area of the annulus (approximate length of each line is about 5 1/2 feet), passes through the vessel flange and is then tightly fitted to the vessel insulation and also close to the concrete structure such that it is not accessible for direct examination. This portion will be examined by checking the bottom of the pipe and adjacent surfaces for evidence of leakage from the pipe above. As with the accessible portion in this area, the pipe will not be pressurized during this examination. These examinations will demonstrate structural integrity of the piping.

(2) The subject piping was constructed to ASME Code Class 1 standards of all stainless steel materials and has no degradation history.

(3) The material specification of the affected piping is American Society for Testing and Materials (ASTM) A213 Grade TP 316 seamless stainless steel tubing, ¾ inch outside diameter by 0.095 inch wall thickness. Socket weld fittings are ASTM A182 Gr F316 or ASTM A403 Gr WP316 stainless steel. Valves RCS-8069A and B are ASTM A182 Gr F316 forged stainless steel. Welds are stainless steel, currently specified as ER-316.

(4) Design pressure of the leakoff lines is 2500 psig at 650°F. The wall thickness of the seamless stainless steel tubing is 0.095 inch (PG&E Piping Specification is "S").

NRC Question 4:

Provide the following related to normal operation:

(1) Describe the system alignment and configuration of the leakoff lines during normal operation (e.g., valve positions, alarm setpoints, etc.). Pictorial representations via the above requested P&IDs may be helpful.

(2) When leakage is detected via high temperature alarm initiation, discuss the procedures in response to that leakage in the O-ring leakoff line (e.g., describe the actions that would be taken after the alarm is initiated).

(3) If the leakoff line itself was degraded (leaking) and diverting leakoff, discuss how it would be determined if the leakoff line itself was leaking, once the above alarm actuated.

PG&E Response:

The following is provided related to normal operation:

(1) The inboard leakoff line (S-22-3/4) is aligned with valve RCS-8069B normally open. Alignment is through valve RCS-8032 past TE-401 to the reactor coolant drain tank (RCDT).

The outboard leakoff line (S-18-3/4) is isolated with valve RCS-8069A normally closed.

These lines are not normally pressurized. During refueling outages with the refueling cavity flooded, the lines are pressurized due to cavity elevation head only.

TE-401 provides an alarm in the control room when the line temperature exceeds 120°F.

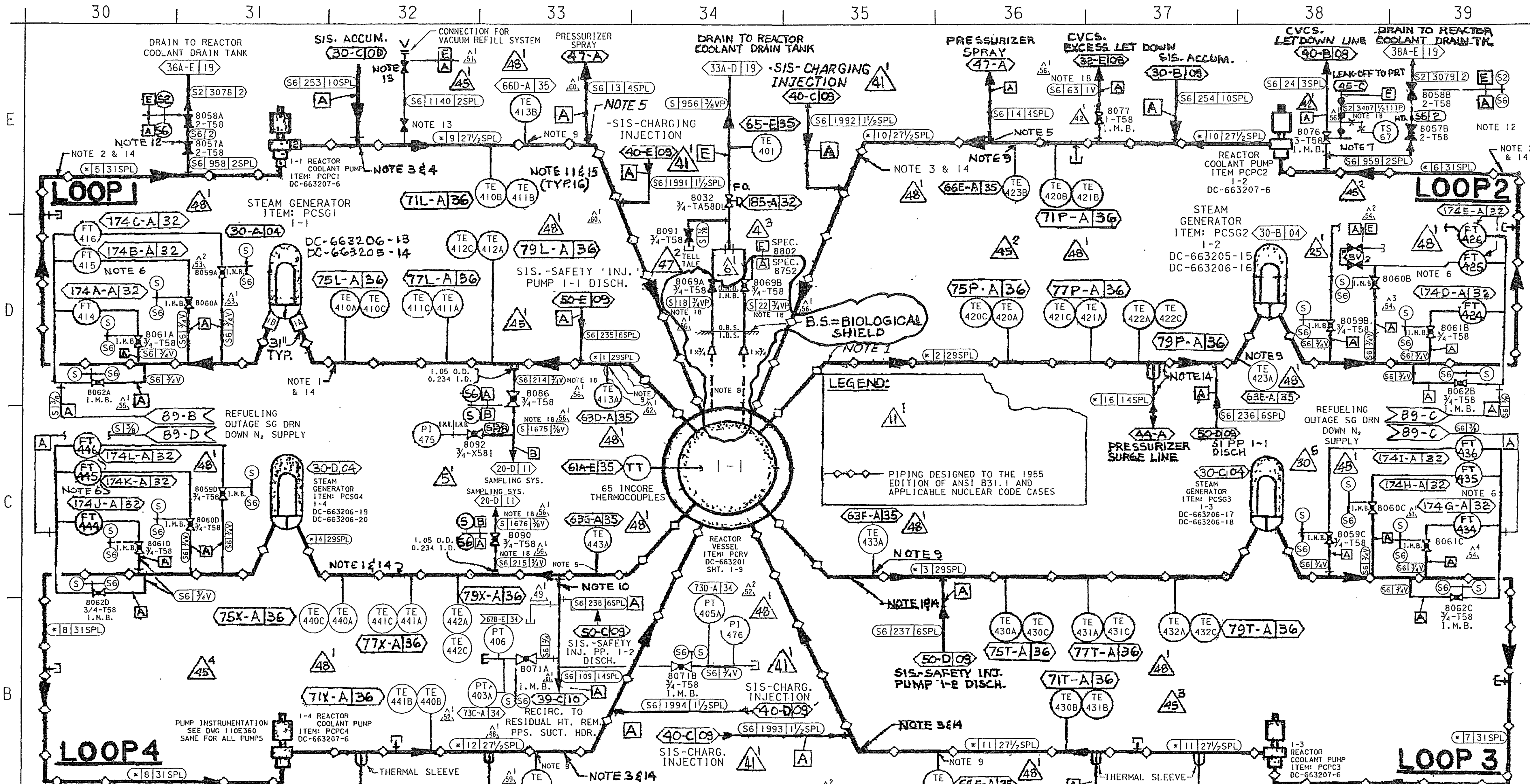
(2) When leakage is detected via high temperature alarm initiated by instrument TE-401, valve RCS-8032 would be closed from the control room, the containment would be entered, and valve RCS-8069B would be closed by

manual operator action, isolating the inboard leakoff line. Valve RCS-8069A would then be opened by manual operator action, aligning the outboard leakoff line to valve RCS-8032 and past TE-401 to the RCDT; then valve RCS-8032 would be reopened from the control room. The system would then be in configuration to monitor for leakage past the second O-ring seal. If a leakage past the second O-ring seal is detected via high temperature alarm initiated by instrument TE-401, the operator would be referred to Technical Specification 3.4.13 "RCS Operational LEAKAGE" for applicable required action(s), completion time(s) and compliance requirements.

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

Attachment 1

Piping and Instrumentation Diagram for Reactor Coolant System for Unit 1



DRAIN TO RCDT

CVCS. REGEN. HEAT EXCH. OUTLET NORMAL CHARGING

CVCS. REGEN. HEAT EXCH. ALT. CHARGING

SIS. ACCUM.

DRAIN TO REACTOR COOLANT DRAIN TK.

REF: DWG DC-663220-1

NOTES:

1. 29" INSIDE DIAMETER (31-D)
2. 31" INSIDE DIAMETER (30-E)
3. 27.5" INSIDE DIAMETER (32-E)
4. ALL (W) ITEM NUMBERS IN RCS ARE SHOWN WITHOUT PREFIX PGE RC.REF (W) STATUS REF. 45
5. SPRAY LINE SCOOP (33-E)
6. ELBOW FLOW METERS (30-D)
7. TS-67 IS STRAPPED TO LEAK-OFF 18" TO 24" TO VALVE (38-E)
8. HEAD GASKET MONITORING CONNECTIONS FURNISHED WITH REACTOR VESSEL (34-D)
9. WIDE RANGE RTD INSTALLED IN WELL & INSULATED, SEE DWG 663320-497 (33-B,C,D,E) (35-C) (36-B,E) (38-D)
10. LOCATE CONNECTION IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTICAL (33-B)
11. DUAL ELEMENT RTD'S (32/33-B/D) (36/37-B/D)
12. SPECIAL LINE INSULATION BEFORE ISOLATION VALVE. INCLUDING VALVE (30-E)
13. ANCHOR DARLING 2" DOUBLE DISC GATE VALVE DWG 6006473-2, S/C 93-6294. (32-E)
14. REACTOR COOLANT LOOPS AND PRESSURIZER SURGE LINE-FURNISHED BY (W), INSPECTION DOCUMENTATION TO ASME B&PV CODE SECT. I, 1968
15. LOOP PIPING IS PROVIDED WITH REMOVABLE, CONVENTIONAL INSULATION AT RTD LOCATIONS PER DRAWING 663320-69, 70, 83, 84, 85, 86, 102 (33-E)
16. DELETED
17. SEE SHT 7A FOR RVRLIS ALIGNMENT FOR OPERATION (30-A)
18. INSULATION ON PORTIONS OF THESE LINES HAVE ADDITIONAL BANDS INSTALLED ON 3" CENTER-TO-CENTER (MAX) SPACING. SEE DWGS 224370, 224371, 224372, 224382, 224389, 224395, 500177, 500181 AND 502183 FOR SPECIFIC LOCATIONS.

DIABLO CANYON POWER PLANT - PG&E CO.
REACTOR COOLANT SYSTEM

UNIT 1

DRAWING	SHEET	PAGE	REV
102007	3	0	62

03-17-2011	JBMX	FAZI		GREGORY E. QUITORIANO	MECHANICAL	M 23169	9/30/2012	REVISED PER DFT-7*1029
DATE	DWN	RE	IV	PROFESSIONAL ENGINEER	PE DISC.	PE#	PE EXP.	

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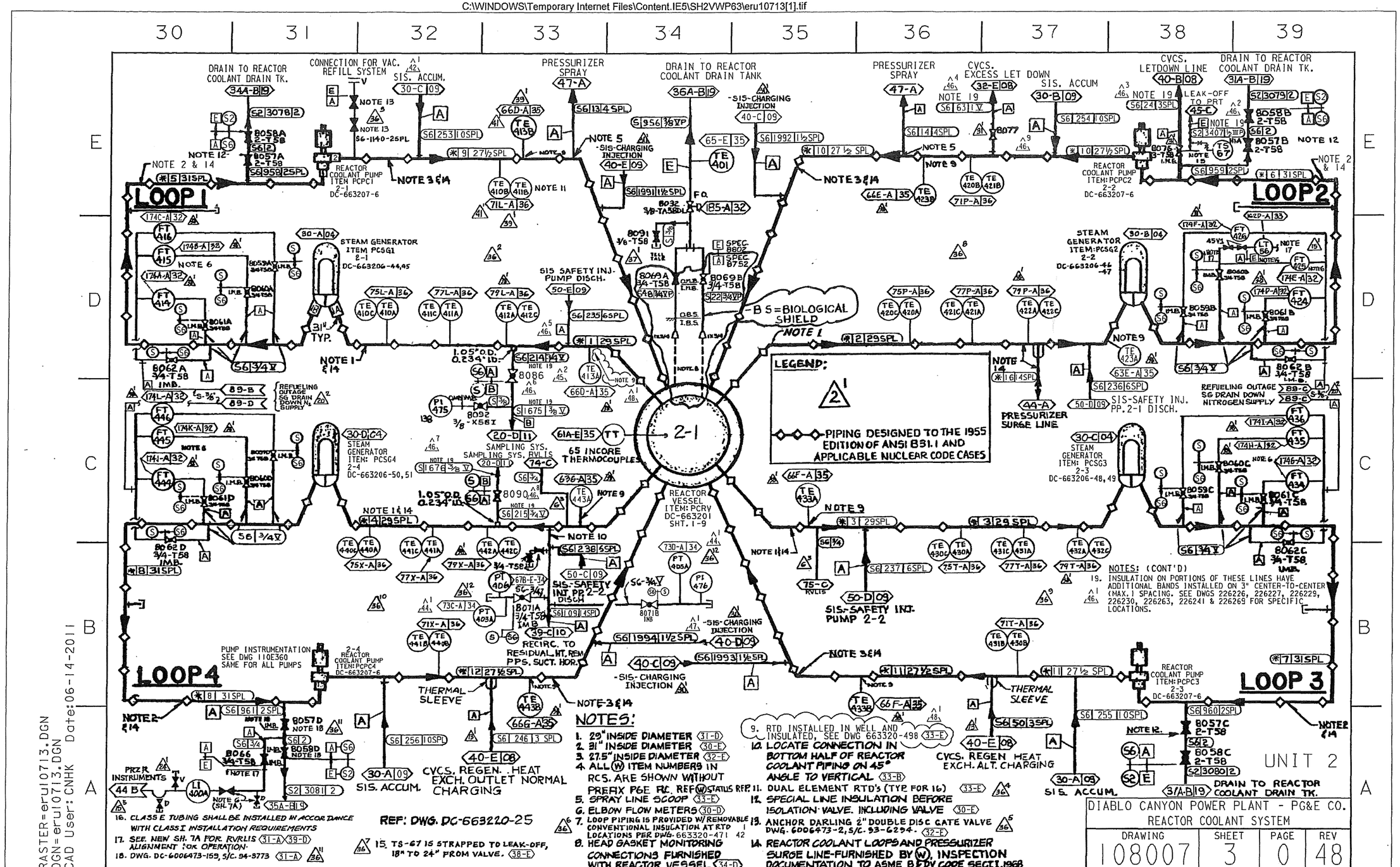
Attachment 2

Piping and Instrumentation Diagram for Reactor Coolant System for Unit 2



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DGN=eru10713.DGN
CAD User:CNHK Date:06-14-2011

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06-14-2011	CNHK	FxC2		KERSI J. DALAL	MECHANICAL	M 16690	3/31/2012	REVISED PER DFT-7*1192-0
DATE	DWN	RE	IV	PROFESSIONAL ENGINEER	PE DISC.	PE#	PE EXP.	