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RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Forwards NDE Relief Requests NDE-008 & NDE-009 for Units 1 & 2,respectively & Relief Requests 10,11 & 12 since conformance w/certain ISI requirements of ASME Section IX impractical.

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Gregory M. Rueger
Senior Vice President and
General Manager
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March 17, 1992

PG&E Letter No. DCL-92-063

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Inservice Inspection (ISI) Relief Requests

Gentlemen:

Pursuant to 10 CFR 50.55a(g)(5)(iii), PG&E is submitting the enclosed relief requests because conformance with certain inservice inspection (ISI) requirements of ASME Section XI are impractical for DCPP Units 1 and 2. PG&E believes that the relief requests are justified in accordance with 10 CFR 50.55a(a)(3) because the proposed alternatives provide an acceptable level of quality and safety, and compliance with the specified requirements would result in hardship and unusual difficulties without a compensating increase in the level of quality and safety. The enclosed relief requests provide detailed descriptions of the components, bases for requesting relief, and proposed alternative testing. The requests are summarized below.

Nondestructive Examination (NDE) Relief Request - Weld Inspections

Enclosure 1 provides Unit 1 NDE relief request NDE-008 and Unit 2 NDE relief request NDE-009. Relief is requested for 21 Unit 1 and 9 Unit 2 welds that were not 100% accessible for examination. The specific welds needing NRC approval are identified in Appendix B to the relief requests. NRC approval of these relief requests is needed by August 1992, prior to the fifth refueling outages, to ensure compliance with the 3-1/3 year periodic inspection requirements of ASME Section XI.

Note: By January 7, 1993 for Unit 1 and November 13, 1993 for Unit 2, 3-1/3 year periodic weld inspections must be either completed by PG&E in accordance with ASME requirements, or NRC-exempted as requested in the above relief requests.

System Pressure Test Relief Requests - Ten Year Pressure Testing

Enclosure 2 provides Units 1 and 2 system pressure test relief request 10 regarding ASME Section XI 10-year pressure test requirements. Relief is requested for 16 Class 1 closed-end drain lines (8 per unit). Pressure testing would require costly modifications to install test connections and result in unnecessary radiation exposure to plant personnel. Piping drawings are attached to show the Unit 1 lines and required modifications, which are also typical of Unit 2. NRC approval

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of this relief request is needed by August 1992 to allow PG&E to remove the contingency modifications and testing from the fifth refueling outage schedules.

Enclosure 3 provides Units 1 and 2 system pressure test relief request 11 regarding ASME Section XI 10-year pressure test requirements. Relief is requested from 56 Class 1 and 2 open-ended tailpipes (28 per unit). Piping drawings are attached to show the Unit 1 lines, which are also typical of Unit 2. NRC approval of this relief request is needed by February 1994, to allow PG&E to remove contingency testing from the sixth refueling outage schedules.

Enclosure 4 provides Units 1 and 2 system pressure test relief request 12 regarding 10-year ASME Section XI pressure test requirements. Relief is requested from 16 Class 1 lines (8 per unit) that are located between RCS pressure boundary isolation check valves. Piping drawings are attached to show the Unit 1 lines, which are also typical of Unit 2. NRC approval of this relief request is needed by February 1994, to allow PG&E to remove contingency testing from the sixth refueling outage schedules.

Note: By May 7, 1996 for Unit 1 and March 13, 1997 for Unit 2, 10-year periodic pressure tests must be either completed by PG&E in accordance with ASME requirements, or NRC-exempted as requested in the above relief requests.

Sincerely,

Greg Rueger

Gregory M. Rueger

Enclosures

cc: Ann P. Hodgdon
George Johnson (w/enc)
John B. Martin (w/enc)
Philip J. Morrill
Harry Rood (w/enc)
Howard J. Wong
CPUC
Diablo Distribution

5648S/85K/JHA/469

ENCLOSURE 1

REQUESTS FOR RELIEF FROM ASME SECTION XI REQUIREMENTS
INSERVICE INSPECTION (ISI) PROGRAM

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

UNIT 1 NON-DESTRUCTIVE EXAMINATION RELIEF REQUEST NDE-008 (WITH APPENDIX B)

UNIT 2 NON-DESTRUCTIVE EXAMINATION RELIEF REQUEST NDE-009 (WITH APPENDIX B)

TEN YEAR EXAMINATION PROGRAM
ASME SECTION XI SYSTEMS

REQUESTS FOR RELIEF NDE-008
TABLE 3.4

COMPONENT OR ITEM	ASME XI CODE CLASS	PROGRAM TABLE	CODE CATEGORY	CODE ITEM
PIPE WELDS	1 & 2	1.4, 2.2	B-J, C-F	B9.X, C5.X

CODE REQUIREMENTS

Volumetric examination per Appendix III for applicable Class 1 pipe welds 4 inch and greater nominal diameter and for applicable Class 2 pipe welds over 1/2 inch nominal wall thickness. Surface examination for all applicable pipe welds.

BASIS FOR REQUEST

Many welds are not 100% accessible for examination. The specific reasons are dependent on each weld's configuration, but in general the limiting features could include the following:

1. Lugs or other welded attachments.
2. Wall or floor penetrations, hangers or components closely adjacent to the examination surface.
3. Surface configuration, such as local roughness or compound curvature, especially at the intrados of elbows or tees.
4. Surface obstructions, including flanges or the bevels at valve bodies or thick wall fittings. These conditions, when present, may obstruct a portion of the test surface, especially from volumetric examination due to transducer liftoff.

A list of specific limited welds with an estimate of the accessible percentage of each and the reason for limitation is included as Appendix B.

PROPOSED EXAMINATION

PG&E proposes to examine each and every scheduled weld to the fullest extent possible. If, during the course of examination, some obstruction or limitation is encountered, that limitation will be fully documented including location, percentage of total examination surface obstructed and the nature of the limitation. (Recording these limitations is required by all present PG&E nondestructive examination procedures.) At that time, the examination result will be subject to approval of the Authorized Nuclear Inservice Inspector, and a list and description of all obstructions will be forwarded to the NRC with the Report of Inservice Inspection following each refueling outage.

SCHEDULE IMPLEMENTATION

Commercial startup to 120 months of operation.

APPENDIX B TO UNIT 1 RELIEF REQUEST NDE-008
WELDS HAVING LIMITED ACCESSIBILITY TO NDE

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
CLASS 1								
1	WIB-RC-1-5SE	75	4	No scan upstream side: steam generator geometry	B5.30/ B5.50	1R2	7/88	--------
2	WIB-RC-2-6SE	65	4	No scan downstream side: steam generator	B5.50	1R4	4/91	12/14/88
2	WIB-RC-2-5	75	4	Best effort scans 2 and 3: elbow configuration	B9.11	1R4	4/91	12/14/88
3	WIB-RC-3-5SE	75	2	Best effort on downstream scan: steam generator configuration	B5.30/ B5.50		PSI	12/14/88
4	WIB-RC-4-5SE	75	2	Best effort on downstream scan: steam generator configuration	B5.30/ B5.50		PSI	12/14/88
5	WIB-RC-1-12	40	2	Best effort on downstream scan: pump body configuration. No scans 4 and 5 due to geometry.	B9.11	1R2	7/88	12/14/88
5	WIB-RC-1-6SE	65	4	Best effort on scan 5: steam generator configuration	B5.30/ B5.50		PSI	12/14/88
6	WIB-RC-2-10	75	4	Transducer liftoff due to weld face geometry	B9.12	1R4	4/91	12/14/88
6	WIB-RC-2-14	40	1	Best effort on downstream scan: pump body configuration. No scans 4 and 5 due to geometry.	B9.11	1R2	7/88	12/14/88
6	WIB-RC-2-7SE	40	4	Best effort on scans 2, 3, 4, and 5: steam generator/elbow configuration	B5.30/ B5.50	1R3	12/89	12/14/88

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
6	WIB-RC-2-9	75	4	Best effort scans 2 and 3: elbow configuration and rough surface	B9.11	1R3	12/89	12/14/88
7	WIB-RC-3-12	75	1	Best effort on downstream scan: pump body configuration	B9.11		PSI	12/14/88
7	WIB-RC-3-6SE	75	2	Best effort on scan 5: steam generator configuration	B5.30/ B5.50		PSI	12/14/88
8	WIB-RC-4-12	75	2	Best effort on upstream scan: pump body configuration	B9.11		PSI	12/14/88
8	WIB-RC-4-6SE	75	2	Best effort on scan 5: steam generator configuration	B5.30/ B5.50		PSI	12/14/88
9	WIB-RC-1-13	25	4	Best effort scan 3. No scans 2, 4, 5: pump body and weld configuration	B9.11	1R3	12/89	12/14/88
10	WIB-RC-2-15	25	4	Best effort scan 3. No scans 2, 4, 5: pump body and weld configuration	B9.11	1R3	12/89	12/14/88
10	WIB-RC-2-18BC	75	4	Partial scan 2: branch connection	B9.31		PSI	12/14/88
11	WIB-RC-3-13	75	2	Best effort on upstream scan: pump body configuration	B9.11		PSI	12/14/88
12	WIB-RC-4-13	75	2	Best effort on upstream scan: pump body configuration	B9.11		PSI	12/14/88
13	WIB-401	90	3	Scan 3 limited 6-9" by tee curvature	B9.11	1R1	12/86	12/14/88

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
13	WIB-408	75	4	No scan 5: valve	B9.11		PSI	12/14/88
13	WIB-417	75	2	No scan 3: proximity of restraint	B9.11		PSI	12/14/88
13	WIB-64	75	4	No scan 3: valve	B9.11		PSI	12/14/88
13	WIB-415B	75	4	No scan 2: valve	B9.11	1R2	7/88	10/25/89
13	WIB-416A	75	4	No scan 3: valve	B9.11	1R2	7/88	10/25/89
13	WIB-57	90	3	Limited scan 4: elbow intrados	B9.11	1R3	12/89	--------
14	WIB-419	75	3	Best effort scan 3: tee configuration	B9.11	1R3	12/89	--------
14	WIB-432B	75	4	No scan 2: valve	B9.11	1R2	7/88	10/25/89
14	WIB-433A	75	4	No scan 3: valve	B9.11	1R2	7/88	10/25/89
14	WIB-422	75	4	No scan 2: valve	B9.11	1R3	12/89	12/14/88
14	WIB-442	90	3	Scan 3 limited 6-9": elbow intrados	B9.11	1R1	12/86	12/14/88
14	WIB-444	75	4	No scan 2: branch connection	B9.11		PSI	12/14/88
15	WIB-383	90	3	Scan 2: limited at elbow intrados	B9.11	1R1	12/86	12/14/88
15	WIB-400	75	3	Partial scan 2 and 3: fitting	B9.11	1R2	7/88	12/14/88
16	WIB-66	75	2	No scan 3: hanger clamp adjacent to weld	B9.11	1R3	12/89	--------

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
16	WIB-67	60	2	Scans 2, 3, 4, 5, limited 3-11" and 13-19". Scan 2 limited 25-32" and 35-45" by rupture restraint crush box.	B9.11	1R1	12/86	12/14/88
16	WIB-68	90	2	Scan 2 limited 15-22" by crush box	B9.11	1R1	12/86	12/14/88
16	WIB-71SE	75	4	No scan 5: pressurizer surge nozzle	B5.20/ B5.50		PSI	12/14/88
16	WIB-RC-2-3	25	4	No scan 3, 4, 5: Branch connection configuration	B9.31	1R3	12/89	--------
24	WIB-124	95 Surface	3	Base metal limited: surface cond (PT)	B9.21	1R2	7/88	10/25/89
24	WIB-RC-2-12	75	3	No scan on branch connection curvature	B9.31		PSI	12/14/88
109	WIB-236	95	1	Scan 2 limited for 4": branch connection Scan 3 limited for 2": drain line	B9.11 B9.11	1R3	12/89	--------
235	WIB-5	75	4	No scan downstream side: due to elbow/valve configuration	B9.11	1R4	4/91	12/14/88
235	WIB-1	75	3	Scan 3: limited by nozzle connection curvature	B9.11	1R2	7/88	12/14/88
236	WIB-77	90	3	Scan 2 limited 4": intrados of elbow	B9.11	1R3	12/89	--------

APPENDIX B TO UNIT 1 RELIEF REQUEST NDE-008

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
236	WIB-RC-2-4	75	4	Limited scan on downstream side of branch connection due to pipe clamp	B9.31	1R4	4/91	--------
237	WIB-175	75	4	No scan downstream side: due to pipe/valve configuration	B9.11	1R4	4/91	12/14/88
238	WIB-289	90	3	Scan 3 limited 1:00-5:00 by intrados	B9.11	1R1	12/86	12/14/88
253	WIB-39	80	2	18-25" and 18-34": welded rupture restraint. Scan 3 limited from 1-7", 9-16".	B9.11	1R3	12/89	--------
254	WIB-151	75	4	No scan 3: branch connection	B9.11		PSI	12/14/88
254	WIB-RC-2-16	70	4	No scan on downstream side-branch connection	B9.31	1R1	12/86	12/14/88
254	WIB-157	80	4	Limited scan downstream: tee Limited scan upstream: elbow	B9.11	1R4	4/91	--------
255	WIB-207	60	2	Limited scans 3, 4 and 5: proximity of welded restraint	B9.11	1R3	12/89	--------
255	WIB-210	65	3	No scan 5: valve, partial scan 2: tee	B9.11	1R1	12/86	12/14/88
256	WIB-271	50	4	No scan downstream side: due to valve body configuration	B9.11	1R4	4/91	--------
256	WIB-267	75	4	No scan 5: branch connection	B9.11		PSI	12/14/88
256	WIB-276	75	4	No scan 2: valve	B9.11	1R2	7/88	10/25/89

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
727	WIB-334	50 65 Surface	1	Scans and PT limited to 1/2 of width on upstream side by code nameplate	B9.11	1R1	12/86	12/14/88
727	WIB-338	90	3	Scan 2 limited for 4": intrados of elbow	B9.11	1R3	12/89	--------
728	WIB-322SE	50	4	No scan upstream side: Nozzle configuration	B5.20/ B5.50	1R1	12/86	12/14/88
728	WIB-322	75	4	No scan upstream side: due to safe end configuration	B9.11	1R4	4/91	--------
729	WIB-319	90	4	Scans limited on upstream side 9" to 12.3" by drain line connection	B9.11	1R1	12/86	12/14/88
729	WIB-313SE	60	4	All scans limited to 1" from SE	B5.20/ B5.50	1R2	7/88	10/25/89
729	WIB-321	85	4	Best effort scan 3: taper of flange fitting	B9.11	1R3	PSI	12/14/88
730	WIB-340SE	75	4	Best effort scans 2 and 3: nozzle connection	B5.20/ B5.50	1R3	12/89	--------
1665	WIB-248	75	4	No scan 5 (valve)	B9.11		PSI	12/14/88
1993	WIB-302D	75 Surface	4	Exam limited due to weld support	B9.40	1R4	4/91	--------
2575	WIB-16	95	2	Scans 3, 4, and 5 limited over 3" area on downstream side by stanchion	B9.11	1R1	12/86	12/14/88

APPENDIX B TO UNIT 1 RELIEF REQUEST NDE-008

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	EXAM (Note 4) DATE	SER DATE (Note 5)
2576	WIB-83	95	1	4" of scan 2 limited by weld support	B9.11		PSI	12/14/88
3844	WIB-48	90	3	Scan 2 limited 4": intrados of elbow	B9.11	1R3	12/89	--------
3845	WIB-168	90	4	Limited scan upstream side: due to 4" pipe section	B9.11	1R4	4/91	--------
3845	WIB-171	75	4	No scan 2: valve	B9.11	1R2	7/88	10/25/89
3846	WIB-214	90	3	Scan 2 limited 4": intrados of elbow	B9.11	1R3	12/89	--------
4081	WIB-361	90	3	Scan 2 limited by reducer curvature	B9.11	1R1	12/86	12/14/88
<u>CLASS 2</u>								
118	WIC-27	90 Surface	2	Surface exam limited by welded restraint @ 12:00 and 6:00	C5.11		PSI	12/14/88
228	WIC-9-1	75	4	No scan 2: elbow	C5.21	1R2	7/88	10/25/89
264	WIC-264-2	75	3	Best effort scans 4 and 5: weld configuration	Required by NRC Letter of 10/16/86	1R3	12/89	--------
556	WICG-101A1-4	85	3	Upstream adjacent weld surface	C5.21		PSI	12/14/88
1357	WIC-1357E	90	3	No scan on upstream side from 8-13": unistrut	C5.21	1R1	12/86	12/14/88
1454	WIC-177	90	2	Scan 3 limited 5": proximity of hanger	C5.21	1R3	12/89	--------

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
1661	74-29A	75 Surface	2	Surface exam of hanger attachment weld limited by support structure and proximity to wall	C3.40	1R4	4/91	12/14/88
1973	WIC-184	75	3	No scan 2: flange	C5.21		PSI	12/14/88
2032	WIC-348	90	3	Scan 2 limited to 5" at 90 and 270 by tee curvature	C5.21	1R1	12/86	12/14/88
2032	WIC-353	95	3	Downstream side limited: valve	C5.21		PSI	12/14/88
2576	WIC-249	75	4	Downstream side limited: elbow intrados	C5.21		PSI	12/14/88

<u>NOTES</u>:

1. % ACCESS: Percentage accessible estimates are based on results from examinations conducted during either a refueling outage or preservice inspection, as noted in the "EXAM" column. Percentage accessible estimates are only applicable for volumetric examinations, except where surface estimates are provided. Surface examination accessibility is 100% except where so noted.

2. BASIS: The basis for each weld inaccessibility is classified as follows:

    1. Lugs or other welded attachments.

    2. Wall or floor penetrations, hangers or components closely adjacent to the examination surface.

    3. Surface configuration, such as local roughness or compound curvature, especially at the intrados of elbows or tees.

    4. Surface obstructions, including flanges or the bevels at valve bodies or thick wall fittings. These conditions when present, may obstruct a portion of the test surface, especially from volumetric examination due to transducer lift-off.

3. CODE ITEM: The ASME Section XI code item provides the specific weld examination requirements, e.g., volumetric and/or surface examination.

4. EXAM: The examination column identifies the refueling outage (and restart date) when the weld was last inspected, and provides the basis for the percentage accessibility. If the weld has not been inspected during commercial operation, preservice inspection (PSI) results provide the basis for percentage accessibility.

5. SER DATE: Indicates the NRC safety evaluation report (SER) date when the weld inspection relief was granted by the NRC. NRC approval is needed for those welds which have no SER listed. (Note: NRC SER dated 12/14/88 was based on PSI estimates for a majority of the welds. Some of these PSI estimates may have changed based on revised estimates from subsequent refueling outage examinations.)

COMPONENT OR ITEM	ASME XI CODE CLASS	PROGRAM TABLE	CODE CATEGORY	CODE ITEM
PIPE WELDS	1 & 2	1.4, 2.2	B-J, C-F	B9.X, C5.X

CODE REQUIREMENTS

Volumetric examination per Appendix III for applicable Class 1 pipe welds 4 inch and greater nominal diameter and for applicable Class 2 pipe welds over 1/2 inch nominal wall thickness. Surface examination for all applicable pipe welds.

BASIS FOR REQUEST

Many welds are not 100% accessible for examination. The specific reasons are dependent on each weld's configuration, but in general the limiting features could include the following:

1. Lugs or other welded attachments.
2. Wall or floor penetrations, hangers or components closely adjacent to the examination surface.
3. Surface configuration, such as local roughness or compound curvature, especially at the intrados of elbows or tees.
4. Surface obstructions, including flanges or the bevels at valve bodies or thick-wall fittings. These conditions, when present, may obstruct a portion of the test surface, especially from volumetric examination due to transducer lift-off.

A list of specific limited welds with an estimate of the accessible percentage of each and the reason for limitation is included as Appendix B.

PROPOSED EXAMINATION

PG&E proposes to examine each and every scheduled weld to the fullest extent possible. If, during the course of examination, some obstruction or limitation is encountered, that limitation will be fully documented including location, percentage of total examination surface obstructed and the nature of the limitation. (Recording these limitations is required by all present PG&E nondestructive examination procedures.) At that time, the examination result will be subject to approval of the Authorized Nuclear Inservice Inspector, and a list and description of all obstructions will be forwarded to the NRC with the Report of Inservice Inspection following each refueling outage.

SCHEDULE IMPLEMENTATION

Commercial startup to 120 months of operation.

APPENDIX B TO UNIT 2 RELIEF REQUEST NDE-009
WELDS HAVING LIMITED ACCESSIBILITY TO NDE

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
1	WIB-RC-1-5SE	75	4	No scan downstream side: steam generator	B5.30 B5.50	2R2	12/88	12/14/88
1	WIB-RC-1-1SE	25	3	No scan upstream side: Rx Vessel	B5.10 B5.50	2R2	12/88	10/25/89
1	WIB-RC-1-2	50	3	Limited scan upstream side: Rx Vessel	B9.11	2R2	12/88	10/25/89
2	WIB-RC-2-5SE	70	4	Scan limited by nozzle configuration	B5.30 B5.50	2R3	4/90	12/14/88
2	WIB-RC-2-1SE	25	3	No scan upstream side: Rx Vessel	B5.10 B5.50	2R2	12/88	10/25/89
2	WIB-RC-2-2	50	3	Limited scan upstream side: Rx Vessel	B9.11	2R2	12/88	10/25/89
2	WIB-RC-2-4	85	4	Scan limited due to part configuration	B5.30 B5.50	2R3	4/90	--------
3	WIB-RC-3-5SE	75	4	No scan downstream side: steam generator	B5.30 B5.50		PSI	12/14/88
4	WIB-RC-4-5SE	75	4	No scan downstream side: steam generator	B5.30 B5.50		PSI	12/14/88
5	WIB-RC-1-11	50	4	Limited scan downstream side: pump	B9.11	2R2	12/88	12/14/88
5	WIB-RC-1-6SE	75	4	No scan downstream side: steam generator	B5.30 B5.50	2R2	12/88	12/14/88

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
6	WIB-RC-2-11	50	4	Limited scan downstream side: pump	B9.11	2R2	12/88	12/14/88
6	WIB-RC-2-6SE	70	4	Scan limited by nozzle configuration	B5.30 B5.50	2R3	4/90	12/14/88
7	WIB-RC-3-11	75	4	No scan downstream side: pump	B9.11		PSI	12/14/88
7	WIB-RC-3-6SE	75	4	No scan downstream side: steam generator	B5.30 B5.50		PSI	12/14/88
8	WIB-RC-4-11	75	4	No scan downstream side: pump	B9.11		PSI	12/14/88
8	WIB-RC-4-6SE	75	4	No scan upstream side: steam generator	B5.30 B5.50		PSI	12/14/88
9	WIB-RC-1-12	95	4	Some loss of coupling when shoe bridges taper	B9.11	2R1	7/87	12/14/88
10	WIB-RC-2-12	12	4	Scan limited by pump/nozzle geometry	B9.11	2R3	4/90	12/14/88
10	WIB-RC-2-15	50	3	Weld crown "phonographic" machine surface	B9.11		PSI	12/14/88
11	WIB-RC-3-12	75	4	No scan upstream side: pump	B9.11		PSI	12/14/88
12	WIB-RC-4-12	75	4	No scan upstream side: pump	B9.11		PSI	12/14/88
13	WIB-56	70	3	No scan upstream side:	B9.11	2R2	12/88	12/14/88
13	WIB-55	65	4	No scan downstream side: branch connection	B9.31	2R2	12/88	10/25/89
13	WIB-62	75	4	No scan downstream side: valve branch connection	B9.11	2R4	10/91	12/14/88

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
13	WIB-71	90	3	Elbow intrados limits downstream scan	B9.11		PSI	12/14/88
14	WIB-104	75	3	Branch connection geometry limits scan	B9.31		PSI	12/14/88
14	WIB-92	75	4	No scan upstream side: valve	B9.11		PSI	12/14/88
14	WIB-97	75	4	No scan downstream side: valve	B9.11		PSI	12/14/88
14	WIB-88	90	3	Elbow intrados limits upstream scan	B9.11	2R1	7/87	10/25/89
15	WIB-345SE	75	4	No upstream scan due to inconnel buttering	B5.20	2R4	10/91	--------
15	WIB-322	90	3	Elbow intrados limits upstream scan	B9.11	2R1	7/87	10/25/89
15	WIB-337	90	1	Welded lugs limit scan	B9.11	2R3	4/90	--------
16	WIB-432	25	4	Scan limited by branch configuration	B9.31	2R3	4/90	--------
51	409-3A	90 surface	2	No PT on plate thickness next to wall	B10.10	2R1	7/87	10/25/89
51	898-2	90 surface	2	PT on 1/8" nearest clamp best effort	B10.10	2R1	7/87	10/25/89
55	WIB-856	99 surface	2	Best effort @ BDC poor access-support	B9.21	2R1	7/87	10/25/89
109	WIB-243	65	3	Branch connection geometry limits scan	B9.31		PSI	12/14/88

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
109	WIB-246	90	2	Rupture restraint limits scan	B9.11	2R2	12/88	10/25/89
109	WIB-249	66	1	Exam limited due to stanchion	B9.11	2R4	10/91	--------
109	WIB-253	35 75 surface	2	Rupture restraint limits scans	B9.11	2R3	4/90	12/14/88
235	WIB-1	65	3	Branch connection geometry limits scans	B9.31		PSI	12/14/88
235	WIB-6	75	4	No scan downstream side: valve	B9.11	2R1	7/87	10/25/89
235	WIB-11	75	4	No scan upstream side: reducer	B9.11		PSI	12/14/88
236	WIB-106	60	4	No scan downstream side: branch connection	B9.11	2R2	12/88	12/14/88
236	WIB-105	50	4	No scan downstream side: branch connection	B9.31	2R4	10/91	--------
238	WIB-265	60	4	No scan downstream side: valve	B9.11	2R2	12/88	12/14/88
253	WIB-39	75	2	Rupture restraint limits scans	B9.11	2R2	12/88	10/25/89
253	WIB-37	25	4	Branch connection geometry limits scans	B9.31	2R2	12/88	10/25/89
254	WIB-164	75	4	Branch connection limits scan on downstream side	B9.11	2R1	7/87	12/14/88
254	WIB-172	75	4	No scan upstream side: valve	B9.11	2R3	4/90	12/14/88
255	WIB-192	60	4	Branch connection geometry limits scan	B9.31		PSI	12/14/88
255	WIB-196	75	4	No scan upstream side: valve	B9.11	2R4	10/91	12/14/88

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
255	WIB-197	90	4	Fittings limit scan	B9.11	2R2	12/88	10/25/89
256	WIB-299	75	4	No scan downstream side: valve	B9.11		PSI	12/14/88
256	WIB-291	95	2	Scan on downstream side limited by support	B9.11	2R1	7/87	10/25/89
727	WIB-359SE	75	4	Safe end geometry limits scan	B5.20	2R4	10/91	12/14/88
727	WIB-362	60 85	2 2	Code nameplate and support limits scan on upstream side. Also surface exam limitation	B9.11	2R3	4/90	12/14/88
728	WIB-430	95	2	Grating limits scans	B9.11		PSI	12/14/88
728	WIB-423SE	90	4	Upstream scan limited by nozzle boss	B5.20	2R4	10/91	10/25/89
729	WIB-369SE	90	4	Safe end configuration limits upstream scan	B5.20	2R4	10/91	12/14/88
729	WIB-378	75	4	No scan on downstream side: valve	B9.11	2R3	4/90	12/14/88
730	WIB-380SE	75	4	Safe end configuration limits scan	B5.20	2R4	10/91	12/14/88
730	WIB-391	75	3	Reducer limits scan upstream	B9.11		PSI	12/14/88
1171	WIB-408	90 surface	1	Code ID Band limits exam to 7/8" on one side	B9.21	2R1	7/87	10/25/89
1171	WIB-410A	90 surface	2	Hanger limits access	B9.21	2R2	12/88	10/25/89
1172	989-29R	95 surface	2	Support steel top and bottom	B10.10	2R1	7/87	10/25/89

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
1990	WIB-909	90 surface	2	Hanger blocks 10% of weld	B9.40	2R4	10/91	--------
1991	WIB-514	90 surface	2	Hanger limits access	B9.40	2R2	12/88	10/25/89
1992	WIB-611	75	2	Hanger blocks 25% of weld	B9.40	2R4	10/91	--------
2576	WIB-119	90	3	Elbow intrados limits scan @ 180	B9.11		PSI	12/14/88
3844	WIB-47	75	3	No scan downstream side: tee	B9.11	2R3	4/90	12/14/88
3845	WIB-177	90	3	Elbow intrados limits downstream scan	B9.11		PSI	12/14/88
3845	WIB-181	90	3	Elbow intrados limits downstream scan	B9.11		PSI	12/14/88
3846	WIB-203	75	3	No scan upstream side: valve	B9.11	2R3	4/90	12/14/88
3847	WIB-300	95	3	No scan upstream side: tee	B9.11	2R2	12/88	12/14/88
<u>CLASS 2</u>								
112	WIC-36	90 surface	1	Welded support limits surface exam 3" @ 0 and 180	C5.11		PSI	12/14/88
554	WICG-101-1	75	4	75% nozzle	C2.20	2R4	10/91	--------
554	WICG-103-1	95	1	Weldolet on upstream side @ TDC	C5.21	2R1	7/87	12/14/88
1357	WIC-1357A	75	4	No scan on upstream side: valve	C5.21	2R1	7/87	10/25/89
1357	WIC-1357C	80 90 surface	2	No scans from 290° to 310°. Scans limited by ventilation duct and elbow intrados	C5.21	2R2	12/88	10/25/89

LINE	WELD NUMBER	% ACCESS (Note 1)	BASIS (Note 2)	LIMITATION	CODE ITEM (Note 3)	EXAM (Note 4) OUTAGE	DATE	SER DATE (Note 5)
1357	WIC-1357E	90	2	No scan upstream from 135° to 180°: penetration	C5.21	2R1	7/87	10/25/89
1454	WIC-332	70	1	Welded attachments limit scan on upstream side	C5.21	2R4	10/91	12/14/88
1454	23-36R	75 surface	2	Support blocks access to lugs nearest plates	C3.40	2R1	7/87	10/25/89
1973	WIC-325	50	2	Tee and pipe restraint structure limit scans	C5.21		PSI	12/14/88
		90 surface	2	Restraint also surface exam limitation				
3844	WIC-105	75	4	No scan on downstream side: valve	C5.21	2R3	4/90	12/14/88

<u>NOTES</u>:

1. % ACCESS: Percentage accessible estimates are based on results from examinations conducted during either a refueling outage or preservice inspection, as noted in the "EXAM" column. Percentage accessible estimates are only applicable for volumetric examinations, except where surface estimates are provided. Surface examination accessibility is 100% except where so noted.

2. BASIS: The basis for each weld inaccessibility is classified as follows:

    1. Lugs or other welded attachments.

    2. Wall or floor penetrations, hangers or components closely adjacent to the examination surface.

    3. Surface configuration, such as local roughness or compound curvature, especially at the intrados of elbows or tees.

    4. Surface obstructions, including flanges or the bevels at valve bodies or thick wall fittings. These conditions when present, may obstruct a portion of the test surface, especially from volumetric examination due to transducer lift-off.

3. CODE ITEM: The ASME Section XI code item provides the specific weld examination requirements, e.g., volumetric and/or surface examination.

4. EXAM: The examination column identifies the refueling outage (and restart date) when the weld was last inspected, and provides the basis for the percentage accessibility. If the weld has not been inspected during commercial operation, preservice inspection (PSI) results provide the basis for percentage accessibility.

5. SER DATE: Indicates the NRC safety evaluation report (SER) date when the weld inspection relief was granted by the NRC. NRC approval is needed for those welds which have no SER listed. (Note: NRC SER dated 12/14/88 was based on PSI estimates for a majority of the welds. Some of these PSI estimates may have changed based on revised estimates from subsequent refueling outage examinations.)

ENCLOSURE 2

REQUESTS FOR RELIEF FROM ASME SECTION XI REQUIREMENTS INSERVICE INSPECTION (ISI) PROGRAM

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

SYSTEM PRESSURE TEST RELIEF REQUEST 10
(includes attached drawings)

#	COMPONENT DESCRIPTION AND FUNCTION	BASIS FOR REQUESTING RELIEF AND ALTERNATE TESTING

10 16 (8 per unit) Code Class 1 closed end drain lines between first and second-off isolation valves:

No.	Line	Size	Function
1	2527	3/4"	RCP seal water supply to RCDT
2	2534	3/4"	RCP seal water supply to RCDT
3	2536	3/4"	RCP seal water supply to RCDT
4	2541	3/4"	RCP seal water supply to RCDT
5	4246	3/4"	Pressurizer spray to RCDT
6	3078	2"	RCP cold leg to RCDT
7	3079	2"	RCP cold leg to RCDT
8	3080	2"	RCP cold leg to RCDT

(Note: The line numbers are the same for Units 1 and 2.)

ASME Section XI (IWB-2500-1, IWB-5222) requires that a pressure test be performed on these piping systems once every 10 years. These closed-end lines serve as drains to the reactor coolant drain tank (RCDT). The lines are short (less than 18" on the average) and small diameter (less than 2" NPS). The lines are not normally pressurized (line pressure may exist due to first-off valve leakage and thermal effects only). Relief is requested from the 10-year pressure test requirement for the following reasons:

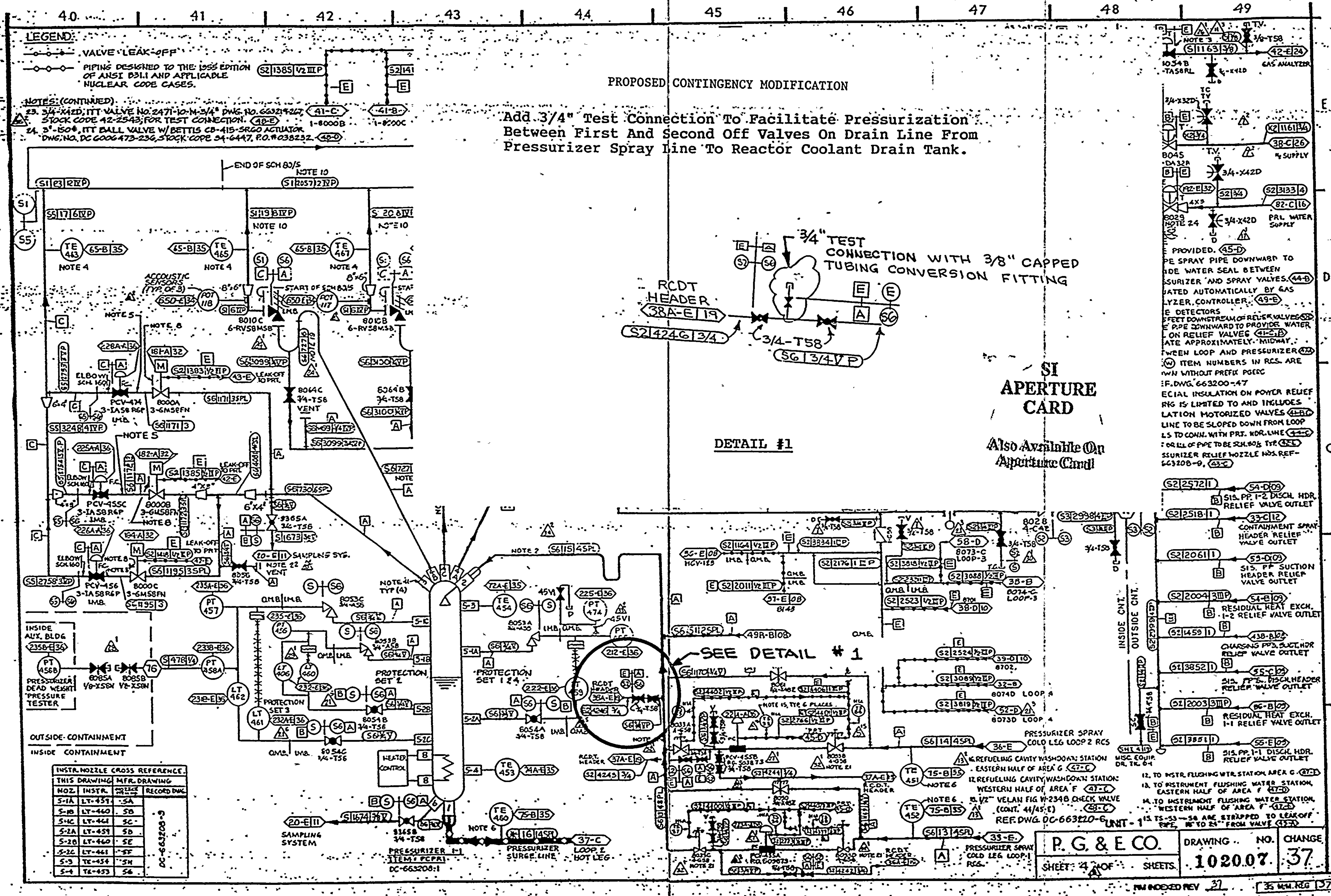
(a) Using system pressure to test these lines would require opening the first-off manual valve in Mode 3 (hot standby) to pressurize between the two valves. However, pressure testing in this manner would result in a violation of Class 1 system design requirements for double isolation valve protection.

(b) The plant design does not include provisions to allow pressure testing at or above normal system pressure. Testing during Mode 6 (refueling) would require modifications to install a test connection with an open-ended isolation valve on each line. The costs associated with design and construction modifications and unnecessary radiation doses to plant personnel would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety.

(c) Testing during Mode 6 without modifications would require defueling the reactor and repressurizing the primary system, extending the critical path outage time by approximately 12 days and resulting in an unnecessary hardship without a compensating increase in the level of quality and safety.

Alternatively, these line segments will be visually inspected during routine walkdowns, per the following procedures, to assure their continued integrity: Each refueling outage per STP R-8A, "RCS Operational Pressure Leak Test," and cold shutdowns of sufficient duration per STP R-8C, "Containment Walkdown for Evidence of Boric Acid Leakage."

RR#10



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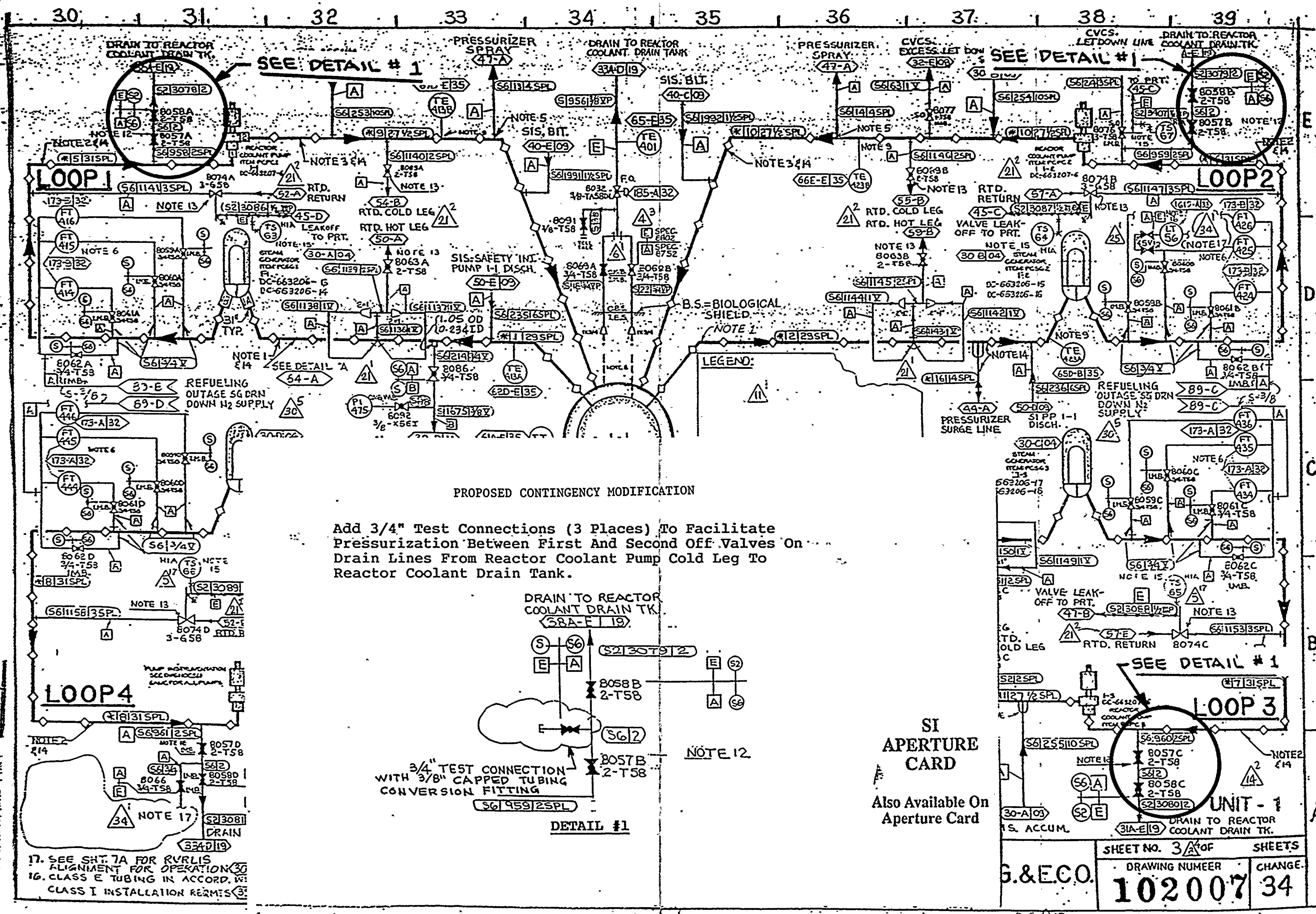
RR#10



PROPOSED CONTINGENCY MODIFICATION

Add 3/4" Test Connections (3 Places) To Facilitate Pressurization Between First And Second Off Valves On Drain Lines From Reactor Coolant Pump Cold Leg To Reactor Coolant Drain Tank.

SI APERTURE CARD

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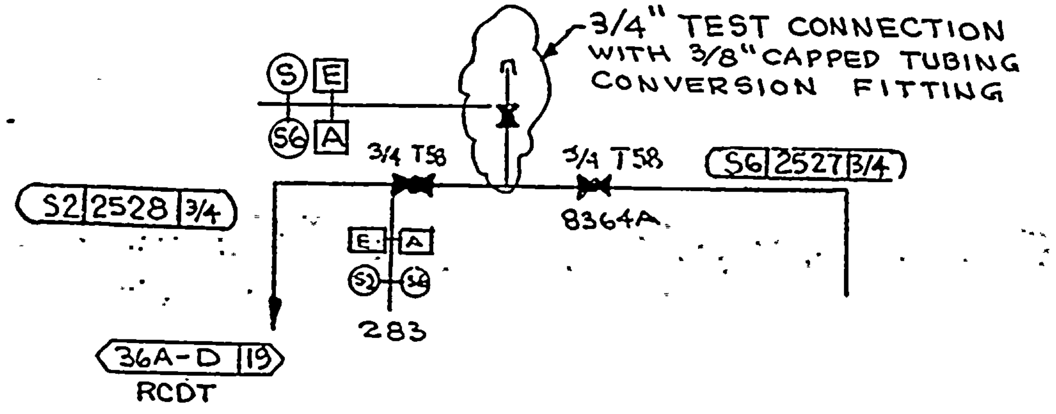
RM INDEXED REV 34

9203250163-02

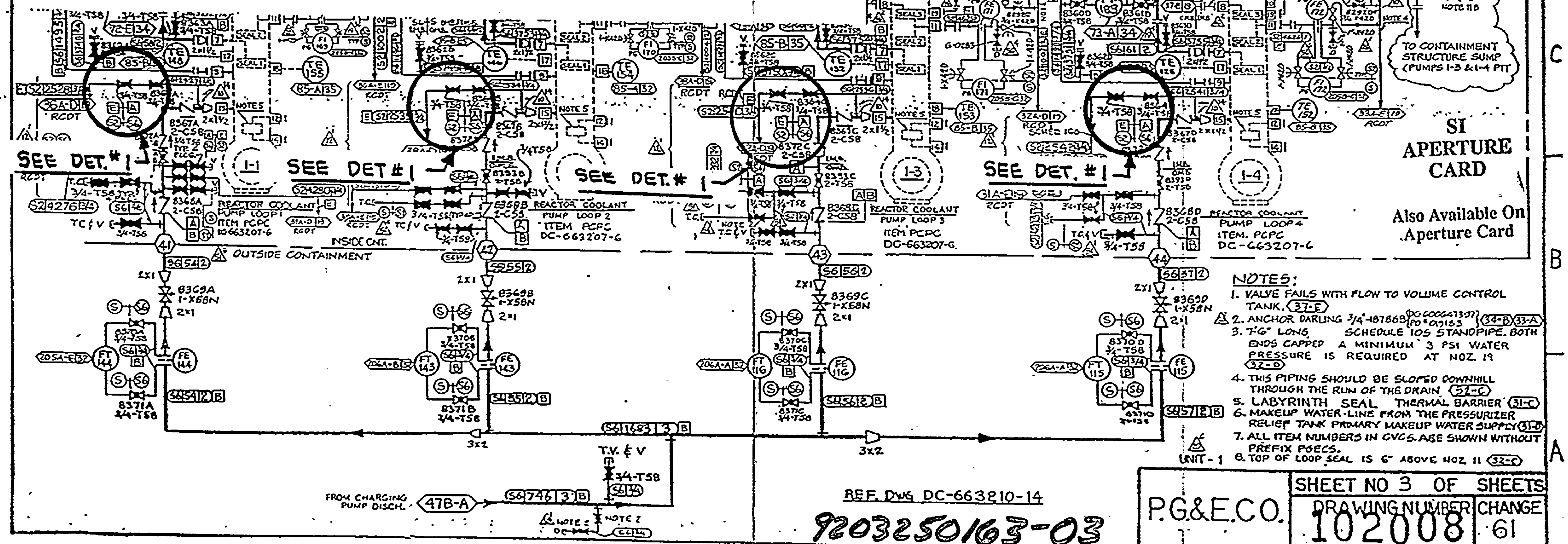
RR#10

PROPOSED CONTINGENCY MODIFICATION

Add 3/4" Test Connections (4 Places) To Facilitate Pressurization Between First And Second Off Valves On Drain Lines From Reactor Coolant Pump Seal Water Supply To Reactor Coolant Drain Tank.



DETAIL #1



SI APERTURE CARD

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REF. DWG DC-663210-14

9203250163-03

P.G.&E.CO.	SHEET NO 3 OF SHEETS	
	DRAWING NUMBER 102008	CHANGE 61

RM INDEXED REV. 61

35 M/M Neg 61

ENCLOSURE 3

REQUESTS FOR RELIEF FROM ASME SECTION XI REQUIREMENTS
INSERVICE INSPECTION (ISI) PROGRAM

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

SYSTEM PRESSURE TEST RELIEF REQUEST NO. 11
(includes attached drawings)

COMPONENT DESCRIPTION AND FUNCTION — BASIS FOR REQUESTING RELIEF AND ALTERNATE TESTING

11 56 (28 per unit) open-end tailpipes between first and second-off isolation valves.

No.	Location	Type/Function (see Note)
1	line 109	drain
2	line 961	drain (2" to 3/4")
3	RO-253	vent (1" Class 2)
4	RO-254	vent (1" Class 2)
5	V-8070	blind connection
6	RVRLIS	second-off connection
7	vent	head vent connection
8	line 14	vent
9	line 14	drain
10	line 14	drain (separate from above)
11	line 13	vent
12	line 13	drain
13	line 1195	vent
14	line 1469	vent
15	line 1495	vent
16	line 1496	vent
17	line 1497	vent
18	line 1498	vent
19	line 246	vent
20	line 253	vent
21	line 254	vent
22	line 255	vent
23	line 256	vent
24	line 235	vent
25	line 236	vent
26	line 237	vent
27	line 238	vent
28	line 109	vent

Note: All piping is Class 1 and 3/4", unless otherwise noted. The piping is not assigned a line number. The line numbers identify the major line that the piping originates from. The locations are identical for both units.

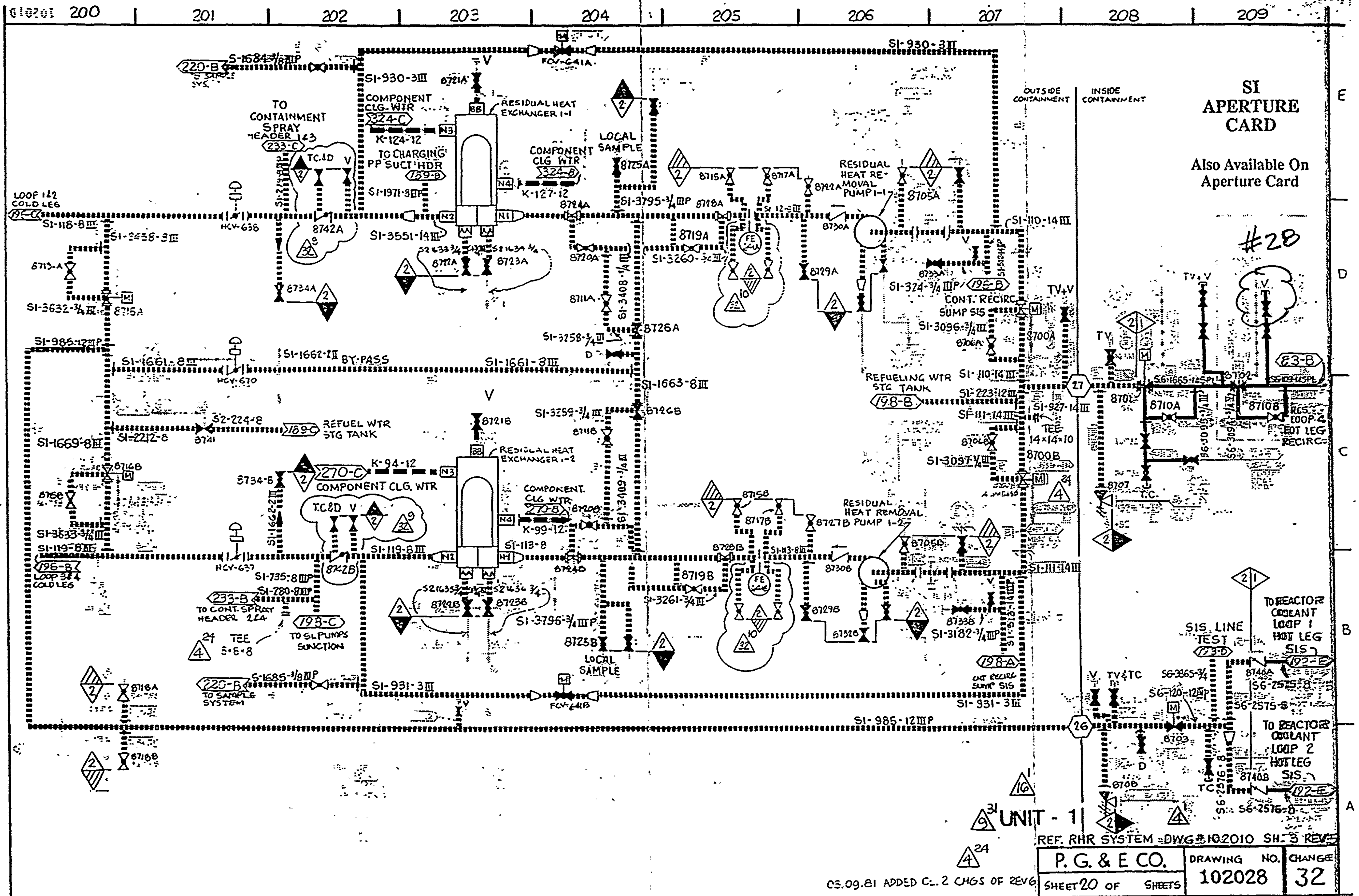
ASME Section XI (IWB-2500-1, IWB-5222, IWC-2500-1, IWC-5222) requires that a pressure test be performed on these piping systems once every 10 years. These line segments between the isolation valves are open-end tailpipes and serve as drain, vent, test, and fill lines. The line segments are short (less than 12" on the average) and small diameter (less than 2" NPS). The line segments are not normally pressurized (line pressure may exist due to first-off valve leakage and thermal effects only) and the isolation valves are not capable of automatic closure. Relief is requested from the 10-year pressure test requirement for the following reasons:

(a) Using system pressure to test these lines would require opening the first-off valve in Mode 3 (hot standby) to pressurize between the two isolation valves. However, pressure testing in this manner would result in a violation of Class 1 system design requirements for double isolation valve protection.

(b) Testing during Mode 6 (refueling) would require use of a hydro test pump at each location to pressurize the line. This method would result in unnecessary radiation exposure to plant personnel and increased risk of contaminated liquid spill, which would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety.

Alternatively, these line segments will be visually inspected during routine walkdowns, per the following procedures, to assure their continued integrity: Each refueling outage per STP R-8A, "RCS Operational Pressure Leak Test," and cold shutdowns of sufficient duration per STP R-8C, "Containment Walkdown for Evidence of Boric Acid Leakage."

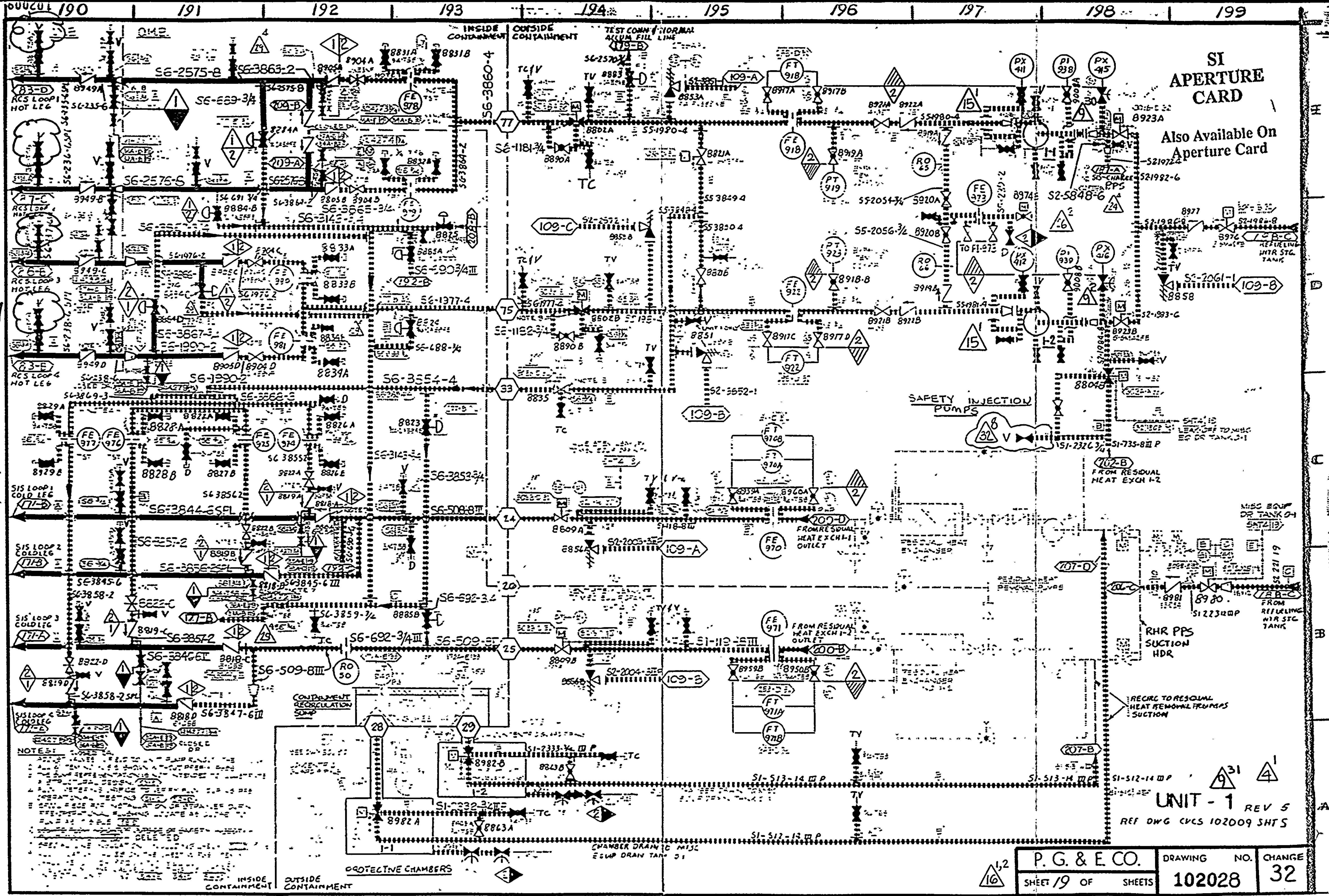
RR#11



35 M/M NEG

RM INDEXED REV.

9203250163-04



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RM INDEXED REV. ______

35 M/M NEG

9203250163-05

RR#11

RR#11

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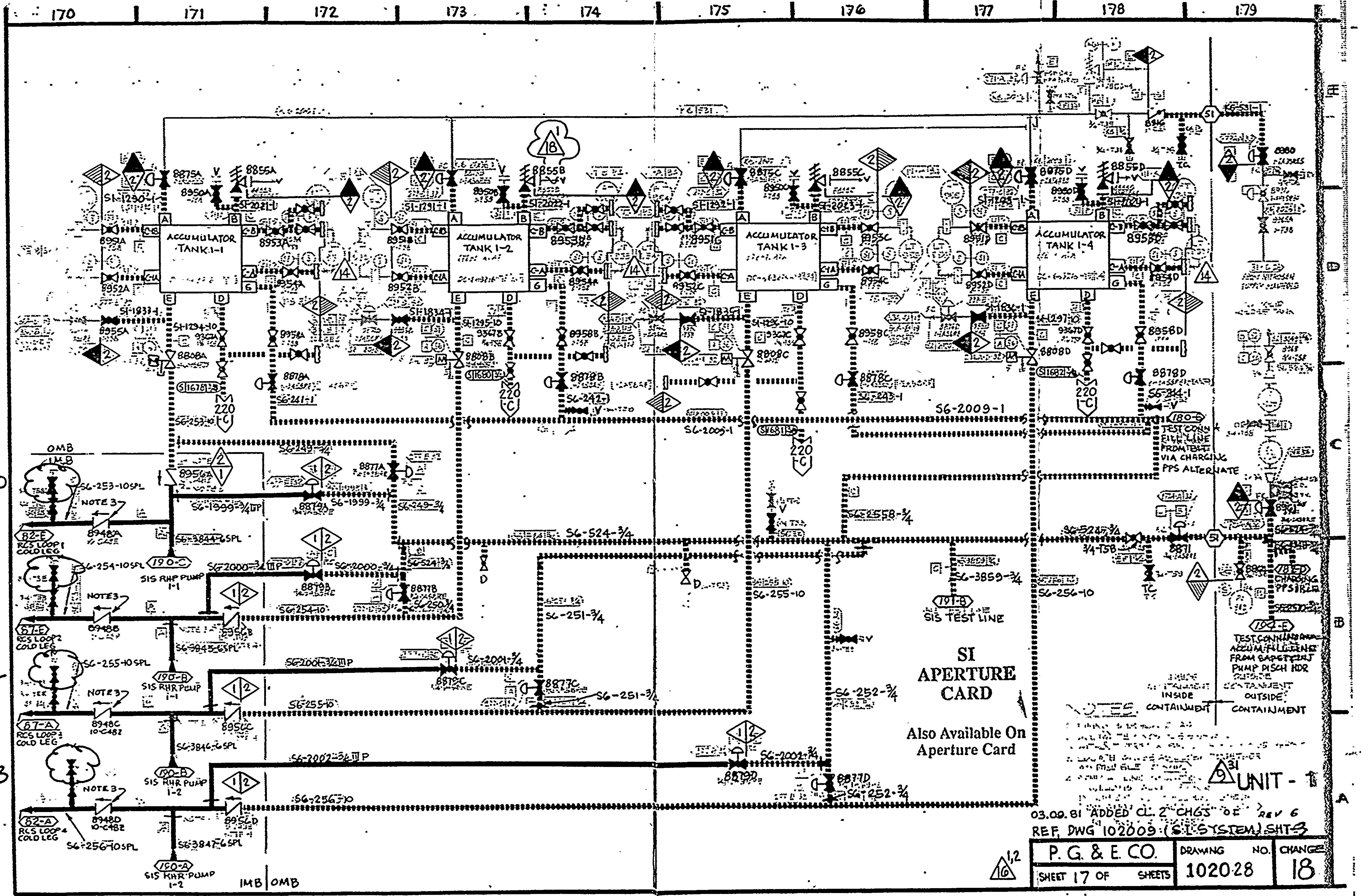
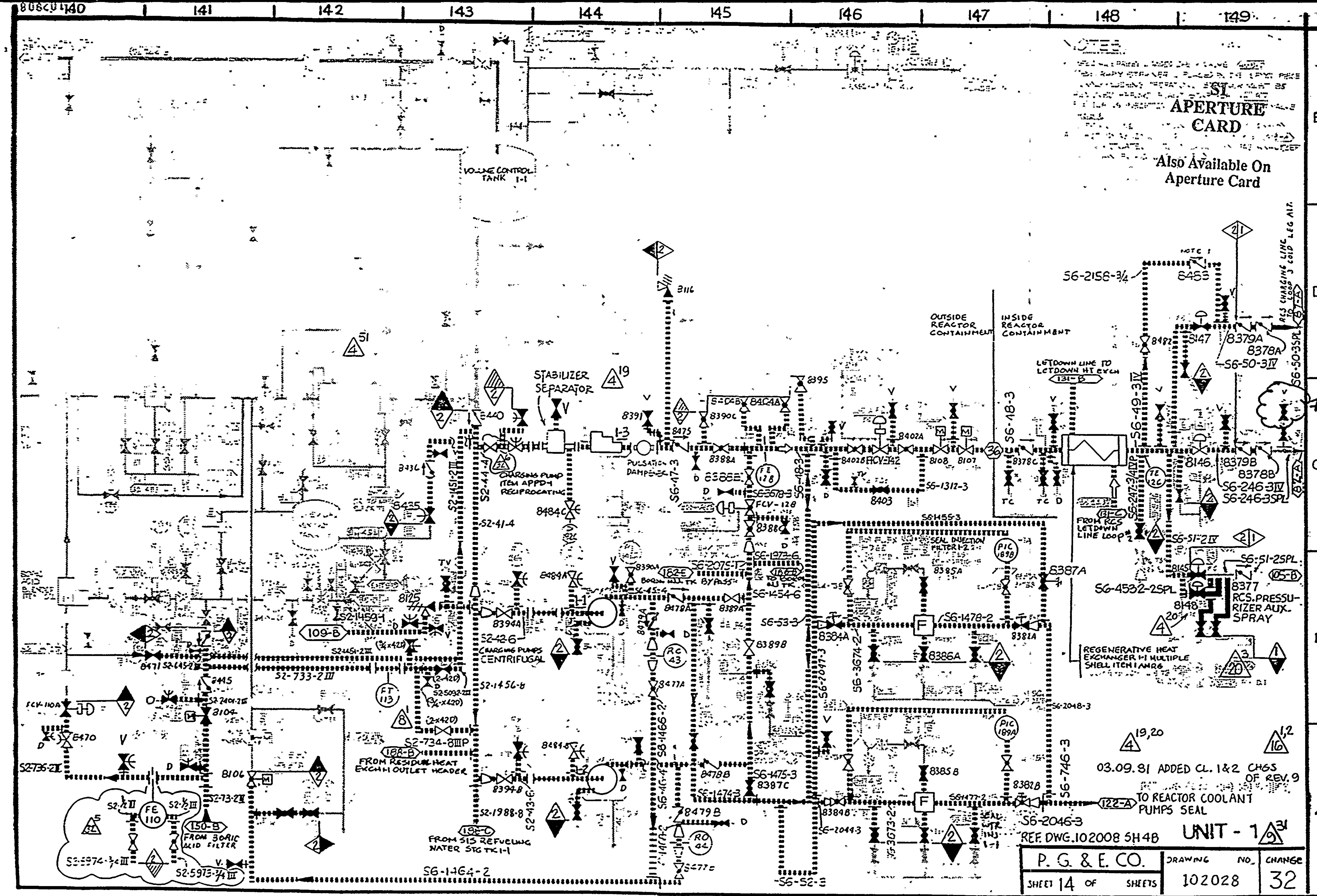


RM INDEXED REV. 18

35 M/M NEG 18

9203250163-06

RR#11

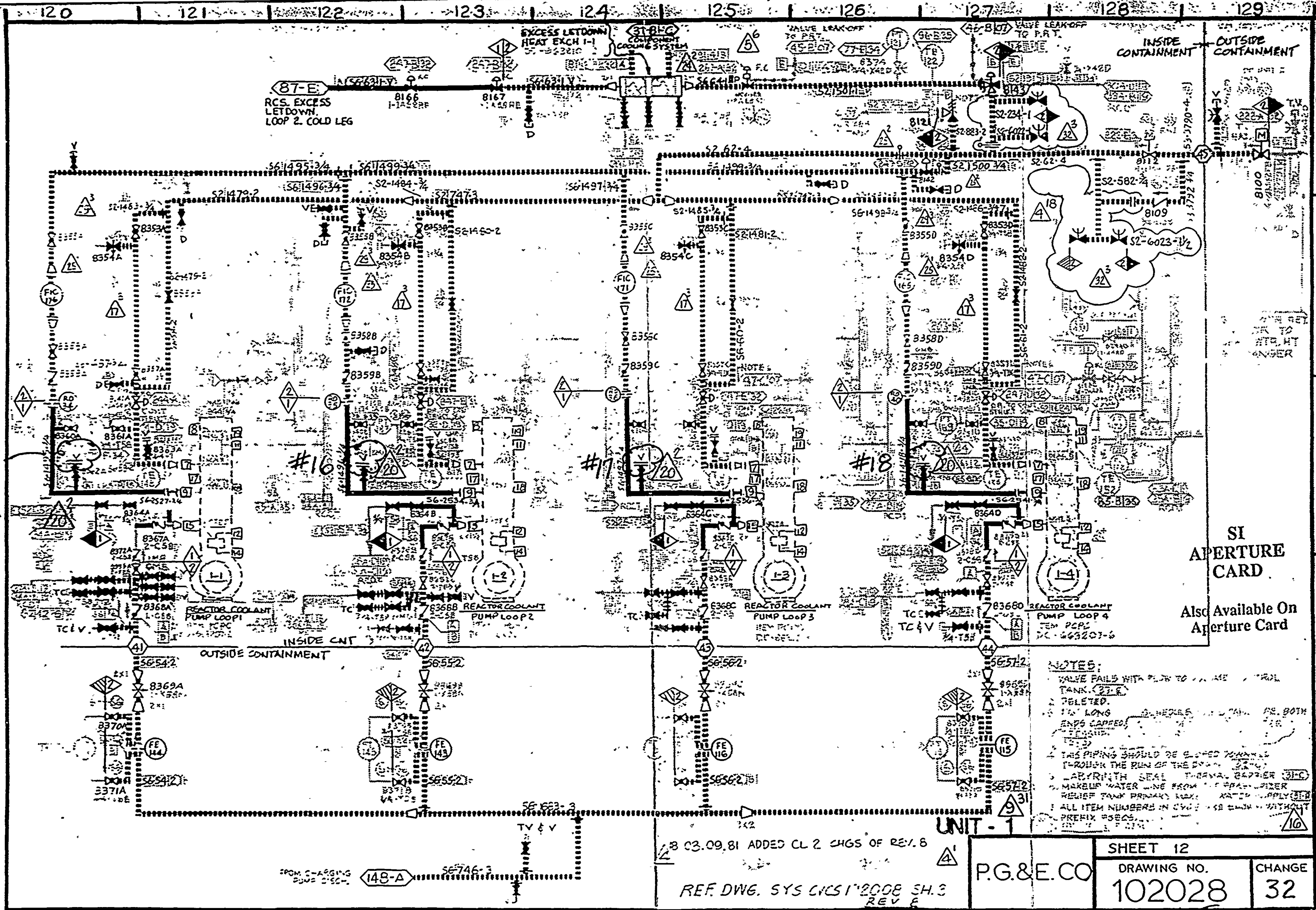


RM INDEXED REV. 32

35 M/M NEG

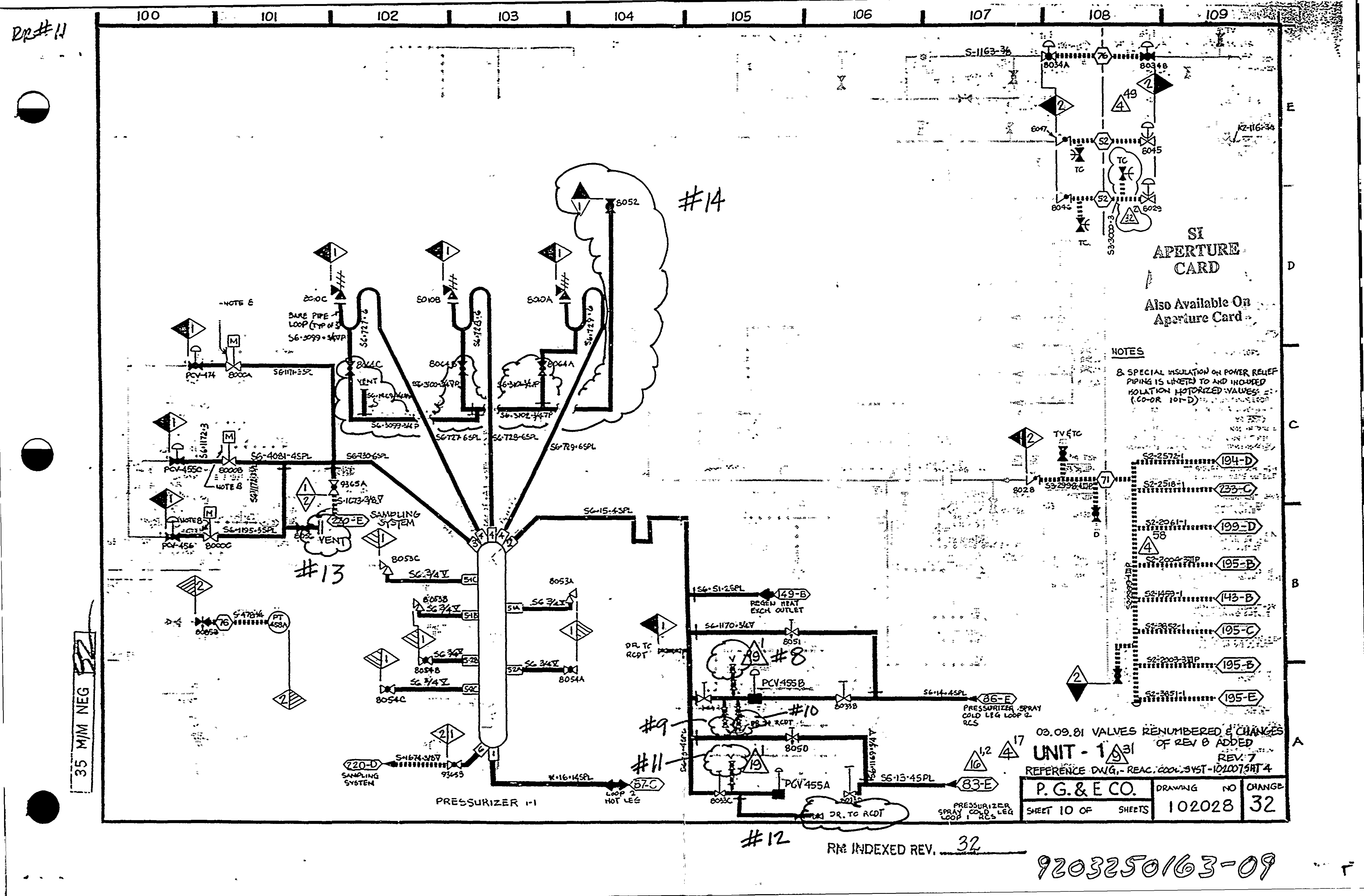
9203250163-07

RR#11



RM INDEXED REV. 32
35 M/M NEG.

9203250163-08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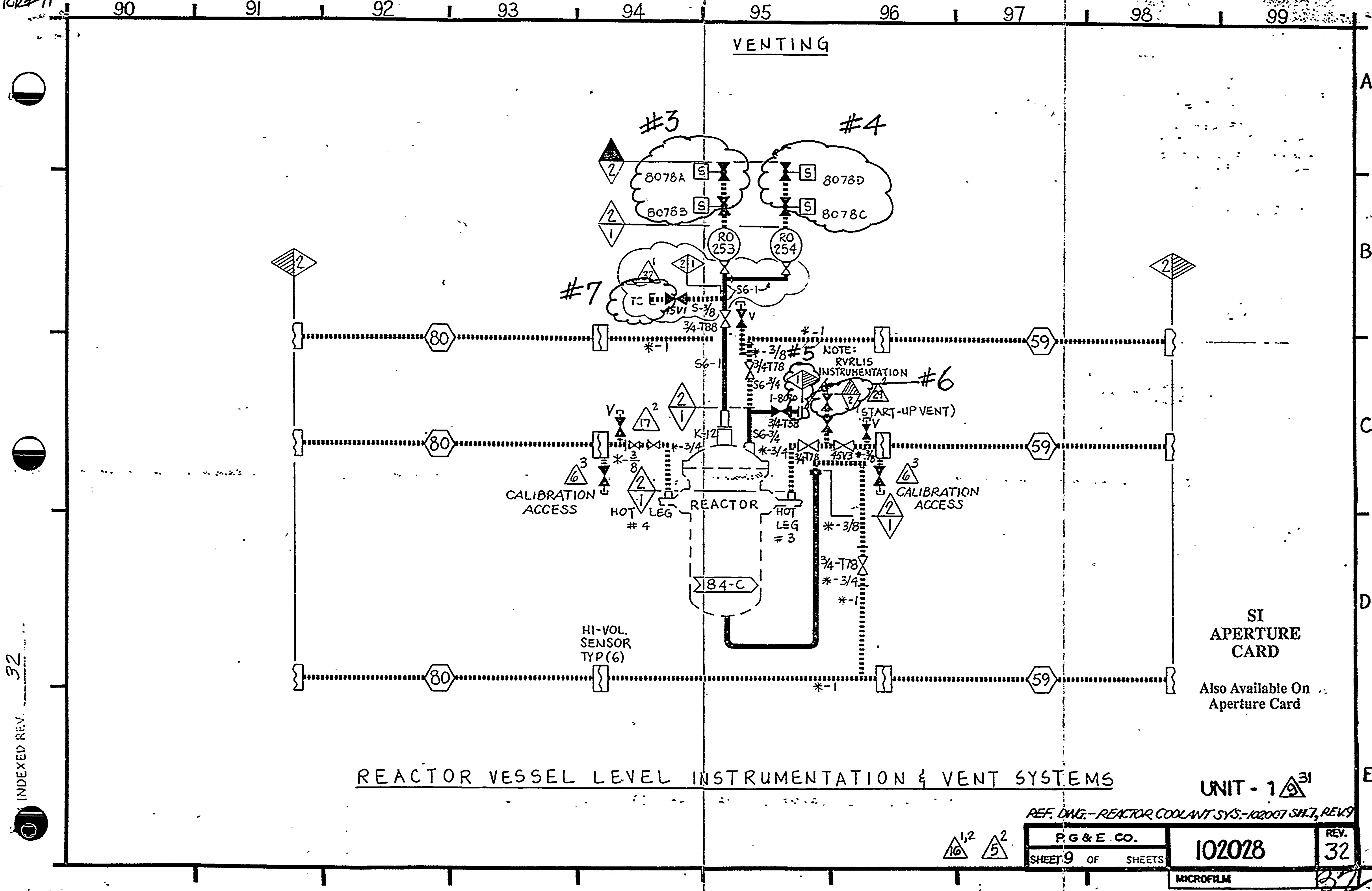
PRESSURIZER 1-1
SI APERTURE CARD
Also Available On Aperture Card
NOTES
& SPECIAL INSULATION ON POWER RELIEF PIPING IS LIMITED TO AND INCLUDED ISOLATION MOTORIZED VALVES. (CO-OR 101-D)
03.09.81 VALVES RENUMBERED & CHANGES OF REV B ADDED
UNIT - 1
REV. 7
REFERENCE DWG.- REAC. COOL. SYST-102007 SHT 4
P. G. & E CO.
SHEET 10 OF SHEETS
DRAWING NO 102028
CHANGE 32
RM INDEXED REV. 32
9203250163-09
35 M/M NEG

RR#11



9203250163-10

ENCLOSURE 4

REQUESTS FOR RELIEF FROM ASME SECTION XI REQUIREMENTS
INSERVICE INSPECTION (ISI) PROGRAM

DIABLO CANYON POWER PLANT, UNITS 1 AND 2

SYSTEM PRESSURE TEST RELIEF REQUEST NO. 12
(includes attached drawings)

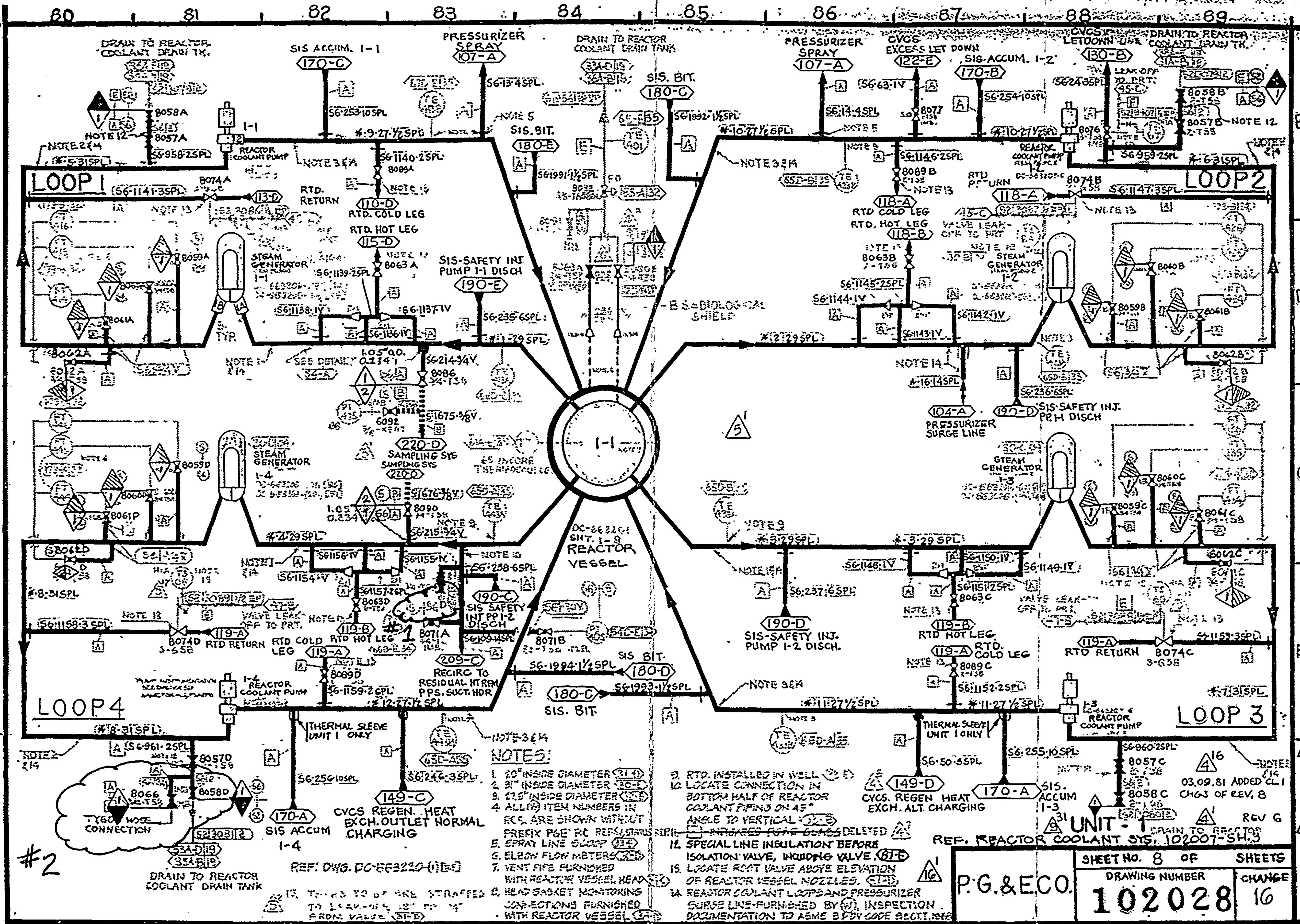
#	COMPONENT DESCRIPTION AND FUNCTION	BASIS FOR REQUESTING RELIEF AND ALTERNATE TESTING
12	16 (8 per unit) Code Class 1 line segments between first and second-off check valves.	ASME Section XI (IWB-5222) requires that a pressure test be performed on these piping systems once every 10 years. These line segments are located between check valves which function as RCS pressure boundary isolation valves, per Technical Specification Table 3.4-1. These check valves are boundary points, making it impossible to test the lines during Modes 4, 5, or 6 due to insufficient RCS pressure to keep the first-off valve shut against test pressure. In Mode 3, pressurization to the nominal test pressure would risk injection to the RCS. Relief is requested from the ten-year pressure test requirement, and the following alternative testing will assure continued integrity of these lines.

No.	Line	Size(in)	Description
1	253 3844 3855 1999	10" 6" 2" 3/4"	Accumulator, RHR, SI injection to loop 1 cold leg between 8948A and 8956A, 8819A, 8818A, 8879A.
2	254 3845 3856 2000	10" 6" 2" 3/4"	Accumulator, RHR, SI injection to loop 2 cold leg between 8948B and 8956B, 8819B, 8818B, 8879B.
3	255 3846 3857 2001	10" 6" 2" 3/4"	Accumulator, RHR, SI injection to loop 3 cold leg between 8948C and 8956C, 8819C, 8818C, 8879C.
4	256 3847 3858 2002	10" 6" 2" 3/4"	Accumulator, RHR, SI injection to loop 4 cold leg between 8948D and 8956D, 8819D, 8818D, 8879D.
5	2575 235 3863 689	8" 6" 2" 3/4"	SI, RHR to loop 1 hot leg between 8949A and 8884A, 8740A, 8905A.
6	2576 236 3864 691	8" 6" 2" 3/4"	SI, RHR to loop 2 hot leg between 8949B and 8884B, 8740B, 8905B.
7	237 1976 3866	6" 2" 3/4"	SI to loop 3 hot leg between 8949C, 8884C, and 8905C.
8	238 1990 3867	6" 2" 3/4"	SI to loop 4 hot leg between 8949D, 8884D, and 8905D.

The first four groups of line segments are normally pressurized at 625 psi (accumulator pressure) during normal operation and Mode 3. These segments will be inspected every refueling outage in Mode 3 at 625 psi (in lieu of the Code-required test pressure), in accordance with STP R-8A, "RCS Operational Pressure Leak Test." The lines are also subject to routine walkdown inspections at cold shutdowns of sufficient duration for STP R-8C, "Containment Walkdown for Evidence of Boric Acid Leakage."

The second group of four line segments are not normally pressurized (line pressure may exist due to first-off valve leakage and thermal effects only). These line segments will be pressurized to 1530 psi once every ten years in Mode 3 during performance of STP P-1B, "Routine Surveillance Test of Safety Injection Pumps." The lines will be visually inspected at that time. Additionally, these line segments will be visually inspected during routine walkdowns per the following procedures: Each refueling outage per STP R-8A, "RCS Operational Pressure Leak Test," and cold shutdowns of sufficient duration for STP R-8C, "Containment Walkdown for Evidence of Boric Acid Leakage."

RR#11



SI APERTURE CARD

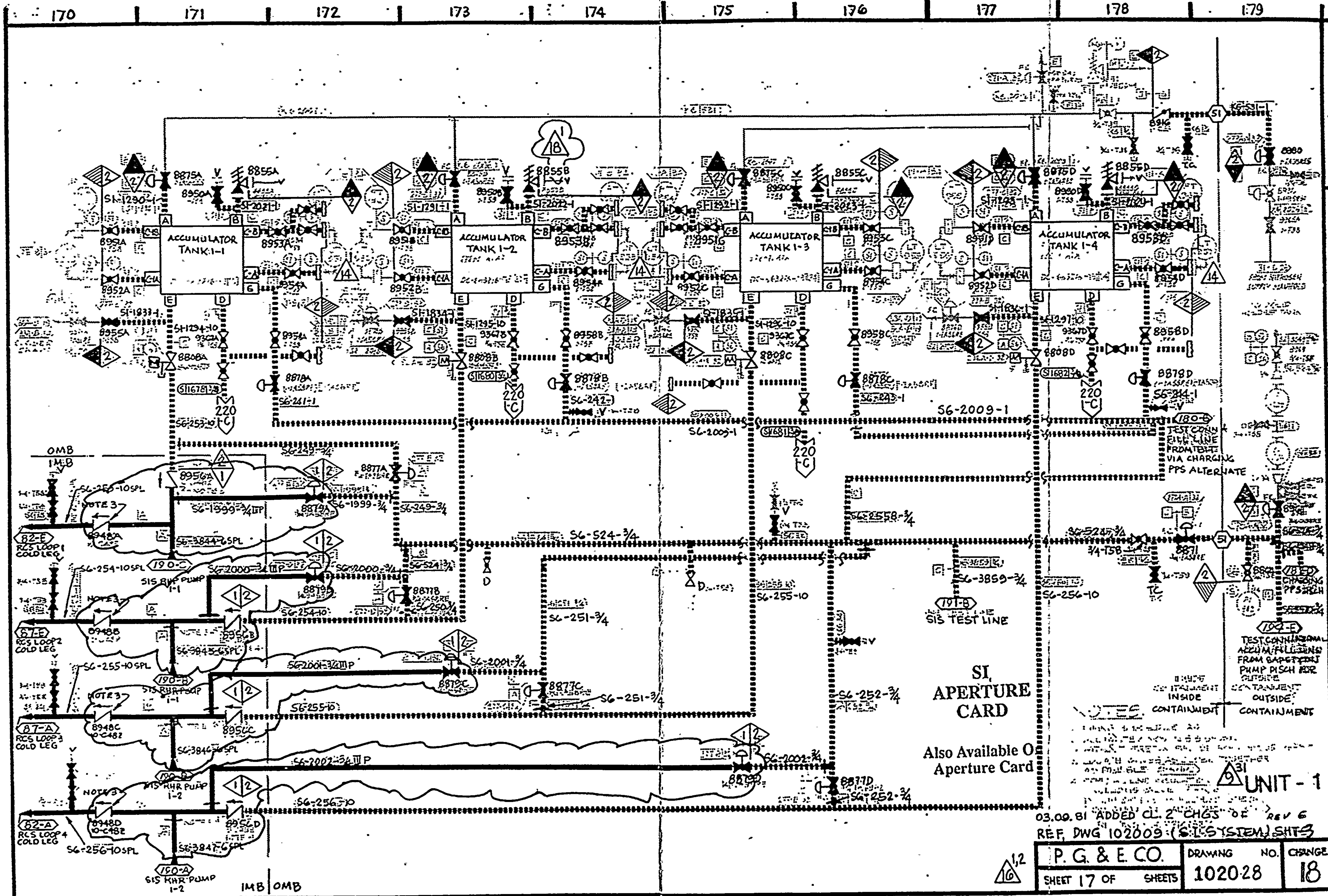
Also Available On Aperture Card

35 MM

RM INDEXED REV 16

9203250163-11

RR#12



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RM INDEXED REV. 18

35 M/M NEG 18

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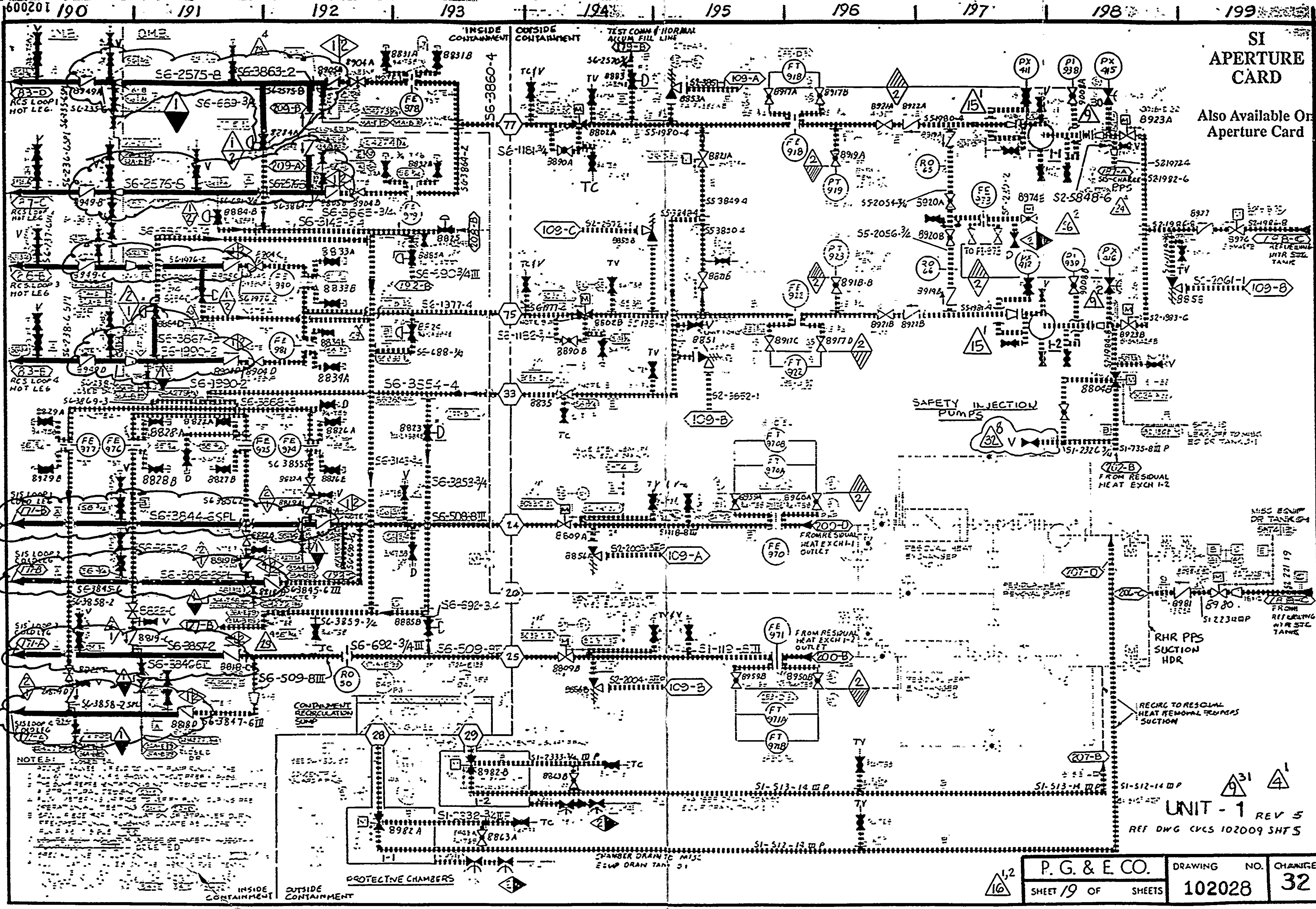

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ENCLOSURE

10 CFR 50.59 ANNUAL REPORT OF FACILITY CHANGES, PROCEDURE CHANGES, TESTS, AND EXPERIMENTS

MARCH 23, 1990 - MARCH 22, 1991

DIABLO CANYON POWER PLANT
UNITS 1 AND 2

DOCKET NOS. 50-275 AND 50-323

Pacific Gas and Electric Company

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SUMMARY OF 10 CFR 50.59 CHANGES FOR THE REPORT PERIOD

March 23, 1990 - March 22, 1991

A. Facility Changes

1. Install new Spent Resin Transfer Filters
DCP M-33470 Rev. 3 (Units 1 and 2)

Two, new Spent Resin Transfer Filters were installed to filter water used in the operation of the Spent Resin Storage Tanks. Associated piping modifications reroute Spent Resin Transfer System (SRTS) water to prevent contamination of the Liquid Radwaste System equipment during SRTS operation.

Safety Evaluation Summary

The modifications are wholly within the Auxiliary Building and do not change the method of processing or the quantity of water used in the Spent Resin Storage Tank System. The new filters are located in the same compartment, have a similar design and perform the same functions as the Waste Filters through which this water flow path previously passed. The reconfigured system has no interface with nuclear safety systems or design basis events evaluated in the FSAR Update. Therefore, no probabilities or consequences of previously analyzed accidents or malfunctions are affected.

The changes made also do not create or render credible any design basis event for the plant which were not previously evaluated.

Similarly, licensing bases' margins of safety are not affected by those changes since no change results in the quantity, type, frequency or total amount of radwaste material processed or discharged; operability of radwaste treatment systems is not compromised.

2. Install Area GE/GW Ducted Exhaust System to Plant Vent
DCP H-37182 Rev. 6 (Unit 1) and H-38182 Rev. 1 (Unit 2)

The GE/GW (Containment Penetration) Area just outside the Containment was previously open directly to the atmosphere which constituted a potential, unmonitored release path for radiation to the environs. This modification enclosed the Area and installed a ducted exhaust system with redundant exhaust fans to maintain this Area at a slight negative pressure and discharge the exhaust air flow to the plant vent, a monitored discharge path. The ductwork, fans, dampers, actuators and controls are Design Class II; the electrical and compressed air supplies are Design Class II.

Safety Evaluation Summary

The new HVAC system and its interfaces with existing plant structures and systems are incapable of causing any of the previously analyzed accidents evaluated in the FSAR Update. The structural interface with the plant vent is designed to prevent a malfunction of the plant vent's ability to exhaust safeguards HVAC flows.

The consequences of previously evaluated accidents and malfunctions are not increased: the required vent paths and areas for GE/GW Area HELB events are maintained through the use of blowout covers in the Area enclosure; the existing radiation monitoring system will not be affected by this addition to the monitored flow stream; there is no change to the potential amounts or types of radiation released by the plant so that offsite doses remain unchanged. Also, the small effect of this modification on the fire hazards analyses has been reviewed and determined to be acceptable.

The new HVAC system's construction, location, interface with existing plant structures and systems and failure modes cannot cause any accidents to be created. No new type of malfunction of important to safety (ITS) equipment is created either, since: environmental qualification of components in Area GE/GW is not compromised and seismically induced systems interactions have been precluded by evaluation and walkdown.

The bases for Technical Specifications for the following structures, systems and components were reviewed: Instrumentation (radiation/accident monitoring and radioactive gaseous effluent monitoring), containment structure, HVAC systems (Aux. Bldg. and Fuel Handling Bldg.), Containment Ventilation Isolation, and radioactive gas effluents. In each respective case, the appropriate requirements for operability, setpoints/sensitivities/detection levels, structural integrity, performance, flow capacities, discharge quantities, etc., have been met without a decrease in any margin of safety.

3. Replace the Emergency Boration Flow Instrumentation
DCP J-37477 Rev. 1 (Unit 1) and J-38477 Rev. 1 (Unit 2)

Per DCPP commitments to Regulatory Guide 1.97, environmentally qualified (EQ'd) emergency boration flow instrumentation was installed to replace unqualified instrumentation. The new, ultrasonic flow transducers are mounted externally on a flanged pipe spool which replaced the previous, magnetic-type, instrument installation. Necessary signal processing and isolation devices are provided. The new instrumentation is powered from an uninterruptible, vital power supply and drives the existing, panel-mounted flow indicators.

Safety Evaluation Summary

This instrument loop's power sources and protection, signal isolators, and lack of interconnections with other loops preclude it from causing any previously analyzed or different accidents or malfunctions. The failure effects have not changed from the former configuration. The emergency boration piping seismic and pressure boundary integrity have been maintained; the emergency boration flow path has been maintained with the reinstallation of heat tracing on the piping; no systems interactions can occur which could degrade or challenge plant safety systems. The boration flow path required by Technical Specifications (TS) has been maintained without a safety margin reduction. The TS-required remote shutdown monitoring capability is maintained by the now-EQ'd emergency boration flow indication.

4. <u>Install a Second Monorail and Hoist on the Reactor Cavity Manipulator Crane</u>
DCP M-39516 Rev. 4 (Unit 1) and M-40516 Rev. 1 (Unit 2)

A second monorail and auxiliary hoist were added to the Reactor Cavity Manipulator Crane, along with a full length walkway. These two auxiliary hoists are used in lifting operations associated with refueling (moving control rods) and retrieval of irradiated capsules from inside the reactor vessel. These functions are not safety related. The controls and power supply for the hoists are Design Class II. Structural, including Hosgri-seismic, integrity of the manipulator crane has been verified for these additions.

Safety Evaluation Summary

The auxiliary hoists are not used for movement of fuel. Previously-evaluated fuel handling accidents are not more probable because: the normal fuel handling equipment and its function have not been changed; the structural integrity of the crane assures it cannot fail and damage fuel in the core; and, the crane remains seismically qualified while handling fuel. The small change in combustible loading does not adversely affect the fire hazards analysis. Malfunctions of ITS equipment are not more likely since: electrical interfaces of the new hoist do not connect to any ITS circuits, and crane structural integrity has been maintained.

These modifications cannot adversely affect any actions, assumptions (including containment aluminum inventory), mitigation or barriers associated with accidents or malfunctions; therefore, there can be no increase in radiological consequences.

These electrical, non-Class IE and passive structural additions are designed to ensure that no new or different design basis events are created or made credible. Administrative controls preclude use of either hoist from lifting loads in excess of 1972 lbs. The new auxiliary hoist and its monorail have a rated capacity equal to the previously-existing equipment. This assures that the minimum capacity requirements of TS 3.9.6 are met.

5. <u>Add a Diesel Fuel Oil Recirculation System and Emergency Fuel Oil Transfer System</u>
DCP M-39858 Rev. 5 (Units 1 and 2)

Based on the results of the DCPP PRA study and a commitment by PG&E to the NRC, the Emergency Diesel Generator (EDG) Fuel Oil System was revised: modifications were made to provide for continuous recirculation of the fuel oil supply when the EDGs are operating which eliminates the fuel oil transfer pump start-stop cycling when refilling the day tanks. Compensatory features were provided to mitigate the slightly increased possibility of fuel oil spillage with the recirculation system. Also added during this work was provision for an independent, portable, engine-driven, emergency fuel oil transfer system to further improve fuel oil system reliability.

Safety Evaluation Summary

The EDG fuel oil transfer system is not the source of any licensing basis events. It supports the accident mitigation function of the EDG System. Based on PRA studies, the recirculation system will decrease the probability of malfunction of the EDG System.

Other potential impacts (e.g., vital bus loading, fire hazard and transfer pump vault ambient temperature changes) will not increase malfunction probabilities of ITS equipment.

The fuel oil system remains qualified for all design and regulatory requirements (e.g., seismic, materials, pressure, redundancy, separation, physical security, fuel oil inventory, etc.)

Since the affected system contains no radioactive materials, is not a fission product barrier and continues to support all assumptions and actions associated with licensing basis event mitigation, no previously evaluated radiological consequences are increased.

The changes, including the added, portable, emergency fuel oil transfer capability, have not created any new or different accidents or malfunctions. Increased fire hazards, oil spill potential and fuel oil system mechanical failures have been evaluated and/or compensated for in the design.

None of the operability, redundancy or independence requirements for vital electrical power systems in the TS have been compromised by this design change.

6. <u>Modify the Fuel Transfer System</u>
DCP M-40441 Rev. 2 (Unit 2)

This change replaced the fuel transfer system air motor and associated chains, sprockets, air lines, underwater limit switches and control panel with an electric motor and winch with control panel, including abovewater, programmable limit switches. The electrical motor and winch (powered from non-vital power supplies), as with the former air motor, is the drive mechanism used to move the transfer cart from the containment, through the transfer tube to the fuel handling building and return during refueling outages. The transfer cart carries one spent or new fuel assembly.

<u>Safety Evaluation Summary</u>

The probability of a fuel handling accident of the limiting-fault-type already analyzed is not increased because: 1) though it carries the fuel, the transfer system is not likely to be a source of damage to a fuel assembly; 2) the modifications do not alter how the fuel is transferred but only change the drive mechanism and transfer cart position control scheme; 3) the new design components and materials are compatible with their environment so that long term integrity of the transfer system is assured and a threat to spent fuel integrity does not result; 4) the new design will improve the reliability of the fuel transfer operation using a simpler and more accessible and maintainable design.

This Design Class III system does not directly interface with any ITS equipment in such a way so as to increase the potential for malfunction of such equipment or the consequence of such a malfunction. It is located and mounted so that it does not pose a hazard to SISI targets. Containment paint and aluminum inventories have been updated and evaluated to assure that no unacceptable conditions or challenges to ITS equipment result. Impact to fire hazards/10 CFR 50, Appendix R, safe shutdown analyses have been documented as acceptable.

The complete failure of a single fuel assembly was previously evaluated. The consequences from this event bound all possible doses resulting from fuel damage due to the transfer system. Consequences from other analyzed accidents (e.g., LOCA, earthquake) have also not increased since the modifications have not adversely affected assumptions or actions related to accident mitigation.

Various failure modes of the new equipment were evaluated to assure that new, design basis events could not be created.

Fuel transfer operations are not directly discussed in the TS.

7. Remove the CCW Heat Exchanger Tubeside Air Removal System
DCP M-41068 Rev. 1 (Unit 1) and M-42068 Rev. 1 (Unit 2)

The Design Class II waterbox air removal system for the CCW Heat Exchangers was isolated from the heat exchangers and removed. Calculations and operating experience revealed that purging of air from the waterboxes occurs naturally during normal operation of the Auxiliary Saltwater (ASW) System. Existing Design Class I manual vent valves were closed to reestablish the pressure boundary integrity of the ASW System. This change eliminates a potential leak path of ASW water. The vital-powered solenoid valves deleted with the system's removal have been removed from the 10 CFR 50, Appendix R safe shutdown components list.

Safety Evaluation Summary

Isolation and removal of the air removal system cannot result in the occurrence of any previously analyzed accidents: the ASW and CCW Systems are accident mitigation systems and the air removal system was provided to enhance their operability. Removal of this system will not degrade the performance or challenge the function of these systems since air will be purged from the ASW side of the heat exchangers during normal ASW System operation. Heat exchanger seismic qualification is maintained. Removal of a small electrical load from the vital electrical buses will not compromise vital power integrity. The air removal system was previously a potential source of leakage from the ASW System which is now eliminated. The service air supply (now capped) for the air removal jet exhauster has not been compromised.

Since: 1) ASW or CCW System integrity function and performance are not adversely affected, and 2) 10 CFR 50, Appendix R safe shutdown capability is not compromised, no new accidents or malfunctions are created.
The TS bases require operability, flow capacity and redundancy of water systems providing for continued operation of safety related equipment. This modification maintains these requirements without a reduction in the safety margin.

8. Replace the Plant Process Computer
DCPs J-41533 Rev. 2 and J-41534 Rev. 1 (Unit 1) and J-42534 Rev. 0 (Unit 2)

The Design Class II Plant Process Computer (PPC) was replaced in each Unit. The PPC is located in the Control Room enclosure adjacent to the Main Control Room. Other, related, man-machine interface upgrades (e.g., CRTs, printers, furniture in the Main Control Room)

were provided as part of this work. In conjunction with this work in Unit 1, a temporary power supply was provided for the Movable Incore Detector System (MIDS) to assure its continued operability during the replacement work.

Safety Evaluation Summary

The PPC is not associated with the cause of any accidents. Its function is to provide process parameters information. It is not required for accident or malfunction mitigation but, as a highly reliable source of plant systems' status, may be used to support safe plant operation and event/condition's diagnosis. The capabilities for information display, printing, etc., were maintained (or enhanced) with the modification.

The PPC is suitably isolated from all Class IE sources so as to not cause degradation of malfunction of ITS equipment (e.g., SSPS). The PPC is powered from a non-vital, uninterruptible power supply. The new, functionally-equivalent replacement PPC system has been evaluated in the following areas to assure no compromises to plant safety were introduced: combustibles loading, SISI, building seismic structural integrity, control console seismic integrity, Control Room HVAC and habitability.

During removal of the old and installation of the new PPC, and the final configuration of the PPC and peripherals, the operability and licensing-basis acceptance of the following TS-required systems was maintained so as to not result in safety margin decreases: Digital Rod Position Indication (TS 3.1.3), Movable Incore Detector System (TS 3.3.3.2) and other instrumentation systems, Control Room Ventilation System (TS 3.7.5), and Fire Barrier Penetrations (TS 3.7.10).

9. Replace the Containment (ECCS) Recirculation Sump Level Instrumentation
DCP J-41715 Rev. 4 (Unit 1)

The narrow range Containment Recirculation Sump level instruments were replaced and the Control Room level indication was rescaled. Thermal dispersion level instruments are now used in place of the former bellows/capillary tube devices. Overall level channel accuracy was reduced from ± 3.5% to ± 6.5%, still well below the ± 20% accuracy required by Regulatory Guide 1.97.

Safety Evaluation Summary

The instrumentation affected by this design change is used for post-accident monitoring; it is not associated with the cause of any FSAR Update accidents. The output of these instruments has no control function; it provides only level indication in the Control Room. These instruments are powered from the same vital power sources as the previous devices. The entire installation is qualified to preclude seismically induced interactions with ITS equipment. For these reasons, the malfunction of this or other equipment is not more likely.

The consequences of accidents or malfunctions are not increased because: 1) vital bus loads remain acceptable, 2) containment electrical penetrations' integrity is maintained; 3) the new instrumentation is environmentally (including seismically) qualified; 4) Containment Recirculation Sump integrity and performance are maintained; 5) the slight

change in fire hazard is acceptable; 6) required redundancy and separation are maintained; and 7) Control Room operator action based on level indication provided by this instrumentation is not compromised.

These instrument loops are incapable in themselves of producing accidents of the design basis type evaluated in the FSAR Update. The new devices, however, employ a different level sensing technology than the former ones. Though the new devices have a much slower response to decreases in liquid level, a different type of RHR pump malfunction (i.e., undetected, rapid loss of suction supply) cannot credibly occur: sump water level changes can occur only slowly and post-accident sump level is not expected to decrease in any event.

The ability to monitor accident conditions via narrow range sump level indication is required by TS. Response characteristics and accuracy are consistent with accident conditions and regulatory requirements. The configuration of the new instrumentation system meets, without reduction in safety margin, the TS requirements for maintenance of containment integrity and ECCS and vital, onsite power system operability.

10. <u>Install an Air Conditioning System for the Plant Process Computer Room</u>
DCP J-42528 Rev. 1 (Unit 2)

The plant process computer (PPC) is located within the Control Room enclosure, but in a room separated from the Main Control Room. The new PPC introduces additional heat load in the computer room; a supplemental, Design Class II air conditioning system is provided in the room to remove the PPC heat load in Unit 2 (as was previously reported for Unit 1 in DCL-91-057). The evaporator (cooling coil) units are in the computer room enclosure. Freon tubing, cooling coil condensate drain piping and electrical conduits penetrate the enclosure boundary for the evaporator units. Non-vital power supplies this air conditioning system.

<u>Safety Evaluation Summary</u>

Except for a potential Control Room habitability issue discussed below, these modifications cannot cause any previously evaluated events. This modification introduces freon, a toxic gas, within the Control Room envelope. Calculations have shown that a major failure of this refrigeration system would have to occur for the freon threshold limit value of 1000 ppm to be reached inside the Control Room. SISI seismic supports and evaluations of the freon pressure boundary inside the Control Room have assured the continued integrity of this system. Control Room habitability is not compromised.

The PPC (for which the air conditioners were installed) is not ITS; failure of the cooling system function will not compromise any ITS equipment. The Control Building structure has been evaluated for the additional loading, including seismic, of the new air conditioning equipment. Required fire barrier integrity at the new penetrations has been documented in Fire Hazards Appendix R Evaluation (FHARE) 89. All power for the system is supplied from non-vital sources.

No dose consequences have been increased because: 1) the Control Room envelope is maintained with sealed penetrations and SISI-supported condensate drain piping which

prevents penetration damage, and 2) radiation shielding at the new penetrations has been maintained for Control Room personnel.

For the reason cited above, the new system is also not capable of producing any new accidents or malfunctions of ITS equipment.

The penetrations provided by this change meet the TS requirements for such penetrations to assure the performance of required fire barriers. Also, this HVAC change cannot affect the Control Room Ventilation System operability requirements related to temperature, habitability, radiation exposure or humidity contained in the TS.

11. Upgrade the Plant Compressed Air System
DCP M-43066 Rev. 0 (Units 1 and 2)

New, Design Class II air dryer and prefilters, designed to appropriate codes and standards, have been installed in the compressed air system. These replace previous equipment and have more reliable controls and better filtration characteristics.

Safety Evaluation Summary

The compressed air system is not associated with the cause of any FSAR Update accidents. The components of the system affected by this DCP interface with the Service Cooling Water and non-vital power supplies. This replacement of equipment does not increase the potential to malfunction of these interface systems. The new equipment will also not interact with safe shutdown equipment due to seismically induced deflections or failures. Compressed air is also supplied to various ITS components (instruments, air operated valves, etc.). The new equipment continues to provide oil-free, low dew point, highly-filtered air to assure that malfunction of these components does not result from the air supply.

The portion of the compressed air system affected by these changes is not required for mitigation of accidents or malfunctions; calculated radiological consequences are not, then, dependent on performance of this Design Class II equipment.

For the reason cited above, this replacement equipment is, similarly, not able to create any different type of accident or malfunction than previously analyzed.

Compressed air is supplied to many systems and components governed by DCPP TS. The Design Class II air system is relied upon to not degrade or otherwise compromise the margins required for the operability of these systems and components. The replacement equipment even more reliably assures the quality of compressed air to meet these operability demands.

12. Upgrade the HVAC System in the Solid Radwaste Storage Facility
DCP H-43273 Rev. 0 (Units 1 and 2)

The use of Bay 6 of the Solid Radwaste Storage Facility (SRSF) was changed: formerly designated for boxed radwaste storage, it has been equipped with facilities for manual sorting of clean and contaminated wastes. The HVAC system for the SRSF was modified

to include fume hoods for personnel protection during sorting activities. A rollup door was removed to provide an emergency egress route from the area.

Safety Evaluation Summary

The design and use of the SRSF does not relate to any of the analyzed accidents in the FSAR Update. It is remote from the main power block structures and all its interfaces with the power block (i.e., electrical power, process monitoring, service air, water, drainage, etc.) are Design Class II. It is not a 10 CFR 50, Appendix R area and contains no SISI targets. No credible events in the SRSF resulting from these modifications can result in the malfunction of any ITS equipment.

Design features in the reconfigured Bay 6 maintain radiological protection for onsite personnel working in the area; offsite doses from potential air or liquid release paths (all of which are monitored) are enveloped by prior analyses.

The modifications implemented to change the use of Bay 6 do not compromise SRSF integrity and do not increase releases of radioactive effluents. The margins required by TS 3.11.4 for the protection of the health and safety of the public are not reduced.

13. Reconfigure one of the available Water Supplies for the Long Term Cooling Water (LTCW) System
DCP M-43348 Rev. 1 (Units 1 and 2)

The Long Term Cooling Water (LTCW) System can use the Raw Water Reservoir as a backup source of water for the Auxiliary Feedwater (AFW) Pumps. A portable, diesel engine-driven pump was used to establish a syphon in a transfer hose to provide reservoir water to the AFW pumps. This pump will now be used to continuously pump the water through the hose. The pump and hose are normally stored. They are moved into place and manually connected and operated, if needed, to provide an LTCW supply.

Safety Evaluation Summary

The LTCW System cannot be the source, cause or contributor to any of the previously-evaluated licensing basis events or any other events of a similar type. It provides a long term backup (near "last resort", fresh water) source of water for the accident-mitigating AFW System. The pump, engine, fuel oil supply inventory and new hoses are suitable for their potential use in continuous service. The new configuration (with the portable and AFW pumps operating in series) does not increase the potential for malfunction of the AFW piping or pumps or create a new type of malfunction. Since no LTCW functions are compromised, no calculated radiological consequences are changed. Although the LTCW System is not required by TS, the AFW system is required. No AFW System safety margins regarding operability, capacity, redundancy or diversity are compromised.

14. Revise H_2 Pressure Regulator Setpoint for VCT to allow for an Operating Band
DCP J-43703 Rev. 0 (Unit 1) and J-44703 Rev. 0 (Unit 2)

The Volume Control Tank (VCT), which provides a source of reactor makeup water for the charging pumps, also aides in chemistry control of the reactor coolant. Specifically, H_2 gas is used to pressurize the VCT in order that it may dissolve in the water to scavenge O_2 in the RCS. Regulation of the H_2 concentration in the reactor coolant is obtained from the H_2 pressure in the VCT.

This DCP permits the formerly-fixed VCT H_2 pressure of 23 psig to be varied from 15 to 35 psig as determined by chemistry and operating needs of the plant.

Safety Evaluation Summary

The VCT is not associated with the initiation of any events or accidents considered in the FSAR Update. Though not required for accident mitigation, the VCT may be used during some events, if available. Permitting the VCT to operate in a range of 15 to 35 psig will still provide: a surge capacity for RCS expansion, permit control of RCS water inventory and chemistry, maintain an RCP seal water injection supply and allow the VCT to act as head tank to provide adequate NPSH to the charging pumps. The high pressure ECCS functions cannot be compromised, thereby assuring that previously evaluated accident consequences are not increased. The VCT will continue to operate within its own design limitations and no credible failure of the VCT can create a new design/licensing basis event.

Since the VCT processes fluid coming from and going to the RCS, it has a potential impact on the function of equipment important to safety. The following areas have been reviewed to assure that no adverse impact results from providing this flexibility in operating pressure range: RCS volume surge capacity, control of H_2 concentration in the RCS, adequate NPSH for the charging pumps, H_2 gas coming out of solution, maintenance of RCP seal back pressure and effects of VCT back pressure on relief valve settings/performance. None of these requirements will be degraded below design basis; no safety functions will be challenged.

Though not specifically identified in the TS, the VCT can influence the required operability of the high head ECCS pumps which are required by TS. The potential for gas binding these pumps (due to rapid VCT depressurization, for example) or damage resulting from operation with inadequate NPSH have been evaluated to assure that margins of safety have not been compromised.

15. Make various Modifications to the Containment (ECCS) Recirculation Sump
DCP N-43919 Rev. 0 (Unit 1)

To meet a commitment to the NRC, numerous enhancements were made to the Containment (ECCS) Recirculation Sump: some materials were replaced with corrosion-resistant substitutes; an additional internal flow path was created; protective grating was installed over screened areas; provision was installed for removing standing, accumulated water from inside the sump; and features unnecessary to sump function were removed.

Safety Evaluation Summary

The sump is a passive, structural component required for the longer-term mitigation of LOCA events; it is not associated with the cause of LOCAs or other analyzed design basis events. It is required so that an adequate supply of strained water with adequate NPSH and no vortexing is provided for ECCS operation in the recirculation mode. The modifications were designed, fabricated and installed to assure this vital function without degradation or challenge to ITS systems. The modifications do not compromise any assumptions, features or functions associated with accidents or malfunctions so that previously-calculated radiological consequences are not increased.

Missiles, SISI, coatings, material compatibility, separation, seismic qualification, containment integrity and other such issues have been addressed to assure that no new accidents or malfunctions are made credible.

TS operability requirements for sump level indication and containment, ECC and Containment Spray Systems continue to be met without reduction in safety margins.

16. Modify the Reactor Vessel Head Vent System
DCP J-43994 Rev. 3 (Unit 1)

Modifications to the Reactor Vessel Head Vent System (RVHVS) were made to improve the ability to detect, isolate, and test for leakage of the RVHVS solenoid valves and to replace these valves, if necessary. New flanged connections, manual isolation valves, separated flow paths and a pressure test connection were added. A revised design to accommodate the blowdown/venting of steam/water/non-condensibles from the RVHVS into the containment was also provided. Minor electrical cable and termination box changes were made to accommodate relocation of and/or access to the piping components.

Safety Evaluation Summary

The required and existing function and integrity of the RVHVS has been maintained with these changes to the system. Design, materials and construction have maintained the integrity of the reactor coolant pressure boundary (RCPB) so that a LOCA is no more likely. Though new potential leak paths (from valve packing) exist, any resulting leakage is not judged to create or make more likely any design basis events of the types previously evaluated. The addition of a small amount of combustibles (PVC drain hose) has been evaluated as acceptable.

Seismic and fluid dynamic loads, separation, redundancy, independence and SISI criteria and flow path integrity of the RVHVS have been maintained so as to not increase the potential for malfunction of this accident mitigation system. Control of the fluid discharged by the system prevents the occurrence or creation of malfunctions of other ITS systems or components. Electrical modifications maintained the integrity (including EQ) of associated components and circuits. No radiological consequences of previously-analyzed events are increased since no mitigating action or assumption or fission product barrier is adversely affected by these changes to the RVHVS.

DCPP TS in the following areas were reviewed to assure no margins of safety were reduced: 1) RCS operational leakage: because of the integrity of the new valves and piping configuration, no new Pressure Boundary Leakage will occur; new manual valve packing is a potential, but small, source of Identified Leakage well within the leak rate limitations specified; 2) RCS Structural Integrity is maintained through appropriate application of the ISI Program on this modification; 3) Reactor Vessel Head Vents: flow path, operability, redundancy, testing, etc. required for this system are maintained undiminished; 4) ECCS Subsystems: failure of the plastic hose connected to the drain trough for the RVHVS cannot compromise the ECCS Recirculation Sump function or performance.

17. <u>Rename and Reconfigure the Traveling Crew's Quarters to the new Operations Ready Room</u>
DCP A-44893 Rev. 0 (Unit 2)

This design change renames the former Traveling Crews' Quarters to the Operations' Ready Room on the 119' level in the southwest corner of the Unit 2 Turbine Building. It also adds lockers and partitions walls in the area.
<u>Safety Evaluation Summary</u>

This area of the plant and its contents are not associated with the cause or credited for the mitigation of any accidents or malfunctions of ITS equipment, nor can they create the possibility of such events. This area is not a 10 CFR 50, Appendix R area and contains no safe shutdown systems, circuits or devices. FHARE 44 documents the acceptability of changes in combustibles loading in this area resulting from the modifications. Modifications do not affect the structural integrity of this portion of the Turbine Bldg.

This area of the plant does not relate to any requirements of the TS or margins of safety related to the licensing basis of the plant.

18. <u>Provide for Smaller Size Particle Ratings for Filters associated with the Spent Fuel Pool and Refueling Water Inventories</u>
DCP N-45082 Rev. 0 (Unit 1) and N-46082 Rev. 0 (Unit 2)

This DCP permits use of filter elements with smaller particle size ratings (<5 microns) for the Spent Fuel Pool (SFP) Filter, SFP Skimmer Filter, SFP Resin Trap Filter and Refueling Water Purification Filter. Reduced levels of radioactive contamination, specific activities, dose rates, etc. are expected, consistent with ALARA practices, with filter elements rated below 5 microns.

<u>Safety Evaluation Summary</u>

The four filters do not have any safety related filtering or flow function. Flow or filtering of fluid through the devices neither can contribute to the cause, nor are required for mitigation, of any FSAR Update accidents. The alternate elements are compatible with their fluid environments and operating conditions.

The replacement elements can have no impact on the structural or pressure boundary integrity of the filter vessels in which they are placed; they will not degrade the design basis nor challenge the function of any ITS equipment.

The TS requirements for systems' integrity and performance and water inventories/levels remain uncompromised so that no margins of safety are reduced.

19. Provide for Segregation of the Cleanup Systems for the Spent Fuel Pool and Refueling Water
DCP N-45099 Rev. 0 (Unit 1)

A localized Design Class II piping modification was made in the Refueling Water Purification (RWP) system. The change allows for the simultaneous use of the RWP Filter (to clean up the water in the refueling canal or RWST) and the Spent Fuel Pool (SFP) Demineralizer and Filter (to clean up the water in the SFP). The piping and valve materials used are compatible with their service conditions.

Safety Evaluation Summary

These portions of the RWP and SFP cleanup systems are not associated with the cause of any FSAR Update accidents. Further, though these cleanup systems interface with the ITS SFP and RWST, the modification made under this DCP does not challenge or degrade these ITS features. There are no new interconnections between plant systems which did not already exist. The potential for an SISI was evaluated and resolved. This configuration change does not affect accident mitigating features, actions, assumptions or fission product barriers. No system or component failure modes are altered. The TS requirements for SFP, RWST and Refueling Canal levels and boron concentrations are not impacted by this modification.

20. Remove the Automatic the Isolation Function for RE-32 and -33
DCP J-45446 Rev. 0 (Unit 1) and J-46446 Rev. 0 (Unit 2)

This modification removed the automatic isolation function for the plant vent (post-accident, mid-range, iodine and noble gas) radiation monitors RE-32 and -33. Before the change, when RE-29 (plant vent gross gamma) reached a high radiation level, the sample valve, RCV-32, closed to stop sample flow to RE-32 and -33. This reduced accumulated contamination of the absorber filters and sample lines. The high range I_2 grab sampler, RX-40, was then to be used until RE-29 levels dropped to allow use of RE-32 and -33 again.

Calculations showed that even if not isolated, RE-32 and -33 would not go offscale high during post-accident conditions. In addition, operators could take action (i.e., stop the sample pump or close manual isolation values) to stop flow to RE-32 or -33, if desired.

Safety Evaluation Summary

The radiation monitoring components associated with this change cannot contribute to the cause of any previously-analyzed accidents. These instruments are used for accident monitoring, evaluation and mitigation. Not isolating the sample flow will not degrade or

challenge the post-accident performance of the two mid-range monitors, since a calculation has shown an adequate monitoring function for the duration of the accident.

Post-accident occupational and offsite exposures to radiation are not increased since: 1) the pressure boundary integrity of the sampling loop has not been compromised, 2) sample flow can still be controlled (on/off), and 3) replaceable filter change out following an accident can still be performed as before, if necessary.

RCV-32 is an MOV and fails as-is. No new failure modes are created by disabling the valve in the open position. This new configuration also remains incapable of creating a different licensing basis event.

No TS directly relates to RE-32 or -33; but, as post-accident instrumentation, these components continue to meet PG&E commitments to Reg. Guide 1.97 and NUREG-0737. Also the TS-required functions of HVAC systems discharging to the plant vent are not affected in any way.

21. Convert and Modify the Monitor Tanks' System to a Boric Acid Reserve Tanks' System
DCP N-45505 Rev. 2 (Units 1 and 2)

To meet a licensing commitment and operational needs for both Units, an additional supply of 4% boric acid solution has been provided. The two, Design Class II, formerly Unit 2 Monitor Tanks (17,500 usable gallons each) were converted to Boric Acid Reserve Tanks (BARTs). Existing pumps plus new Design Class II pumps and boric acid heaters comprise the major equipment. Piping, valves and instruments are provided to add from or discharge BART contents to other tanks. However, their primary purpose is to allow for rapid refilling of the Boric Acid Storage Tanks (BASTs). The heaters, tank and piping insulation and recirculation of BART contents assures the boric acid solution will remain above 65°F to prevent precipitation. Each BART has an internal, floating Hypalon cover and N_2 purge gas.

The Laundry/Distillate Tanks in Unit 1 now collect the Unit 2 rejected distillate that was previously collected in the now-converted Monitor Tanks.

Safety Evaluation Summary

The BARTs, in their previous and present functions, have no relation to the cause or mitigation of the design basis events described and analyzed in the FSAR Update. They provide a support function for the ITS BASTs. No BART system malfunctions or failures can be propagated to the BASTs which will compromise the integrity, capacity or function of the BASTs.

The modifications cannot affect any other ITS systems or components because: only non-vital power is provided for all electrical loads; no 10 CFR 50 Appendix R circuits are affected; area combustible loadings have not significantly changed; there is no change to internal flooding potential inside the Auxiliary Building; the safety related Auxiliary Building structure is adequate for all BART modifications; and, no SISI issues are created by these changes.

In keeping with a license amendment commitment, the BARTs enhance DCPP's ability to meet TS requirements for boric acid inventories in the BASTs.

22. Abandon the Containment Humidity Monitoring System
DCP J-45574 Rev. 0 (Unit 1) and J-46574 Rev. 0 (Unit 2)

This design change abandons in place the containment humidity monitoring/indication system. It was originally installed to provide an additional, indirect method of (RCS) leak detection; it performed no control function and was Design Class II. Other, remaining systems continue to provide the ability to detect, to the extent possible, the source, size and location of leakage from the various steam and water-containing systems within the containment. This humidity monitoring system was not used for the periodic containment ILRT.

Safety Evaluation Summary

Abandoning an instrumentation system with no control or interlock functions cannot relate to the cause of occurrence of any accidents or malfunctions. Similarly, accident-malfunction consequences cannot be increased since four other diverse leak detection systems still remain in the containment.

Even with this change, the RCS leakage detection systems continue to meet TS requirements, PG&E commitments to Regulatory Guide 1.45, and NRC SER acceptance of DCPP leak detection system design per GDC 30.

23. Repipe Waste Filter 0-5 as an Ion Exchanger Effluent Polishing Filter
DCP N-45591 Rev. 0 (Units 1 and 2)

A redundant, Liquid Radwaste System (LRS) discharge filter was repositioned in the LRS process to serve as a polishing filter to remove particulates from the Waste Ion Exchanger effluent flow. The filter remained in the same physical location but piping was rerouted to change its process location. The filter is suitable for the conditions (flow, pressure, etc.) to which it will be exposed. In addition, wiring changes were made related to alarm and pump trip associated with the filter's function.

Safety Evaluation Summary

The structures, systems and components associated with this design change cannot cause and do not contribute to or mitigate any FSAR Update accidents or their consequences. Similarly, the modifications to the LRS maintain its functional process and pressure boundary integrities so that no degradation or challenge to ITS equipment can result; since no ITS equipment can be impacted by this change, no malfunction consequences are increased.

All of these modifications to the LRS are Design Class II, localized in areas without SISI targets, maintain LRS design and function and comply with DCPP commitments to Regulatory Guide 1.143; therefore, no new licensing basis events are created or made credible.

These changes have not compromised the margins of TS 3.11.1 for liquid radioactive effluents in the following areas: concentration and detection of radioactive materials discharged, exposures and doses to the public and operability of the LRS.

24. Justify As-Built Configuration/Separation of Safety-Related Ammeters
DCP E-45673 Rev. 0 (Units 1 and 2)

The main control boards in the Control Room contain redundant Class IE ammeters for the major, safety related pump motors and containment fan cooler motors. Between the two Units, a total of 40 ammeters on the Control Room control boards monitor the currents for these motors which are provided with 4160 volt or 480 volt vital power. The ammeters are connected to the secondary side of current transformers associated with these power circuits. As originally installed, they have a minimum separation of 3" between redundant circuits. This is less than the 5" separation criterion as stated in the FSAR Update. With appropriate technical basis, the plant design criteria documents and FSAR Update were revised to allow for this as-built condition. All other (including future) ammeter installations must comply with the 5" criterion (or other approved configuration) for mutually redundant installations.

Safety Evaluation Summary

The separation criterion for plant electrical systems assures that, in part, a failure occurring in one circuit cannot compromise a mutually redundant circuit. This criterion may be satisfied by various means: i.e., barriers or air gaps. These ammeters were installed during original construction of the plant with an air gap between mutually redundant meters of $\geq$ 3". An adequate technical basis for this configuration, including review of industry standards, justifies that a single component failure will not be propagated to adjacent, redundant components.

These ammeters and their various credible failure modes are not associated with the cause of any FSAR Update accidents. An air gap of as little as 3" is not more likely to cause a malfunction of the vital bus power supply or the motor which is monitored by the meter. Since, also, accident mitigation cannot be compromised with this physical layout and equipment failure modes are not changed, no radiological consequences are increased.

This localized condition in the main control boards is not capable of creating a new design basis event of the types already evaluated. Also, based on the single failure criterion, one ammeter in a mutually redundant group may fail; this, however, (apart from potentially causing a loss of its respective motor or motor power supply) cannot result in a new or different degradation or challenge to ITS equipment.

No numerical acceptance limit was identified by the NRC for minimum separation. The requirements for separation are judged to still be met for TS related to vital power and ECC systems. Therefore, no margins of safety are reduced.

25. Shorten the Spent Fuel Handling Tool Lifting Bail
DCP N-45869 Rev. 1 (Unit 1)

The lifting bail (handle) on the spent fuel handling tool used in the Spent Fuel Pool was shortened by 4". This was done to provide additional clearances during fuel handling operations.

Safety Evaluation Summary

The structural integrity of the tool was maintained; it maintained the same design configuration for use in fuel handling operations; therefore, the analyzed fuel handling accident (FHA) is no more likely. The minimum, committed shielding depth of borated water, while moving fuel, is still maintainable with the shorter bail. The onsite and offsite dose consequences from the FHA are, therefore, not increased.
The tool is a passive, structural element between the spent fuel bridge crane and a fuel assembly. This modification will not impair operation of the crane, storage or use of the tool or the integrity of fuel assemblies. A malfunction during the fuel handling operation will not (measurably) increase the onsite (crane operator) dose consequence above that already analyzed.

Existing accidents and malfunctions associated with use and failure of this tool bound all credible, potential events such that no different events are created.

TS 3.9.10 through 3.9.14 define the requirements for spent fuel handling, storage and protection. None of these requirements is compromised by this change.

B. Procedure Changes

1. Operating Procedure: Emergency Operation of Motor Operated Valves
OP 0-22 Rev. 2 (Units 1 and 2)

This procedure change was prepared to give instructions for operating motor operated valves using the open and close contactors located inside the breaker cubicle. This procedure would be used in an emergency upon the direction of the Site Emergency Coordinator. This procedure would be used only if the normal control switch circuit did not function because of a faulty torque switch and local actuation using the valve handwheel is not feasible.

Safety Evaluation Summary

This procedure is implemented only after an accident or malfunction has occurred. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

The operator manually manipulating the open and close contactors in the breaker cubical is in direct contact with an operator in the Control Room. The Control Room operator instructs the operator at the electrical cubical in the operation of the contactors while observing the valve position lights on the control panel. This coordinated operation prevents damage to the valves. Therefore, the consequences of an accident or a

malfunction of equipment important to safety already evaluated in the FSAR Update is not increased.

This procedure will be used only after the plant is in an accident condition and any valve operation will be so coordinated so as not to damage the valves. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not increased.

This procedure is implemented only after the plant is in an accident condition. Therefore, the margin of safety as defined in the basis of any Technical Specifications is not reduced.

2. Operating Procedure: Reactor Vessel-Draining to Half Loop/Half Loop Operations with Fuel in Vessel
OP A-2:III Rev. 7 (Unit 1) and Rev. 4 (Unit 2)

These procedure revisions changed the title of the procedures, making them applicable only when fuel is in the reactor. The 10 CFR 50.59 change to both procedures was to change a procedure prerequisite to verify that the electrical breaker for at least one of the RHR suction valves is closed. The closure of the electrical breaker for one of RHR suction valves would make the valve available to perform the its isolation function if needed during shutdown.

Safety Evaluation Summary

The FSAR Update allows power to the RHR suction valves during shutdown. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The consequences of a complete loss of RHR is not affected by the position of the RHR suction valve breakers. Also, maintaining power available to the RHR suction valves during shutdown has no affect on the consequences of a malfunction of equipment important to safety. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

Maintaining power to the RHR suction valves during shutdown is allowed by the FSAR Update. Therefore, the possibility of an accident or a malfunction of equipment important to safety, is not created.

The positions of the RHR suction valves and their associated electrical breakers are not specified in the Technical Specification for the operating modes where these procedures will be implemented. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

3. Temporary Procedure: Prevent Reactor Trip During 12KV Bus "E" Potential Transformer Fuse Replacement
TD-9007 Rev. 0 (Unit 1)

The implementation of this temporary procedure will make the 12KV Bus "E" underfrequency relays inoperable and place Unit 1 in a Technical Specification 3.0.3, 1

hour action statement. Jumpers prevent the 12KV Bus "E" underfrequency relays from tripping the reactor and reactor coolant pumps 1-1 and 1-3 breakers while the 12KV Bus "E" potential circuit is de-energized to replace potential transformer fuses. The 12KV Bus "E" bus undervoltage trip and the auto transfer feature will also be cut out during the bus potential transformer fuse replacement.

Safety Evaluation Summary

The implementation of this temporary procedure puts Unit 1 in a Technical Specification 3.0.3, 1 hour action statement. The procedure requires that an operator be stationed at the main control board to manually trip the two reactor coolant pumps in case of an underfrequency reactor trip. The procedure requires that circulating water pump 1-1 on 12KV bus "D" be selected for restart on auto-transfer. No uncompensated action will be taken which will alter system design specifications or degrade the reliability of systems important to safety. Therefore, the probability of occurrence an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The 12KV bus "D" will provide a reactor trip on undervoltage and the operator manual trip of the two reactor coolant pumps in the event of a reactor underfrequency trip will compensate for no underfrequency breaker trip. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The loss of reactor coolant pump breaker underfrequency trip is compensated for by the operator initiated breaker trip and disabling the 12KV Bus "E" auto-transfer to prevent a possibility of equipment malfunction. Therefore, the possibility of an accident or the malfunction of equipment important to safety of a different type than previously evaluated in the FSAR Update is not created.

The loss of forced reactor coolant flow coincident with reactor trip has been demonstrated to not exceed the safety margin. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

4. Chemical Analysis Procedure: Sampling of Primary Systems
CAP E-1 Rev. 7 (Units 1 and 2)

This procedure revision clarifies the section on sampling the Volume Control Tank gas spaces.

Safety Evaluation Summary

Implementation of this procedure makes no change to any accident scenario or source terms and the reactor coolant sampling system valve panel is PG&E Design Class II. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

Sample collection and analysis are outside the scope of the accident analyses and the reactor coolant sampling system is PG&E Design Class II. Therefore, the consequences of

an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

This procedure is a sampling procedure and is outside the scope of analyzed accidents and the reactor coolant sampling system valve panel is PG&E design class II. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

Implementation of this procedure makes no change to source terms. Therefore, the margin of safety as defined in the basis of any Technical Specification is not reduced.

5. <u>Operating Procedure: CVCS - Excess Letdown - Place in Service</u>
   OP B-1A:IV Rev. 3 XPR (Unit 1) and Rev. 2 XPR (Unit 2)

   These procedure revisions add discussion identifying the need to realign the seal water return flow to the Volume Control Tank (VCT) when operating with the excessive letdown in service for an extended time. Also added are the procedural steps to perform the realignment.

   <u>Safety Evaluation Summary</u>

   The realignment of the seal water return flow to the VCT does not significantly impact normal system operation and steps are included in the procedure to prevent equipment damage. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

   Implementation of these procedures requires the operator performing the realigning to be in constant communication with the Control Room and instructs the operator to return the excess letdown alignment to normal configuration in the event of a Safety Injection. The valves used to isolate charging pump recirculation are normally operated in the emergency procedures. Therefore, the consequences of an accident or a malfunction equipment important to safety, previously evaluated in the FSAR Update is not increased.

   The realignment of the seal water return to the VCT does not significantly impact normal system operation and steps are included in the procedure to prevent equipment damage. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

   Implementation of this procedure has no impact on Limiting Conditions of Operation during normal operations. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

6. <u>Temporary Procedures: Replacement of CRDM Fan 2-E14 or 2-E11</u>
   TP TD-9010 Rev. 0 (Unit 2)
   <u>Replacement of CRDM Fan E-11</u>
   TP TD-9011 Rev. 0 (Unit 1)

   These procedures provide guidance for removal, repair and reinstallation of Unit 2 CRDM Exhaust Fans 2-E14 or 2-E11 and Unit 1 CRDM Exhaust Fan E11 while in plant operating

modes 1-4. In addition, these procedures provide guidance for the temporary placement of an air flow restriction cover on the CRDM ducting while the fan is being repaired.

Safety Evaluation Summary

These procedures provide for all necessary prerequisites and critical time/temperature monitoring requirements for safely maintaining CRDM cooling. Jet impingement, seismic integrity, missile control issues, combustible loadings, aluminum loadings, and parts control have also been analyzed or addressed by these procedures. Therefore, the probability of occurrence and the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

Exhaust fan removal and repair has been performed in the past utilizing maintenance procedures. Additional restrictions have been placed in these procedures due to the implementation of the procedures in operating Modes 1-4 and the continuous operation of the fans. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

The probability of a double-ended Main Steam Line Break at a specific location is extremely small and the probability of having a break while the polar crane is traversing the jet impingement zone is considered incredible. The containment integrity is not affected by the performance of this procedure. The work area for performing the repair work and these procedures supply the necessary physical and administrative controls to prevent any materials from leaving the work area in the event of a LOCA/water spray condition. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

7. Temporary Procedure: CCW Heat Exchanger Performance Test
TB-9049 Rev. 0 (Units 1 and 2)

This procedure installs temporary instrumentation to support testing of the Component Cooling Water Heat Exchanger. Temporary ultrasonic flow indicators are installed on the outside of lines, temporary temperature sensors replace some permanent temperature sensors in existing thermowells, two temporary temperature sensors replace permanent thermowells and temperature sensors, and temporary flow transmitters are installed across existing flow transmitters.

Safety Evaluation Summary

The majority of the instruments installed by this procedure do not form any part of the existing pressure boundary. The temporary temperature sensors that replace the permanently installed thermowells and sensors are installed downstream of the Component Cooling Water Heat Exchanger where their failure will not block flow and any leakage caused by a temperature sensor failure will not affect the heat removal capability of the Component Cooling Water Heat Exchanger. The temporary instrumentation will not impact the seismic analysis and the equipment on the Auxiliary Saltwater and Component Cooling Water systems will be operated per normal plant procedures for normal or emergency conditions. Therefore, the probability of occurrence of an accident or a

malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The flooding analysis for the Auxiliary Saltwater system in the FSAR Update would envelope any failure of the temporary sensors. The removal of the thermowells does not increase the radiological consequences of any accident because it does not affect any radiological system. Also none of the instruments are required for normal operation or accident mitigation. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The existing and added temporary instrumentation do not affect the operation of components on either the Auxiliary Saltwater or the Component Cooling Water systems. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not increased.

The basis for the Technical Specifications associated with this change does not define any margin. Therefore, the margin of safety as defined in the basis for any Technical Specification is not increased.

8. <u>Chemical Analysis Procedure: Gaseous Radwaste Discharge Management</u>
CAP A-6 Rev. 8 (Units 1 and 2)
<u>Offsite Dose Calculations</u>
CAP A-8 Rev. 10 (Units 1 and 2)

The correction factor applied to the calculated high alarm setpoint for gaseous effluent monitors was changed from 0.375 to 0.300. The existing correction factor was in error and was unnecessarily conservative according to the manufacturer's specifications.

<u>Safety Evaluation Summary</u>

This change does not involve accident scenarios or their probability of occurrence and the change does not involve the malfunction of any equipment. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

This change does not involve accident scenarios or any consequence of an accident or a malfunction of relevant equipment. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

This change does not involve the possibility of an accident of any kind, different or that already evaluated in the FSAR Update or the malfunction of equipment not already considered in the FSAR Update. Therefore, the possibility of an accident or a malfunction of equipment important to safety, of a different type than any previously evaluated in the FSAR Update is not created.

The margin of safety is not defined in the basis for any Technical Specification. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

9. Temporary Procedure: Bypass Relay 27VER2 and Relay 81VER2 for Performance of STP I-9A
TP TB-9027 Rev. 0 (Unit 1)

This procedure bypasses the reactor trip signals generated by 12KV Undervoltage Relay 27VER2 and 12KV Underfrequency Relay 81VER2 so that Surveillance Test Procedure STP I-9A may be performed on the remaining operable undervoltage and underfrequency channel without producing a reactor trip.

Safety Evaluation Summary

Implementation of this procedure will put Unit 1 in a four hour and a two hour action statement. Both action statements will be in effect; however, they will not be entered at the same time. Should an undervoltage or underfrequency condition occur during the performance of this procedure, the automatic reactor trip, the reactor coolant pump breaker trip, and the auxiliary feedwater pump start circuitry for this condition will still be available. The addition of the jumpers will not degrade or alter the performance of the reactor trip system. Therefore, the probability of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

During the performance of this procedure, the undervoltage and underfrequency automatic reactor trip, reactor coolant pump breaker trip, and the auxiliary feedwater pump start circuitry are still available. Also the single failure criterion as described in the FSAR Update and Criterion 20 of the General Design Criteria is still maintained. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The loss of coolant flow reactor trip remains available at all times, all reactor trips and engineered safety feature actions associated with the 12KV system remain available at all times, and verification is made prior to the addition of the jumpers that any power supply disturbances induced during the jumper installation will not create an undesirable plant transient. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

Even though channel redundancy is reduced during the performance of this procedure, sufficient redundancy is to maintained in the reactor trip system to protect against loss of coolant flow and to adhere to the Technical Specification Action Statements. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

C. Tests and Experiments

1. Surveillance Test Procedure: Auxiliary Building Ventilation Charcoal Preheater Test
STP M-3C Rev. 6 (Units 1 and 2)

This Surveillance Test Procedure was changed to allow proper trending of the performance of the Auxiliary Building Ventilation Charcoal Preheater. A correction factor and chart were added to the procedure to enable comparison of the heater output with the Technical Specification value of the heater at a nominal voltage of approximately 460 Volts.

Safety Evaluation Summary

The addition of the correction factor to the calculated power dissipation in the procedure does not affect the Residual Heat Removal (RHR) pumps and therefore the likelihood of a pump seal break. This procedure change does not affect the overall performance of the RHR system or the safety function of the heater. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The addition of the correction factor to the calculated power dissipation in the procedure does not affect the operation of the RHR pumps or affect the ability of the preheater to perform its safety function. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

This procedure change does not operate the heater outside its design limits. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than already evaluated in the FSAR Update is not created.

The application of the correction factor in the procedure will ensure the Technical Specifications are met. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

2. Surveillance Test Procedure: Nuclear Power Range Incore/Excore Single-Point Calibration Data
STP R-13B Rev. 0 (Units 1 and 2)

This is a new surveillance test procedure that was written to allow a single incore flux map to be used for excore axial flux difference calibration. The change only affects the methodology used to estimate excore axial flux difference calibration data to be used for Nuclear Instrumentation System calibration. The methods used, the power range Nuclear Instrumentation System's hardware, and the Technical Specification's calibration requirements do not change.

Safety Evaluation Summary

This procedure requires no hardware changes and does not affect the performance of any system in a manner which would lead to the occurrence of an accident or increase the probability of an accident or malfunction already evaluated in the FSAR Update. Therefore the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The power range Nuclear Instrument System will continue to perform its reactor protection functions as designed. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update are not increased.

This procedure change does not degrade the performance of any system either directly or indirectly. It does result in more, but not excessively, conservative axial flux difference

indications. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

The required axial flux differences calibration criterion and surveillance frequency specified in the Technical Specifications is maintained. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

3. Surveillance Test Procedure: Reload Ascension Testing
STP R-40 Rev. 7 (Units 1 and 2)

This procedure change allows the Reactor Coolant System primary coolant flow testing be performed at the 73% Reactor Thermal Power (RTP) testing plateau instead of the 50% testing plateau. Testing at the higher level provides greater confidence in the results of the flow measurements.

Safety Evaluation Summary

This procedure change only sets a new power level at which the reactor coolant flow rate data is taken and the new level is allowed by the Technical Specifications. Therefore, the probability of occurrence of an accident or the malfunction of equipment important to safety, previously evaluated in the FSAR Update, is not increased.

Indicated reactor coolant system flows point to no change in the flows from the last fuel cycle. Therefore, the consequences of an accident or of a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

This procedure change makes no changes to the plant configuration and does not affect plant equipment. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than already evaluated in the FSAR Update is not created.

This procedure is consistent with the Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

4. Surveillance Test Procedure: Routine Surveillance Test of the Fuel Handling Building Ventilation System
STP M-5 Rev. 9 XPR (Units 1 and 2)

This procedure allows operation of the Fuel Handling Building ventilation system with the supply fan off in operational Modes 4, 5, and 6. This will allow the negative pressure of the building to meet the Technical Specifications requirement of -1/8" W.G.

Safety Evaluation Summary

This procedure is implemented only in Modes 4, 5, and 6 where the prime functions of the system in these modes are not degraded. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update will not be increased.

This procedure is implemented only in Modes 4, 5, and 6 where the exhaust portion of the Fuel Handling Building Ventilation System is the critical component to mitigate consequences of a spent fuel pool fuel handling accident by maintaining a negative pressure in the Fuel Handling Building and treating exhaust air. Therefore, the consequences of an accident, or a malfunction of equipment important to safety, previously evaluated in the FSAR Update are not increased.

This procedure is implemented only in Modes 4, 5, and 6 where the original intent and assumptions in a Fuel Handling Building Fuel Handling Accident is that the building is maintained at a negative pressure and the exhaust air is treated for removal of radionuclides. Therefore the possibility of an accident or of a malfunction of equipment important to safety, of a different type than any already evaluated in the FSAR Update is not created.

The Technical Specifications' area temperature limitations are not affected by the implementation of this procedure in Modes 4, 5, and 6, since the Technical Specifications' equipment is not operated in these modes. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

5. <u>Surveillance Test Procedure: Penetration 79 Containment Isolation Valve Leak Testing</u>
   STP V-679 Rev. 7 (Unit 1)

   This procedure provides firewater system jumpers that allow the containment firewater system to be operable during the local leak rate testing of the firewater containment isolation valves.

   <u>Safety Evaluation Summary</u>

   The FSAR Update postulates only one fire at a time. The firewater jumpers added by this procedure provide sufficient water for more than two firewater stations. Additional fire extinguishers and a continuous fire watch are provided during the time the containment is open during outages. Therefore, the probability of occurrence or the consequences of an accident or malfunctions of equipment important to safety previously evaluated in the FSAR Update is not increased.

   The manual fire fighting capability provided in the containment and the addition of the fire hose jumpers provided by this procedure that supply firewater to the containment provide more-than-adequate fire fighting capability. Therefore, the possibility for an accident or a malfunction of equipment important to safety of a different type than any evaluated previously in the FSAR Update is not created.

   The firewater jumpers provided by this procedure will supply firewater to the fire hose stations in containment as specified in the Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

6. Surveillance Test Procedure: Thimble Tube Inspection Program
STP R-22 Rev. 1 XPR (Units 1 AND 2)

This revision to procedure STP R-22 changed the method of implementing the maximum thimble tube wall loss evaluation criteria. The 10% measurement uncertainty was deleted now that the test method has been verified and equations for both fresh and old growing scars were added.

Safety Evaluation Summary

The thimble tubes support operation of the Movable Incore Detector System (MIDS) and do not provide a safety related application. The accident of concern is a small break loss of coolant accident that has been evaluated in the FSAR Update. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The FSAR Update evaluation of a small break loss of coolant addresses a hole size of 0.375 inches and the nonsafety-related thimble tube ID size is 0.21 inches. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The thimble tubes support the operation of the MIDS, a nonsafety-related system, and the failure of a thimble tube is already bounded by the small break coolant loss evaluation in the FSAR Update. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated is not created.

The failure of a thimble tube does not prevent the use of the MIDS to meet Technical Specification requirements. Therefore, the margin of safety as defined in the basis of any Technical Specification is not reduced.

7. Surveillance Test Procedure: Charging Injection System Piping Pressurization
STP X-234 Rev. 0 (Unit 1)

This procedure is an ISI Pressure Test performed on the Charging Injection System at normal operating pressure.

Because the test is performed at normal operating pressure the plant must be in operational Mode 3 and the Charging Injection System is required to be operational. During the test the Charging Injection System is not in its normal Mode 3 alignment.

Safety Evaluation Summary

To maintain the integrity of the system the test rig will use only high pressure fittings and the test rig check valve will be installed as close as possible to the system test isolation valve. During the test, communication with the Control Room will be maintained so the test isolation valve can be closed immediately to return the system to its normal Mode 3 alignment if a Safety Injection should occur. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

The accidents affected by this test are the ones that require actuation of the Charging Injection System. The malfunction of equipment important to safety would be a leak in the test rig and its connection to the system test isolation valve. The test rig uses only high pressure fittings and is pressure tested prior to opening the isolation valve and communication with the Control Room is maintained during the performance the test. Therefore, the consequences of an accident or of a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The only accidents affected by this test are those that require actuation of Safety Injection. The Charging Injection System failure mode that would be affected by this test is the loss of system integrity. The Charging Injection System loss of system integrity is mitigated by the use of high pressure fittings on the test rig and the immediate closure of the isolation valve if a Safety Injection occurs. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

During the performance of this test, system integrity will be maintained and the isolation valve will be closed immediately if a Safety Injection occurs. Therefore, the margin of safety as defined in the basis of any Technical Specification is not reduced.

8. <u>Surveillance Test Procedure: Exercising CVCS Charging Isolation Valves 8107 and 8108</u>
STP V-3K13 Rev. 4 XPR (Units 1 and 2)

This procedure revision changes the time allowed for closing the Chemical and Volume Control System (CVCS) charging line isolation valves from 9.7 seconds to 14 seconds. The new time meets the system requirements after the authorized removal of the Boron Injection Tank.

<u>Safety Evaluation Summary</u>

The closing time for the CVCS charging line isolation valves is of concern with respect to accident mitigation only. the closing time of these valves is unrelated to the cause of any sort of accident evaluated in the FSAR Update. The only safety related equipment that could be adversely affected by increasing the closing time of the CVCS charging line isolation valves are the Centrifugal Charging Pumps (CCPs). It has been determined that if the operation of one of the pumps in a runout condition were to occur in the few seconds between the time that safety injection line valves are fully open and the CVCS charging line isolation valves are fully closed, that the pump would continue to run and be able to pump its design flow rate. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased; and the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

The CVCS charging line isolation valves close to support a safety injection function only in the event of a loss of coolant accident (LOCA). A Westinghouse evaluation has determined that where only one CCP was running, (worst case condition) a 14 second closing time for the CVCS charging line isolation valves would not cause the pump to trip or cause any cavitation that would adversely affect the pumps capability to perform its

design basis safety function. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update are not increased.

The Technical Specifications require operability of the CCPs for safety injection purposes in the Technical Specifications since the operation of at least one of the pumps is assumed in the accident analysis. As discussed above, the ability of the CCPs to perform their design basis safety function is not reduced by the increased CVCS charging line isolation valves closure time. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

D. Mechanical Bypasses, Jumpers, and Lifted Circuits

1. Jumper: Power to Nuclear Instrument Back-up Transformer TY-21
   Jumper Log No. 90-XX (Unit 2)

   This electrical jumper provided temporary power to the Instrument A.C. Back-up Power Supply Transformer TY-21 during a Bus 26 outage. The jumper was removed prior to Mode 4 operation.

   Safety Evaluation Summary

   The equipment that was connected to the back-up transformer by this jumper was not required to operate in operational Modes 5 and 6. Therefore, the possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR Update was not created.

   Affected equipment was not required to be operable when this jumper was installed. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

   The affected equipment was not required to be operable and no credit was taken for the equipment while the jumper was installed. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, previously evaluated in the FSAR Update was not increased.

2. Jumper: Bypass Open Torque Switch During Open Stroke on Motor Operated Valves SI-1-8802B and SI-1-8803B
   Jumper Log No. 90-XX (Unit 1)

   The purpose of this jumper was to electrically bypass the open-circuit torque switch for the full stroke of valves SI-1-8802B and SI-1-8803B and automatically permit placing it back in the circuit at the end of the stroke to provide its intended backup to the limit switch function.

   Safety Evaluation Summary

   The primary function of the open circuit torque switch is backup to the open limit switch. The ultimate function of the torque switch remains the same. Therefore, the possibility

for an accident or malfunction of a different type than any evaluated previously in the FSAR Update was not created.

The limit switch will continue to be the primary motor control with the torque switch providing a backup function. Therefore, the margin of safety as defined in the basis of any Technical Specification was not reduced.

There has been no change to the functioning of the limit and torque switches. Both the limit switch and the torque switch will retain their intended function. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update, was not increased.

3. <u>Jumper: Main Turbine Autostop Oil/EH Interface Valve PCV-23</u>
Jumper Log No. 90-XX (Unit 2)

This jumper is a mechanical jumper that is put in place only when the Unit is shutdown. It supplies a pressurized source of nitrogen gas to the diaphragm of valve PCV-23 allowing the EH Header to be pressurized, latching the turbine steam valves (STOP, GOVERNOR, REHEAT and INTERCEPT VALVES).

<u>Safety Evaluation Summary</u>

The jumper will be in effect only when the Unit is shutdown and there is no steam to run the turbine in this condition. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

The jumper will be used when the reactor is cold and steam is not available and the turbine is not operating. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

The jumper is installed when the Unit is shutdown and no steam is available. The jumper is removed prior to entry into operational Mode 4. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

4. <u>Jumper: "Hutch Interlock" for the Main Steam Isolation Valve (MSIV) Bypass Valve Control Switch</u>
Jumper Log No. 90-040 (Unit 2)

This jumper utilizes a "Hutch Interlock" to physically hold the MSIV bypass valve control switch in the open position and allow the steam line condensation to be drained via the MSIV steam traps in operational Mode 4 only.

<u>Safety Evaluation Summary</u>

This jumper will be installed only in Mode 4 when the MSIVs are not required by to be operable by the Technical Specifications. Therefore the possibility for an accident or malfunction of a different type than any evaluated in the FSAR Update is not created.

Main steam isolation is not required in Mode 4 to maintain a margin of safety. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

No steam line isolation is required in Mode 4. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

5. Jumper: Auxiliary Building Supply Fans S-33 and S-34
Jumper Log No. 90-053 (Unit 2)

These jumpers are temporary air jumpers with regulators connected to the inlet vane positioners FC-5038 and FC-5039 that allows manual positioning of inlet vanes on the Unit 2 Auxiliary Building Supply Fans S-33 and S-34.

Safety Evaluation Summary

The safety function of the fans is not affected by the jumper. The regulator allows positioning the inlet vanes from full closed to full open. The system logic will still switch the system to the Safeguards Only Mode if one fan should fail when the jumper is installed. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

The fans will be able to perform their safety function while the inlet vanes are jumpered. The temperature in the building and pump rooms will not rise during a fan failure as the fans would operate at rated output. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

Both fans will be operating at full output when the jumper is installed. In the event of a fan failure the remaining fan continues to supply designed airflow in the Safeguards Only mode. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

6. Mechanical/Bypass: CO_2 System Hose Reels
Jumper Log No. 90-077 (Unit 2)

This jumper is temporary mechanical jumper, a blind flange, installed in the CO_2 Supply line to the Unit 2 CO_2 hose reels. The jumper provides isolation only to the portion of the CO_2 system that requires modification to support the installation of the third diesel generator for Unit 2, while other portions of the system continue to provide protection.

Safety Evaluation Summary

Three hour rated fire barriers are utilized throughout the area that is normally protected by the isolated portions of the CO_2 system. These barriers provide the necessary separation required to protect redundant trains of shutdown equipment. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

A 24 hour firewatch with portable fire suppression equipment will be stationed as required while this the temporary jumper is in place. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

Although portions of the CO_2 system are being rendered inoperable when the jumper is in place, the 3-hour rated barriers separating the redundant trains of shutdown equipment preclude the possibility of a single fire damaging both trains of equipment. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

7. Mechanical/Bypass: CO_2 System, Diesel Generator Rooms
Jumper Log No. 90-079 (Unit 2)

This jumper is a temporary mechanical jumper, a blind flange, installed in the CO_2 supply line to isolate the portion of the Unit 2 CO_2 fire suppression system which protects the Unit 2 diesel generator and vital switchgear areas. The installation of the jumper allows other portions of the CO_2 system to provide protection during modification of the isolated portion of the system.

Safety Evaluation Summary

All areas affected by this jumper are also provided with firewater hose stations or portable fire extinguishers or a combination of both which provide the same level of coverage as the CO_2 system. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

While this jumper is in place, a 24 hour firewatch, equipped with portable fire suppression equipment, will be stationed between the two Unit 2 diesel generators, providing coverage commensurate with the Technical Specifications. Therefore the margin of safety as defined in the basis for any Technical Specifications is not reduced.

While this jumper is in place, portions of the CO_2 system are rendered inoperable. Each area covered by an inoperable portion of the CO_2 system is also provided with water hose stations, or fire extinguishers or a combination of both that provide adequate fire suppression. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

8. Temporary Modification, Plant Jumper: Installation of Test Equipment
Jumper Log No. 91-XX (Unit 1)

This jumper will install more accurate temperature and flow instruments on the inlet and outlet of the Component Cooling Water Heat Exchanger. Operational temperature instruments on the inlet and the outlet of the Component Cooling Water are replaced temporarily with more accurate instruments. Test flow D/P cells are connected across the existing operational flow D/P cell. The jumper is required to be in place to perform Temporary Procedure TB-9049 and will be removed prior to entry into operational Mode 4.

Safety Evaluation Summary

The temperature instruments are light in weight and meet the exclusion criteria of the SISI Manual. The flow D/P cells are valved out of the system except when in use. The new temperature instruments are installed in the existing thermowells so the design integrity of the system will be preserved and any leakage from the new flow D/P connections has been evaluated. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

The instruments replaced by the installation of this jumper are not described, defined, or used to monitor the Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

The temperature instruments will be placed in existing thermowells, seismic concerns have been addressed; and, the existing leakage analysis will cover the new instrument installation. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

9. Temporary Modification, Plant Jumper: Installation of Test Equipment
Jumper Log No. 91-XX (Unit 1)

This jumper installs temporary instrumentation to support testing of the Component Cooling Water Heat Exchanger. Two ultrasonic flowmeters that do not penetrate the piping are temporarily installed; two temperature sensors are removed from the thermowells and are replaced with temporary RTD detectors; and, two temperature sensors along with their thermowells are removed and replaced with special temperature probes.

Safety Evaluation Summary

The relevant accidents for the temporary instrument installation are damage from a seismic event and leakage caused by instrument failure. The installed test instruments are light in weight compared to supporting components and meet the exclusion criteria of the SISI Manual. The temporary flowmeters are mounted outside the pressure boundary, so no leak path is created. Two of the temporary temperature sensors are installed in existing thermowells so no leakage path is created. Leakage from the special temperature sensors installed is enveloped by the existing safety analysis. Therefore, the possibility for an accident or malfunction of a different type than evaluated previously in the FSAR Update is not created.

The affected instruments are not described in the basis for any Technical Specifications. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

The temporarily installed instruments will not impact system operation under normal or accident conditions. The special temperature probes will not restrict the flow in the system any more than the normally installed instruments. Therefore, the probability of

occurrence or the consequences of an accident or malfunction of equipment important to safety is not increased.

10. Jumper: Installation of Temporary Electrical Jumpers in the "Open" Circuit of the Residual Heat Removal (RHR) System Suction Valves
Jumper Log No. 91-XX (Unit 1)

This jumper consists of the installation of two electrical jumpers. One electrical jumper is installed in the "Open" circuit of each of the RHR suction valves.

The jumpers bypass the Solid State Protective System (SSPS) contacts and allow the suction valves to stay open while the SSPS is being modified. The jumpers will be installed when the plant operating mode is such that the SSPS "prevent opening" interlock has no effect on plant operations.

Safety Evaluation Summary

The function of the "prevent opening" interlocks, bypassed by the installation of this jumper, is to protect the RHR suction piping from overpressurization when the RHR system is aligned to the Reactor Coolant System (RCS). The jumper will be installed only when the plant is on RHR and will be removed before the RCS is pressurized. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

The plant operating mode already has the RHR suction valves open to provide an OPERABLE RHR train when the jumper is installed and the jumper will be removed before pressurizing the RCS. Bypassing the "prevent opening" interlock has no effect in this operational mode. Therefore, the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

The "prevent opening" interlock performs no function while the RHR suction valves are open to provide an OPERABLE RHR train and the interlock will be returned to service before pressurizing the RCS. Therefore the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

The jumper will be removed before the RCS is pressurized, restoring the "prevent open" interlock to service. Therefore, the proposed temporary modification does not reduce the margin of safety as defined in the basis of any Technical Specification.

11. Temporary Modification, Plant Jumper: Lifted Circuit to Disable the Automatic Containment Evacuation Alarm From the High Flux at Shutdown Alarm
Jumper Log No. 91-XX (Unit 1)

This lifted circuit removes the automatic actuation of the containment evacuation alarm (high flux at shutdown), generated by the source range (nuclear instrumentation system) instrumentation. The manual actuation from the Control Room is not affected. The lifted circuit eliminated spurious alarms. No credit is taken for the automatic alarm actuation feature in relation to fuel handling accidents as described in the FSAR Update.

Safety Evaluation Summary

The probability of boron dilution events is not increased by this lifted circuit. The lifted lead affects only the automatic alarm feature, placing the reliance for containment evacuation alarm on the manual actuation, which is not affected by the lifted circuit. Therefore, the probability of occurrence of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

Sufficient time is available and appropriate indications are monitored to preclude consequences, associated with not evacuating containment, from increasing. Therefore, the consequences of an accident or a malfunction of equipment important to safety, previously evaluated in the FSAR Update is not increased.

This lifted circuit only affects the automatic actuation of the containment evacuation alarm. Source range counts and count rates are closely monitored during core alterations, and manual containment evacuation alarm actuation can be quickly executed. Therefore, the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR Update is not created.

This lifted circuit does not affect the Technical Specifications or the bases for the Technical Specifications. Therefore, this lifted lead does not reduce the margin of safety as defined in the basis for any Technical Specification.

E. Temporary Shielding Requests

1. Temporary Shielding on Pressurizer Surge Line for Replacement of Pressurizer Heater Wires
TSR 90-008 Rev. 0 (Unit 2)

This request calls for the installation of lead shielding around the pressurizer surge line for ALARA concerns. The lead shielding weighs approximately 480 pounds. The lead shielding will it be tied around the surge line with some of the shielding resting on the top of a rupture restraint. The pressurizer surge line is not required to be operable when the temporary shielding is installed and will be removed prior to returning the pressurizer to service.

Safety Evaluation Summary

The impact of the additional weight of the shielding on the piping and rupture restraint has been reviewed and found to be insignificant. The shielding tie down arrangement has been reviewed and is considered structurally adequate so that it will not fail during a seismic event. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

A review of the impact of the temporary shielding on piping and rupture restraint stress has verified that the stresses will remain within the code allowables. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

The review of the temporary shielding demonstrates that the seismic qualifications of affected components are not adversely affected; and the installation of the shielding is such that in the event of a seismic event the shielding will not act as a missile. Therefore, the probability of the occurrence of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

2. Temporary Shielding on Line 2-S6-13-4SPL, for Pipe Support 92-9R Rework
TSR 90-057 Rev. 0 (Unit 2)

This Temporary Shielding Request installs approximately 900 pounds of lead shielding on line 2-S6-13-4SPL bounded by supports 92-23A and 92-71R. The shielding does nor alter the system functional parameters and will be removed prior to returning to operations Mode 4.

Safety Evaluation Summary

The piping calculations indicate that the weight of the shielding has insignificant impact on the piping stresses and piping seismic qualifications. The shielding weight is significantly less than the design capacity of the support.

The shielding tie down installation arrangement is considered structurally adequate so that it will not fail during a seismic event. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

With the review of the piping stress analyses and since the weight of the temporary shielding is significantly less than the support design capacity it is concluded that the piping and restraint stresses will remain within the code allows. Therefore, the margin of safety as defines in the basis for any Technical Specification is not reduced.

It has been demonstrated by analysis that the seismic qualification of affected components are not adversely affected by the addition of the temporary shielding and the shielding tie down design is such that in the case of a seismic event the shielding will not act as missiles. Therefore, the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

3. Temporary Shielding on Line 2-S2-62-4, for Reactor Coolant Pump Seal Water Out, Letdown Piping Repair
TSR 90-071 Rev. 0 (Unit 2)

Temporary shielding is being installed around line 2-S2-62-4 while it is operating. The shielding is required for ALARA concerns for the workers repairing a cracked weld on the letdown piping. Piping seismic evaluation has been reviewed to demonstrate the piping integrity with the additional load on the pipe.

Safety Evaluation Summary

The additional weight of the shielding has been evaluated for its impact on the piping seismic qualifications and pipe restraints and has been found acceptable. The shielding is installed so as to prevent it from falling, sliding or swinging. The system operation will not be affected. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created

A review of the piping, and applicable support and component stresses has verified that the additional shield weigh will not adversely affect the piping integrity and operability. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

It is demonstrated that the seismic qualification of the affected components are not adversely affected by the addition of the temporary shielding. The installation of the shielding is such that a new source of missiles is not created. Therefore, the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

4. Temporary Shielding on Line 2-S2-62-4, for Reactor Coolant Pump Seal Water Piping, Letdown Line Repair
TSR 90-091 Rev. 0 (Unit 2)

Temporary shielding is being installed around line 2-S2-62-4 while it is operating. The shielding is required for ALARA concerns for workers repairing a cracked weld on the letdown piping. Piping seismic evaluation has been reviewed to demonstrate the piping integrity with the additional load on the pipe.

Safety Evaluation Summary

The additional weight of the shielding has been evaluated for its impact on the piping seismic qualifications and pipe restraints and has been found to be acceptable. The shielding is installed so as to prevent it from falling, sliding or swinging. The system operation will not be adversely affected. Therefore, the possibility of an accident or malfunction of a different type than any previously evaluated in the FSAR Update is not created.

A review of the piping and applicable support and component stresses has verified that the additional shielding weight will not adversely affect the piping system integrity or operability. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

It is demonstrated that the seismic qualification of the affected components are not adversely affected by the addition of the temporary shielding. The installation of the shielding is such that a new source of missiles is not created. Therefore, the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

5. <u>Temporary Shielding on Lines 1-S1-927-14, 1-S6-508-8, 1-S6-509-8 and Temporary Steel Attached To Containment Annulus Structure</u>
TSR 91-025/-026 Rev. 0 (Unit 1)

The temporary shielding is required for ALARA concerns for the crew performing work activities in the area and the RHR sump. Piping Seismic Evaluation Stress Analysis and Civil Evaluation have been reviewed to demonstrate the integrity of piping and building structure for addition load due to temporary shielding.

<u>Safety Evaluation Summary</u>

The additional weight of the temporary shielding has been evaluated for its impact on the seismic qualifications of the RHR piping and the steel attached to the annulus structure and was found acceptable. The installation of the temporary is considered structurally adequate so that it will not fail during a seismic event and damage any SISI targets in the area. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

A review of piping, applicable support, and component stresses have verified that the additional shielding weight will not adversely affect the piping system operability or the annulus structure and are acceptable. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

It is demonstrated that the seismic qualification of affected components is not adversely affected by the temporary shielding. The installation of the temporary shielding is considered structurally adequate so that it will not fail during seismic event. Therefore, the probability of the occurrence of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

6. <u>Temporary Shielding placed on Supports 46-9V and 66-19R, Supports for Lines 1-S6-3798-3 and 1-S6-3801-3, for Work Performed in Cubicles for Reactor Coolant Pumps 1-1 and 1-4</u>
TSR 91-028/-030, Rev. 0 (Unit 1)

This lead shielding is required for ALARA concerns for the crew performing work in Reactor Coolant Pump 1-1 and 1-4 cubicals.

<u>Safety Evaluation Summary</u>

The additional weight of the temporary shielding has been evaluated for its impact on the seismic qualification of supports 46-9V and 66-19R and found acceptable. The installation of the temporary shielding is considered structurally adequate so that it will not fail during a seismic event. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

A review of the analysis for the supports has verified that the additional temporary shielding weight will not adversely affect the integrity of the pipe supports and the temporary shielding is acceptable. Therefore, the margin of safety as defined in the basis for any Technical Specification is not reduced.

It is demonstrated that the seismic qualification of affected components are not adversely affected. The installation of the temporary shielding is such that it will not create a new source (as defined in the SISI Manual) or affect any other safety related system. The lines that are located on these supports will be out of service during the time the temporary shielding is installed. Therefore the probability of the occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

7. Temporary Shielding on Line 1-S6-1141-3, Loop 1 RTD Manifold Return Header, Work on RCP 1-1 Pump and Motor
TSR 91-034 Rev. 0 (Unit 1)

Temporary Shielding is being installed on line 1-S6-1141-3 while the line itself is not operational during operating Modes 5 and 6. The temporary shielding is required for ALARA concerns for crew performing work on RCP 1-1 pump/motor.

Safety Evaluation Summary

The additional weight of the temporary shielding has been evaluated for its impact on the seismic qualification of piping and supports and found acceptable. The installation of the temporary shielding is considered structurally adequate so that it will not fail during a seismic event. Therefore, the possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR Update is not created.

A review of the piping and support analyses have verified that the additional weight of the temporary shielding will not adversely affect the integrity of the piping and supports and the temporary shielding is acceptable as documented. Therefore, the margin of safety as defined in the basis of any Technical Specification is not reduced.

It is demonstrated that the seismic qualification of affected components are not adversely affected. The installation of the temporary shielding is such that it will not create a new source (as defined in the SISI Manual) or affect any other safety related systems. The temporary shielding will be installed when the line is out of service. Therefore, the probability of the occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR Update is not increased.

F. Fire Hazards Appendix R Evaluations

1. Fire Hazards Appendix R Evaluation for Leaked Oil in the Reactor Coolant Pump Cubicle
FHARE 96 Rev. 0 (Unit 1)

Based on guidance provided in NRC Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," an engineering evaluation was completed to determine the impact of the lube oil leaked from Reactor Coolant Pump 1-4. The leak was repaired soon after it was discovered. Due to ALARA concerns plant personnel were unable to adequately clean up all the leaked oil. Engineering evaluated the condition of the lube oil remaining in the cubicle until cleanup efforts could be performed during the next refueling outage. FHARE 96 evaluated the fire hazards associated with the lube oil.

Safety Evaluation Summary

The results of FHARE 96 documented that the small amount of oil remaining in the Reactor Coolant Pump (RCP) 1-4 cubicle would not adversely impact DCPP's Fire Protection Program or its safe shutdown capabilities due to the following: 1) the quantity of oil remaining in the cubicle was minimal, 2) the flash point of the oil was 425°F which is much greater than the ambient temperature of the cubicle (approximately 105°F) or the temperature of any component exposed to the oil, 3) lack of ignition sources within the confines of the cubicle, 4) the RCP area is provided with automatic detection and suppression, and 5) the loss of RCP 1-4 due to a fire that had already been postulated for Appendix R safe shutdown. Thus, the probabilities or consequences of previously analyzed accidents and malfunctions of equipment important to safety were not affected. This condition did not create the possibility of an accident or a malfunction of equipment important to safety of a different type than any previously evaluated in the FSAR because the leaked oil was confined to the cubicle, and no new ignition sources had been introduced.