PSEG Public Service Electric and Gas Company

80 Park Plaza, Newark, NJ 07101 / 201 430-8217 MAILING ADDRESS / P.O. Box 570, Newark, NJ 07101

Robert L. Mittl General Manager
Nuclear Assurance and Regulation

June 15, 1984

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Attention: Mr. Albert Schwencer, Chief
Licensing Branch 2
Division of Licensing

Gentlemen:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
DRAFT SAFETY EVALUATION REPORT
OPEN ITEM STATUS

Attachment 1 is a current list which provides a status of the open items identified in Section 1.7 of the Draft Safety Evaluation Report (SER). Items identified as "complete" are those for which PSE&G has provided responses and no confirmation of status has been received from the staff. We will consider these items closed unless notified otherwise. In order to permit timely resolution of items identified as "complete" which may not be resolved to the staff's satisfaction, please provide a specific description of the issue which remains to be resolved.

Attachment 2 is a current list which identifies Draft SER Sections not yet provided.

In addition, enclosed for your review and approval (see Attachment 4) are the resolutions to those Draft SER open items listed in Attachment 3.

Should you have any questions or require any additional information on these open items, please contact us.

Very truly yours,

R L Mittl

B001
1/1

Attachments

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

FM05 1/2

ATTACHMENT 1

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
5a&d	2.4.5	Wave impact and runup on service water intake structure	Complete	6/1/84
7b	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	6/1/84
9	2.5.4	Soil damping values	Complete	6/1/84
10	2.5.4	Foundation level response spectra	Complete	6/1/84
11	2.5.4	Soil shear moduli variation	Complete	6/1/84
12	2.5.4	Combination of soil layer properties	Complete	6/1/84
13	2.5.4	Lab test shear moduli values	Complete	6/1/84
14	2.5.4	Liquefaction analysis of river bottom sands	Complete	6/1/84
15	2.5.4	Tabulations of shear moduli	Complete	6/1/84
16	2.5.4	Drying and wetting effect on Vincentown	Complete	6/1/84
17	2.5.4	Power block settlement monitoring	Complete	6/1/84
18	2.5.4	Maximum earth at rest pressure coefficient	Complete	6/1/84
19	2.5.4	Liquefaction analysis for service water piping	Complete	6/1/84
20	2.5.4	Explanation of observed power block settlement	Complete	6/1/84
21	2.5.4	Service water pipe settlement records	Complete	6/1/84
22	2.5.4	Cofferdam stability	Complete	6/1/84
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14	Complete	6/1/84
24	2.5.4	Soil depth models for intake structure	Complete	6/1/84
27	2.5.5	Slope stability	Complete	6/1/84

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
30	3.5.1.2	Internally generated missiles (inside containment)	Complete	6/1/84
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	6/1/84
43	3.8.2	SRV/LOCA pool dynamic loads	Complete	6/1/84
44	3.8.3	ACI 349 deviations for internal structures	Complete	6/1/84
45	3.8.4	ACI 349 deviations for Category I structures	Complete	6/1/84
46	3.8.5	ACI 349 deviations for foundations	Complete	6/1/84
47	3.8.6	Base mat response spectra	Complete	6/1/84
48	3.8.6	Rocking time histories	Complete	6/1/84
49	3.8.6	Gross concrete section	Complete	6/1/84
50	3.8.6	Vertical floor flexibility response spectra	Complete	6/1/84
53	3.8.6	Design of seismic Category I tanks	Complete	6/1/84
54	3.8.6	Combination of vertical responses	Complete	6/1/84
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	6/1/84
57	3.8.6	Rotational time history inputs	Complete	6/1/84
58	3.8.6	"O" reference point for auxiliary building model	Complete	6/1/84
59	3.8.6	Overturning moment of reactor building foundation mat	Complete	6/1/84
60	3.8.6	BSAP element size limitations	Complete	6/1/84
61	3.8.6	Seismic modeling of drywell shield wall	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
62	3.8.6	Drywell shield wall boundary conditions	Complete	6/1/84
63	3.8.6	Reactor building dome boundary conditions	Complete	6/1/84
64	3.8.6	SSI analysis 12 Hz cutoff frequency	Complete	6/1/84
65	3.8.6	Intake structure crane heavy load drop	Complete	6/1/84
67	3.8.6	Critical loads calculation for reactor building dome	Complete	6/1/84
68	3.8.6	Reactor building foundation mat contact pressures	Complete	6/1/84
69	3.8.6	Factors of safety against sliding and overturning of drywell shield wall	Complete	6/1/84
70	3.8.6	Seismic shear force distribution in cylinder wall	Complete	6/1/84
71	3.8.6	Overturning of cylinder wall	Complete	6/1/84
72	3.8.6	Deep beam design of fuel pool walls	Complete	6/1/84
73	3.8.6	ASHSD dome model load inputs	Complete	6/1/84
74	3.8.6	Tornado depressurization	Complete	6/1/84
75	3.8.6	Auxiliary building abnormal pressure	Complete	6/1/84
76	3.8.6	Tangential shear stresses in drywell shield wall and the cylinder wall	Complete	6/1/84
77	3.8.6	Factor of safety against overturning of intake structure	Complete	6/1/84
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	6/1/84
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84
81	3.8.6	Seismic displacement of torus	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
82	3.8.6	Review of seismic Category I tank design	Complete	6/1/84
83	3.8.6	Factors of safety for drywell buckling evaluation	Complete	6/1/84
84	3.8.6	Ultimate capacity of containment (materials)	Complete	6/1/84
85	3.8.6	Load combination consistency	Complete	6/1/84
92	3.9.2.2	Triple flued-head containment penetrations	Complete	6/15/84
95	3.9.3.2	Fatigue evaluation on SRV piping and LOCA downcomers	Complete	6/15/84
96	3.9.3.3	IE Information Notice 83-80	Complete	6/15/84
98	3.9.3.3	Design of bolts	Complete	6/15/84
99	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
110b	4.6	Functional design of reactivity control systems	Complete	6/1/84
124	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
125	6.2.1.5.2	Design drywell head differential pressure	Complete	6/15/84
129	6.2.2	Insulation ingestion	Complete	6/1/84
132	6.2.4	Containment isolation review	Complete	6/15/84
134	6.2.6	Containment leakage testing	Complete	6/15/84
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	6/15/84
145	9.2.2	ISI program and functional testing of safety and turbine auxiliaries cooling systems	Complete	6/15/84
146	9.2.6	Switches and wiring associated with HPCI/RCIC torus suction	Complete	6/15/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
152	9.4.4	Radioactivity monitoring elements	Complete	6/1/84
154	9.5.1.4.a	Metal roof deck construction classificiation	Complete	6/1/84
158	9.5.1.5.a	Class B fire detection system	Complete	6/15/84
159	9.5.1.5.a	Primary and secondary power supplies for fire detection system	Complete	6/1/84
161	9.5.1.5.b	Fire water valve supervision	Complete	6/1/84
162	9.5.1.5.c	Deluge valves	Complete	6/1/84
163	9.5.1.5.c	Manual hose station pipe sizing	Complete	6/1/84
164	9.5.1.6.e	Remote shutdown panel ventilation	Complete	6/1/84
165	9.5.1.6.g	Emergency diesel generator day tank protecton	Complete	6/1/84
174	13.5.2	Resolution explanation in FSAR of TMI Items I.C.7 and I.C.8	Complete	6/15/84
182	15.9.10	TMI-2 Item II.K.3.18	Complete	6/1/84
185	7.2.2.2	Trip system sensors and cabling in turbine building	Complete	6/1/84
190	7.2.2.7	Regulatory Guide 1.75	Complete	6/1/84
193	7.2.2.9	Reactor mode switch	Complete	6/1/84
194	7.3.2.2	Standard review plan deviations	Complete	6/1/84
197	7.3.2.5	Microprocessor, multiplexer and computer systems	Complete	6/1/84
200	7.4.2.2	Remote shutdown system	Complete	6/1/84
205	7.5.2.4	Plant process computer system	Complete	6/1/84
209	7.7.2.3	Credit for non-safety related systems in Chapter 15 of the FSAR	Complete	6/1/84
210	7.7.2.4	Transient analysis recording system	Complete	6/1/84
218	9.5.1.1	Fire hazards analysis	Complete	6/1/84
TS-3	4.4.5	Core flow monitoring for crud effects	Complete	6/1/84
LC-1	4.2	Fuel rod internal pressure criteria	Complete	6/1/84

JS:gs
M P84 80/12 5-gs

ATTACHMENT 2 DATE: 6/15/84

DRAFT SER SECTIONS AND DATES PROVIDED

SECTION	DATE	SECTION	DATE
3.1			
3.2.1		11.4.1	
3.2.2		11.4.2	
5.1		11.5.1	
5.2.1		11.5.2	
6.5.1		13.1.1	
8.1		13.1.2	
8.2.1		13.2.1	
8.2.2		13.2.2	
8.2.3		13.3.1	
8.2.4		13.3.2	
8.3.1		13.3.3	
8.3.2		13.3.4	
8.4.1		13.4	
8.4.2		13.5.1	
8.4.3		15.2.3	
8.4.5		15.2.4	
8.4.6		15.2.5	
8.4.7		15.2.6	
8.4.8		15.2.7	
9.5.2		15.2.8	
9.5.3		15.7.3	
9.5.7		17.1	
9.5.8		17.2	
10.1		17.3	
10.2		17.4	
10.2.3			
10.3.2			
10.4.1			
10.4.2			
10.4.3			
10.4.4			
11.1.1			
11.1.2			
11.2.1			
11.2.2			
11.3.1			
11.3.2			

CT:db

<u>ATTACHMENT 3</u>

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
92	3.9.2.2	Triple Flued-Head Containment Penetrations
95	3.9.3.2	Fatigue Evaluation of SRV Piping and LOCA Downcomers
96	3.9.3.3	IE Information Notice 83-80
98	3.9.3.3	Design of Bolts
99	3.9.5	Stress categories and limits for core support structures
125	6.2.1.5.2	Design drywell head differential pressure
132	6.2.4	Containment Isolation Review
134	6.2.6	Containment Leakage Testing
141g	9.1.3	Spent fuel pool cooling and cleanup system
145	9.2.2	ISI program and functional testing of safety and turbine auxiliaries cooling systems
146	9.2.6	Switches and wiring associated with HPCI/RCIC torus suction
158	9.5.1.5.a	Class B fire detection system
174	13.5.2	Resolution explanation in FSAR of TMI Items I.C.7 and I.C.8

ATTACHMENT 4

DSER Open Item No. 92 (DSER Section 3.9.2.2)

TRIPLE FLUED-HEAD CONTAINMENT PENETRATIONS

Additional information is needed to address the design considerations used for the triple-flued head containment penetration.

RESPONSE

For the information requested above see response to Question 210.37.

QUESTION 210.37 (SECTION 3.9.2)

In FSAR Figure 3.8.6, a Type A triple flued head containment penetration with bellows is shown for those cases where thermal expansion is to be accommodated. Provide a list of those systems and the penetration identification numbers where these triple flued head containment penetrations have been used. Describe the design considerations applicable to those penetrations and specifically address how fatigue, torsion, and stiffness modelling in piping analyses were considered.

RESPONSE

The bellows used in Type A triple flued head containment penetrations are designed, fabricated, tested, and examined in accordance with the requirements for Class 2 components of ASME B&PV Section III Code.

Non-NSSS:

The list of non-NSSS systems and their penetration identification numbers that use a Type A triple flued head are shown in Table 210.37-1.

The design considerations for Nuclear Class 1 flued heads consist of evaluation of the loads transmitted to the flued head by the piping from both sides due to:

a. Thermal expansion

b. Seismic reactions

c. Dead weight loads

d. Internal pressure

e. Dynamic loads

f. Thermal gradient effects through the flued head body

g. Thermal transient effects as a result of temperature and pressure changes in the system

h. Discontinuity effects resulting from dissimilar metal welds, if any

i. Fatigue analysis using cumulative usage approach (NB-3653.5 of Section III).

Nuclear Class 2 flued heads are evaluated to the loads listed above with the exception of items f., g., and i.

The Type A triple flued head containment penetrations are anchored to the building steel as shown in revised Figure 3.8-6. In the connecting piping analyses, the flued head is considered a rigid anchor. Piping reaction loads (forces, bending moments and torsion) are evaluated as stated above. Fatigue is considered per item (i) above and includes evaluation of the flued head and the butt weld between the flued head and the process pipe.

NSSS:

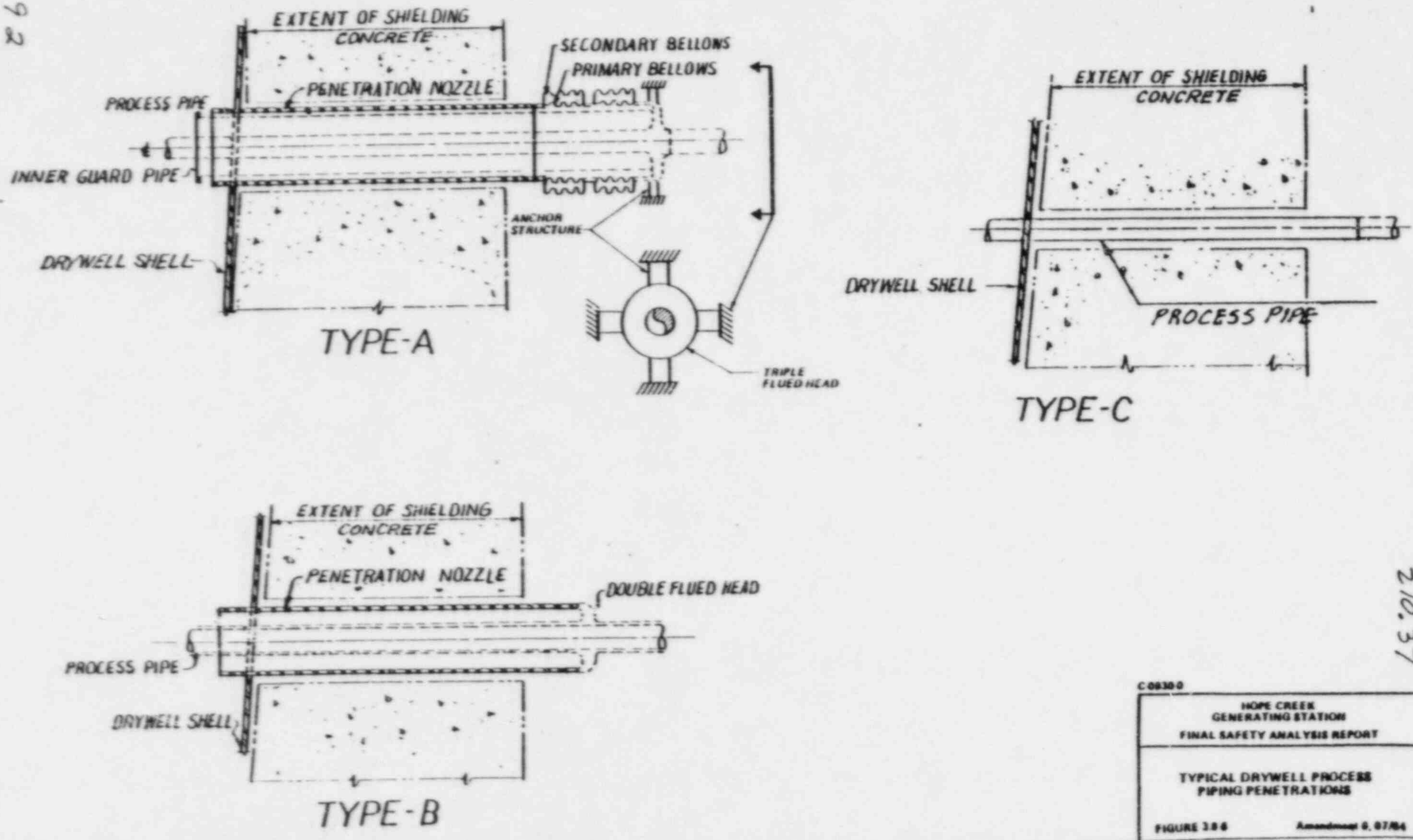
The main steam piping and the head fittings are designed and fabricated to the requirements of the 1971 edition of Section III for the ASME B&PV Code with addenda through and including those of Summer 1972. The main steam head fittings are analyzed to the requirements of NB-3200 and the 1977 editions of Section III of the ASME B&PV Code and are evaluated to more restrictive stress limits of BTP MEB 3-1 in SRP 3.6.2. The design report for main steam head fittings includes the evaluation of fatigue and the effect of pipe rupture loads. The head fitting is modelled as a pipe element with rigid stiffness, and its effect on the main steam piping is evaluated to the requirements of NB-3600 of Section III of the ASME B&PV Code.

TABLE 210.37-1

TYPE A TRIPLE FLUED HEAD CONTAINMENT PENETRATIONS

Penetration Assembly Number	Nom. Elev.	Service	Process Pipe Size	Scope of Supply
P-1A	107'-0"	Main Steam	26"	NSSS
P-1B	107'-0"	Main Steam	26"	NSSS
P-1C	107'-0"	Main Steam	26"	NSSS
P-1D	107'-0"	Main Steam	26"	NSSS
P-2A	113'-2"	Feed Water	24	Non-NSSS
P-2B	113'-2"	Feed Water	24"	Non-NSSS
P-3	106'-0"	RHR Shutdown Cooling From RPV	20"	Non-NSSS
P-4A	106'-0"	RHR Shutdown Cooling Return	12"	Non-NSSS
P-4B	106'-0"	RHR Shutdown Cooling Return	12"	Non-NSSS
P-5A	108'-9"	Core Spray to Reactor	12"	Non-NSSS
P-5B	108'-9"	Core Spray to Reactor	12"	Non-NSSS
P-6A	106'-0"	LPCI	12"	Non-NSSS
P-6B	106'-0"	LPCI	12"	Non-NSSS
P-6C	106'-0"	LPCI	12"	Non-NSSS
P-6D	106'-0"	LPCI	12"	Non-NSSS
P-7	106'-0"	HPCI Turbine Steam Supply	10"	Non-NSSS
P-8A	114'-0"	Chilled Water From Drywell Coolers	8"	Non-NSSS
P-8B	114'-0"	Chilled Water From Drywell Coolers	8"	Non-NSSS
P-9	148'-0"	RWCU Supply	6"	Non-NSSS
P-10	148'-0"	RHR Shutdown CLG to Reactor Headspray	6"	Non-NSSS
P-11	106'-0"	RCIC Turbine Steam Supply	4"	Non-NSSS
P-12	103'-0"	Main Steam Drain	3"	Non-NSSS
P-38A	110'-0"	Chilled Water to Drywell Coolers	8"	Non-NSSS
P-38B	114'-0"	Chilled Water from Drywell Coolers	8"	Non-NSSS

DSER OPEN ITEM 92



210.37

C-08300

HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

TYPICAL DRYWELL PROCESS
PIPING PENETRATIONS

FIGURE 3.8-6 Amendment 6, 07/84

DSER Open Item No. 95 (DSER Section 3.9.3.2)

FATIGUE EVALUATION ON SRV PIPING AND LOCA DOWNCOMERS

The staff requires a fatigue evaluation be performed on the SRV piping and LOCA downcomers in the torus suppression pool.

RESPONSE

The response to this item has been provided in the response to Question 210.42.

QUESTION 210.42 (SECTION 3.9.2)

The safety relief valve discharge piping and downcomers are ASME Class 2 and 3 components. A through-wall leakage crack in these lines resulting from fatigue caused by SRV actuations and small LOCA conditions would allow steam to bypass the pressure suppression pool. This could result in an unacceptable over-pressurization of the containment. We, therefore, require that the applicant perform a fatigue evaluation on these lines in accordance with the ASME Class 1 fatigue rules and include the effects of pressure, moment, and thermal gradient loadings. The staff will require that the results of the fatigue evaluation be documented in the FSAR.

RESPONSE

Fatigue evaluation of the safety relief valve discharge piping is discussed in Section 5-3.4.3 of the HCGS Plant Unique Analysis Report (PUAR). The Mark I Owners' Group prepared and submitted a generic fatigue evaluation report to the NRC on November 30, 1983, which applies to HCGS. The HCGS wetwell SRV piping is adequate for fatigue based on this generic evaluation.

The vent system, which includes the vent lines, vent header and downcomers, is ASME Class MC. A fatigue evaluation has been performed for the vent system. This evaluation was performed in accordance with the ASME Class MC fatigue rules which are identical to the ASME Class 1 fatigue rules. PUAR, Sections 3-2.4.5 and 3-2.5.1 describe the fatigue analysis methods and results.

DSER OPEN ITEM 95

DSER Open Item No. 96 (DSER Section 3.9.3.3)

IE INFORMATION NOTICE 83-80

The staff's review of Section 3.9.3.4 of the applicant's FSAR relates to the methodology used by the applicant in the design of ASME Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. More information regarding the design and construction of ASME Class 1, 2, and 3 component supports is required.

The applicant should address its actions taken in response to IE Information Notice 83-80.

RESPONSE

For the information requested above see response to Question 210.53.

QUESTION 210.53 (SECTION 3.9.3)

Describe what actions have been taken to address the staff concerns regarding stiff pipe clamps as described in IE Information Notice 83-80.

RESPONSE

The applications of stiff pipe clamps on HCGS will be reviewed based on IE Information Notice 83-80. Section III of the ASME B&PV Code does not provide rules for evaluating stresses due to loadings from nonintegral attachments such as clamps; however, clamp-induced stresses will be evaluated by methods consistent with the intent of the Section III of the ASME B&PV Code. The procedure will include the following:

1. Identify the locations of "stiff" clamps installed on ASME Section III Nuclear Class 1 piping systems.

2. Identify the types of clamps, the loads acting on the clamps and the bolt pre-load values used in their installation. In piping stresses due to all loading conditions at the locations of stiff clamps will also be identified and reviewed.

3. Add the primary membrane and bending stresses caused by the snubber load being transmitted to the pipe through the clamp to the stresses caused by internal pressure and bending computed by equation 9 of NB-3652. Clamp-induced stresses caused by the constraint of the expansion of the pipe due to the internal pressure will be added to other secondary and peak stresses by calculating the effective increases in the C_1 and K_1 stress indices in accordance with NB-3681. Clamp induced stresses due to differential-temperature and differential-thermal-expansion coefficients will be accounted for by computing the effective C_3 and K_3 stress indices. Clamp-induced stresses on elbows caused by the constraint of pipe wall ovalization will be accounted for by computing the effective increases in C_2 and K_2 bending indices. The fatigue usage from clamp-induced plus other stresses will be calculated at governing locations.

Although bolt preloads are not addressed under the ASME B&PV Code rules for piping, bolt preloads could result in damage to a pipe if a clamp were poorly designed. Calculations will be made to ensure that bolt preloads could not result in plastic deformation of the pipe walls.

RESPONSE (Cont'd)

A brief summary of the criteria used and the results of the analysis will be submitted in August, 1984.

HCGS clamps were not used to meet stiffness criteria. They were designed to meet the requirements for strength and load distribution using a minimum of space.

The clamp design utilizes a double nut arrangement to prevent the nuts from backing off. The low temperature (<600°F) and stresses in the bolt from preloads will not cause a relaxation of the material. Consequently, no lift off from the piping will occur.

DSER Open Item No. 98 (DSER Section 3.9.3.3)

DESIGN OF BOLTS

The staff's review of Section 3.9.3.4 of the applicant's FSAR relates to the methodology used by the applicant in the design of ASME Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. More information regarding the design and construction of ASME Class 1, 2, and 3 component supports is required.

The staff needs further information on the design of bolts.

RESPONSE

For the information requested above see response to Question 210.47.

QUESTION 210.47 (SECTION 3.9.3)

The staff's review of your component support design finds that additional information is required regarding the design basis used for bolts.

a. Describe the allowable stress limits used for bolts in equipment anchorage, component supports, and flanged connections.

b. Provide a discussion of the design methods used for concrete expansion anchor bolts used in component supports.

RESPONSE

Non-NSSS:

a. The allowable stress limit used for bolting in equipment anchorage and in pipe support components is $0.5S_u$ but shall not exceed $0.9S_y$ under all service levels.

   For flanged connections, the bolt allowable stress used in the piping analysis are ASME Subsection III, 1979 Summer Addenda, Subsections NB, NC and ND for Class 1, 2 and 3, respectively.

b. The capacities of concrete expansion anchors are based on actual testing of anchors to failure. The failure loads are divided by the factor of safety (typically 4 in accordance with NRC Bulletin 79-02) to establish the allowable design loads. Baseplate flexibility is considered in the design of concrete expansion anchor bolts in accordance with IE Bulletin 79-02.

NSSS:

a. 1. Equipment Anchorage

   Equipment anchorage is not in the NSSS scope.

   2. Component Support Bolting

   The following bolting design limits are typical of components mounted directly on base plates.

   - RWCU Pump

     The support bolting of this pump, which is not safety-related, is designed for the effects of pipe load and SSE load to the requirements of the ASME B&PV Code, Section III, Appendix XVII. The stress limits of 0.41 Sy for tension and 0.15 Sy for shear are used.

RESPONSE (Cont'd)

- RCIC/SLC Pumps and RCIC Turbine

The equipment-to-base-plate bolting satisfies the following design criteria: For normal and upset conditions, 1.0 S is used for primary membrane (or tension), and 1.5 S for primary membrane plus bending (if applicable), where S is the allowed stress limits from the ASME B&PV Code, Section III, Appendix I, Table I-7.3. For emergency and faulted conditions, stresses shall be less than 1.2 times the allowed limits for normal and upset conditions.

The allowed stress limits used for bolting in pipe supports and pipe-mounted equipment supports are as per ASME B&PV Code, Section III, Subsection NF.

For service level A and B, the bolts meet the criteria of Paragraph NF-3280. For service level C and D, Article 2460 of Appendix XVII, with the factors indicated in Article 2110 of Appendix XVII, is the applicable design requirements for bolting. The stresses calculated under these criteria do not exceed the specified minimum yield stresses at temperature.

3. Flanged Connections

Flanged connections are not in the NSSS scope.

b. Expansion Anchor

Expansion anchors are not in the NSSS scope.

DSER Open Item No. 99 (DSER Section 3.9.5)

STRESS CATEGORIES AND LIMITS FOR CORE SUPPORT STRUCTURES

Further information has been requested on the stress categories and limits for core support structures and the applicable codes used for evaluation of the faulted condition. This is an open item.

RESPONSE

For the information requested above, see the responses to Questions 210.55 and 210.48.

QUESTION 210.48 (SECTION 3.9.3)

Describe those short-term and long-term actions being takon to preclude the occurrence of cracking in jet pump hold down beams as described in IE Bulletin 80-07.

RESPONSE

The preload on the hold-down beams will be reduced from 30 to 25 kips in accordance with General Electric recommendations. This will increase the expected life of the beams to 19-40 years. The need for inservice inspection will be based on a lead-plant experience and GE testing, and will be conducted such that any crack initiation will be detected prior to beam failure.

Section 3.9.5.1.2.1 has been revised to include this information.

3.9.5.1.1.6 Control Rod Guide Tubes

The control rod guide tubes, located inside the vessel, extend from the top of the CRD housings up through holes in the core plate. Each tube is designed as the guide for a control rod, as well as being the vertical support for a four-lobed orificed fuel support and the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing, which in turn transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. A thermal sleeve is inserted into the CRD housing from below and is rotated to lock the control rod guide tube in place. A key is inserted into a locking slot in the bottom of the CRD housing to hold the thermal sleeve in position.

3.9.5.1.2 Reactor Internals

The reactor internals consist of those items listed in Section 3.9.5.1.b. Those that involve coolant flow paths are described in the following paragraphs.

3.9.5.1.2.1 Jet Pump Assemblies

The jet pump assemblies are located in two semicircular groups in the downcomer annulus between the core shroud and the RPV wall. The design and performance of the jet pumps are discussed in References 3.9-14 and 3.9-15. Each stainless steel jet pump consists of a driving nozzle, a suction inlet, a throat or mixing section, and a diffuser, as shown on Figure 3.9-5. The driving nozzle, suction inlet, and throat are joined together as a removable unit, and the diffuser is permanently installed. High-pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle thermal sleeve. A riser brace consists of cantilever beams welded to a riser pipe and to pads on the RPV wall.

The nozzle entry section is connected to the riser by a metal-to-metal, spherical-to-conical seal joint. Firm contact is maintained by a holddown clamp. The throat section is supported laterally by a bracket attached to the riser. There is a slip-fit joint between the throat and diffuser. The diffuser is a gradual conical section changing to a straight cylindrical section at the lower end.

INSERT I →

INSERT I

The preload on the hold-down beams will be reduced from 30 to 25 kips in accordance with General Electric recommendations. This will increase the expected life of the beams to 19-40 years. The need for inservice inspection will be based on a lead-plant experience and GE testing, and will be conducted such that any initiation will be detected prior to beam failue.

QUESTION 210.55 (SECTION 3.9.5)

Verify that the design and analysis of your reactor internals is equivalent to Subsection NG.

RESPONSE

Hope Creek reactor internals were designed and procured prior to the issuance of Subsection NG of Section III of the ASME B&PV Code. However, design criteria in an earlier draft of the ASME B&PV Code was used as a guide in the design of the reactor internals. These criteria are presented in Section 3.9.5.3.3 and were used in lieu of Subsection NG. Subsequent to the issuance of Subsection NG, comparisons were made to assure that the pre-NG design meets the equivalent level of safety as presented by Subsection NG.

DSER Open Item No. 125 (DSER Section 6.2.1.5.2)

DESIGN DRYWELL HEAD DIFFERENTIAL PRESSURE

The applicant has performed a pressure response analysis for the head region postulating the rupture of the 6-inch head spray line.

The initial conditions of 15.45 psia, 135°F, and 30 percent relative humidity were used in the analysis. The maximum differential pressure between the head region and the drywell was calculated to be 15.64 psi. However, the applicant has not provided the design drywell head differential pressure. The staff requires this information and will perform a confirmatory analysis to verify the applicant's results. We will report on this matter in a supplement to this SER.

RESPONSE

Section 6B.3 has been revised to provide the information requested above.

6B.3 DRYWELL HEAD REGION SUBCOMPARTMENT ANALYSIS

~~The design basis pressure differential between the drywell head and containment region is a structural requirement of the drywell head~~. A pressure analysis of the drywell head region for a postulated head spray line break was performed. The effects of a 6-inch residual heat removal (RHR) head spray line break bound those of a 2-inch RPV head vent line, which is the only other line that runs through the drywell head region.

Figure 6B-12 illustrates the basic arrangement of the head region. Venting from the head region is accomplished through ventilation openings as shown on Figure 6B-12. These vent openings provide a total vent area of 7.722 square feet, with an equivalent orifice discharge coefficient of 0.75 to relieve pressure buildup caused by the postulated break.

Figure 6B-13 is the schematic flow diagram with vent flow areas and discharge coefficients used in the drywell head venting analysis.

To determine the peak pressure in the drywell head, all insulation was assumed to remain in place. Initial conditions of 15.45 psia, 135°F, and 30% relative humidity were used in this analysis.

The pressure transient of this analysis is presented on Figure 6B-14. It can be seen that the maximum pressure in the drywell head region is 31.086 psia and occurs 1.7534 seconds after the head spray line break. Considering the containment pressure to be atmospheric (no drywell air displaced into the suppression chamber), a drywell head to containment pressure differential of 15.64 psid occurs at approximately 2.0 seconds after the break.

INSERT →

6B.4 REFERENCES

6B-1 "Subcompartment Pressure Analyses," BN-TOP-4, Revision 1, November 1977, Bechtel Power Corporation, San Francisco, California.

INSERT

The manufacturer's design differential pressure (between the drywell head and containment region) for the water seal plate is conservatively defined as 20.0 psid. For the refueling bellows, the manufacturer's design differential pressure is defined as 15.0 psid. However, an evaluation of the bellows manufacturer's design report revealed that sufficient margin exists between the calculated stresses and the allowable stress such that the bellows can easily withstand the increased stress due to an applied 15.64 psid.

DSER Open Item No. 132 (DSER Section 6.2.4)

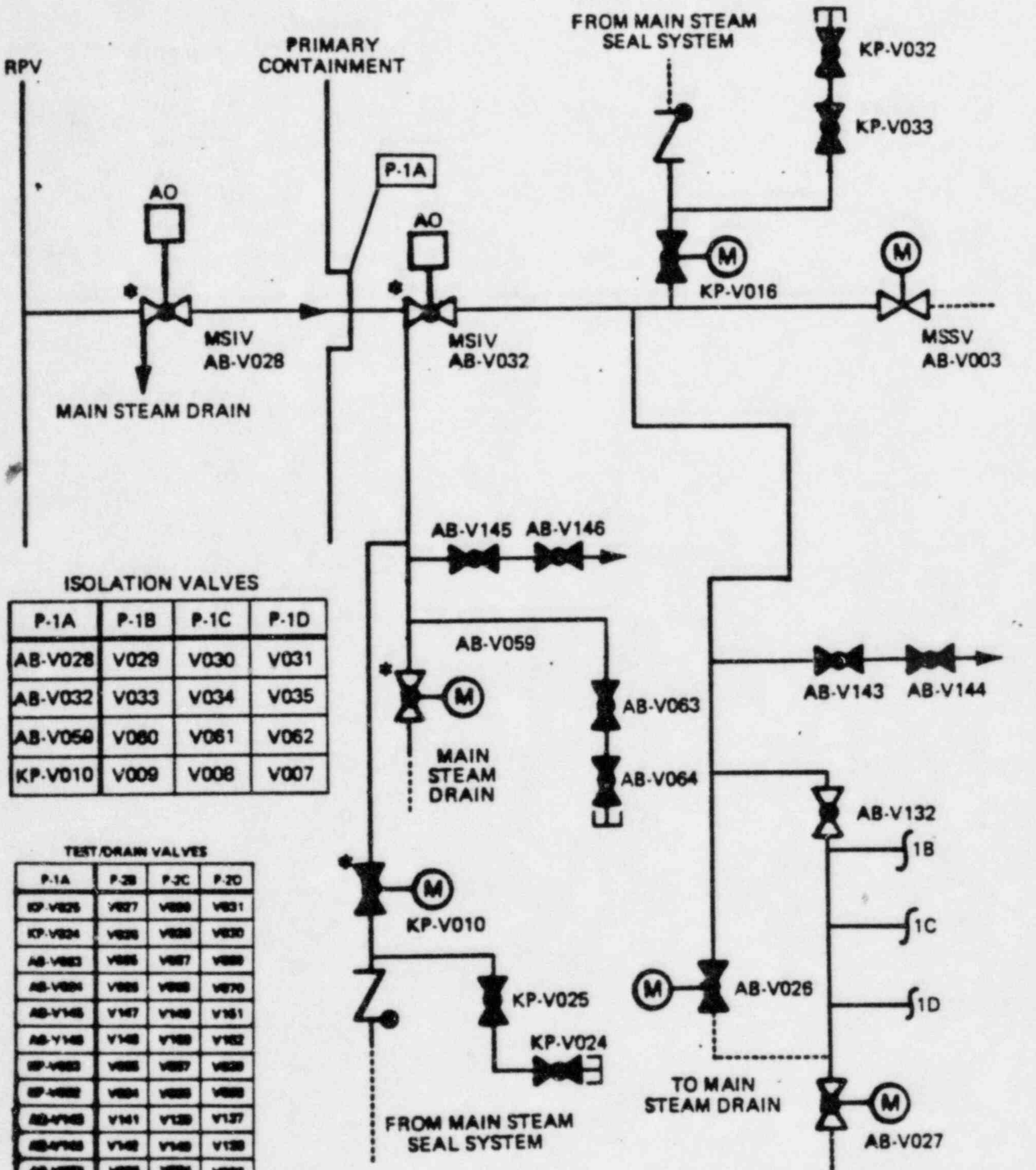
CONTAINMENT ISOLATION REVIEW

The basis for the staff's acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in general design criteria and applicable regulatory guides, staff technical positions, the acceptance criteria of NUREG-0800, and industry codes and standards.

[However, the applicant has not provided the staff with the information requested to conduct the containment isolation review. We will require the applicant to provide for the containment isolation figures contained in the FSAR proper valve identification and the location of the test, vent, and drain lines on these figures. We will also require large enough P&ID drawings in order to conduct our review.] We will report on this matter in a supplement to this SER.

RESPONSE

The penetration details in Figure 6.2-28 have been revised to show proper valve identification and the available test provisions.



ISOLATION VALVES

P-1A	P-1B	P-1C	P-1D
AB-V028	V029	V030	V031
AB-V032	V033	V034	V035
AB-V059	V060	V061	V062
KP-V010	V009	V008	V007

TEST/DRAIN VALVES

P-1A	P-2B	P-2C	P-2D
KP-V025	V027	[illegible]	V031
KP-V024	[illegible]	[illegible]	V030
AB-V063	[illegible]	V067	[illegible]
AB-V064	[illegible]	[illegible]	V070
AB-V145	V147	[illegible]	V151
AB-V146	V148	[illegible]	V152
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	V141	[illegible]	V137
[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]
AB-V132	V131	V130	V129

EXTENDED ISOLATION VALVES

P-1A	P-2B	P-2C	P-2D
AB-V003	V004	V005	V006
KP-V016	V015	V014	V013
AB-V027	-	-	-

DETAIL 1

*(SEE LEGEND)

DSER OPEN ITEM /32

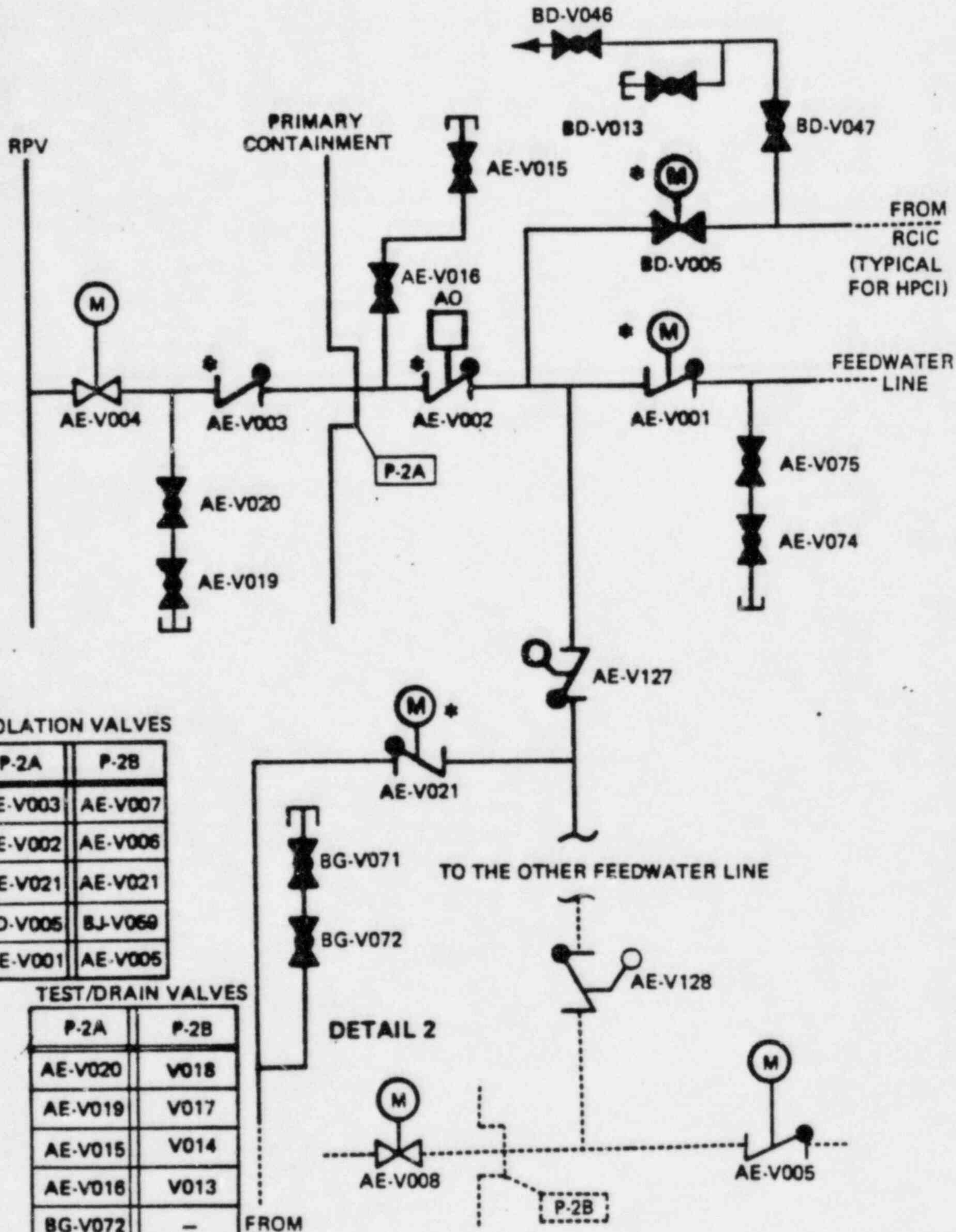
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

MAIN STEAM LINES

FIGURE 6.2-28
SHEET 1 OF 48

AMENDMENT 6, 06/8



ISOLATION VALVES

P-2A	P-2B
AE-V003	AE-V007
AE-V002	AE-V006
AE-V021	AE-V021
BD-V005	BJ-V069
AE-V001	AE-V005

TEST/DRAIN VALVES

P-2A	P-2B
AE-V020	V018
AE-V019	V017
AE-V015	V014
AE-V016	V013
BG-V072	–
BG-V071	–
AE-V075	–
AE-V074	–
BD-V047	BJ-V018
BD-V046	BJ-V017
AE-V004	AE-V008
AE-V127	AE-V128

* (SEE LEGEND)

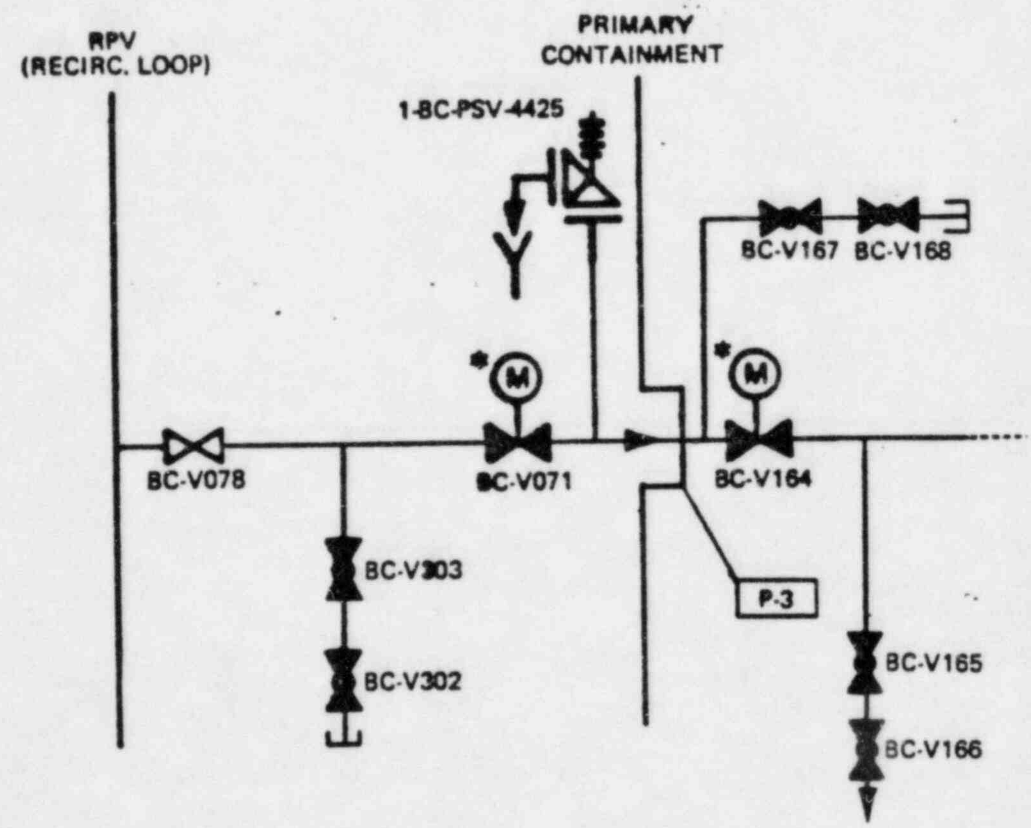
DSER OPEN ITEM /32

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

FEEDWATER LINES

FIGURE 6.2-28
SHEET 2 OF 48

AMENDMENT 6 06/



DETAIL 3

DSER OPEN ITEM 132

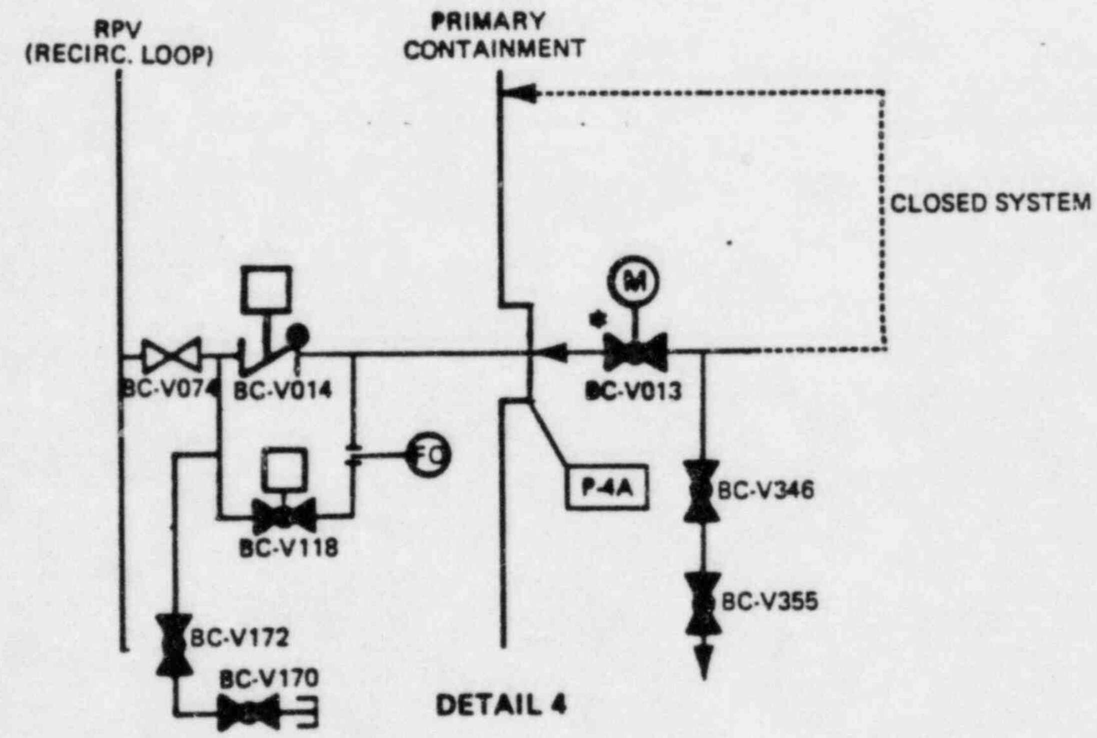
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RHR SHUTDOWN COOLING SUCTION LI

FIGURE 6.2-28
SHEET 3 OF 48

AMENDMENT 6, 0

*(SEE LEGEND)



DETAIL 4

ISOLATION VALVES

P-4A	P-4B
BC-V013	BC-V110

TEST/DRAIN VALVES

P-4A	P-4B
BC 172	V169
BC-170	V171
BC-V346	V334
BC-V355	V335
BC-V074	V183
BC-V118	V117

OTHER VALVES

P-4A	P-4B
BC-V014	V111

DSER OPEN ITEM 132

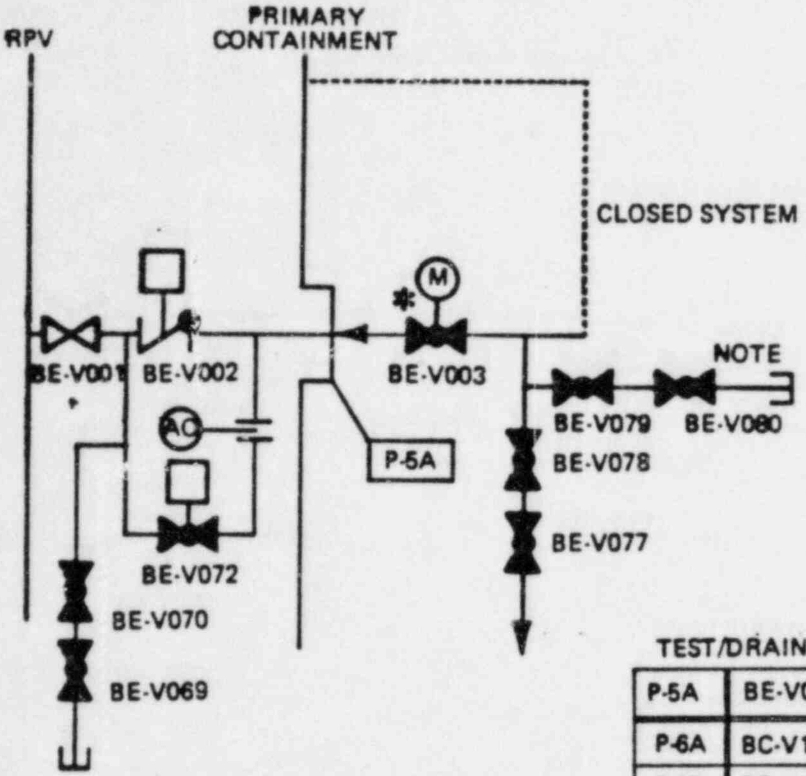
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RHR SHUTDOWN
COOLING RETURN LINES

FIGURE 6.2-28
SHEET 4 OF 48

AMENDMENT 6, 06/



OTHER VALVES

P-5A	BE-V002
P-6A	BC-V005
P-6B	BC-V017
P-6C	BC-V114
P-6D	BC-V102

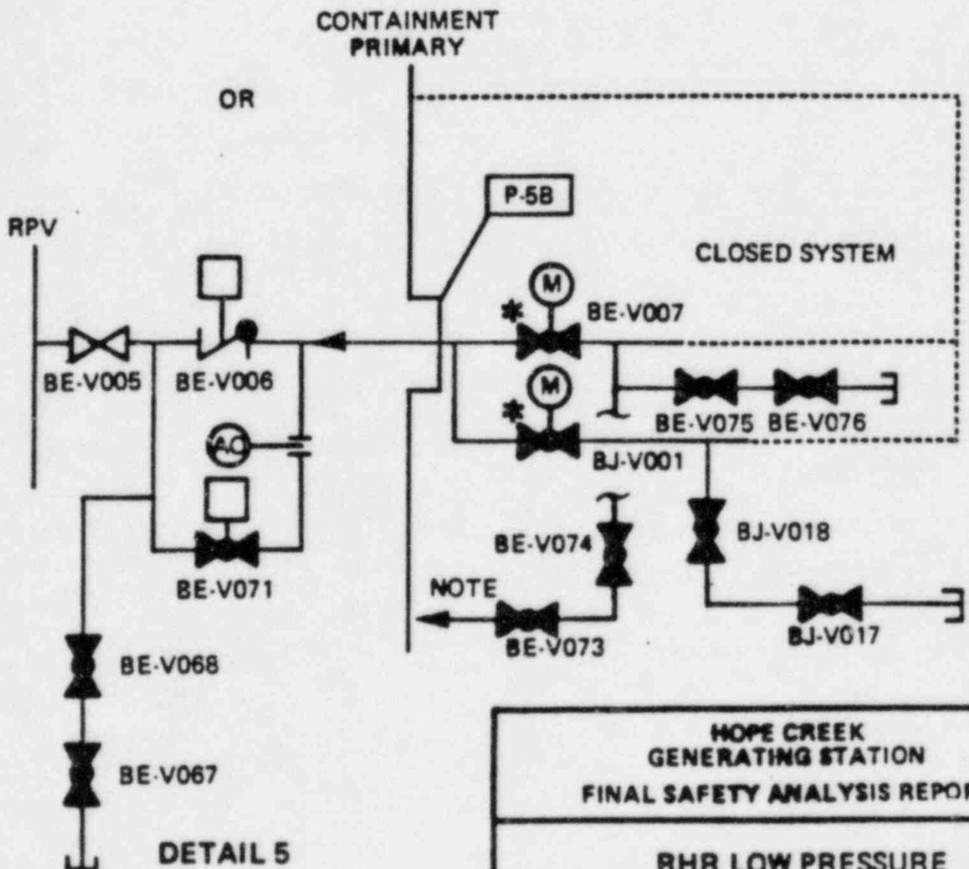
TEST/DRAIN VALVES

P-5A	BE-V070	V069	V078	V077	V072	V001
P-6A	BC-V173	V177	V320	V321	V119	V076
P-6B	BC-V174	V178	V324	V325	V120	V075
P-6C	BC-V175	V179	V353	V354	V121	V182
P-6D	BC-V176	V180	V332	V333	V122	V181

ISOLATION VALVES

P-5A	BE-V003
P-6A	BC-V004
P-6B	BC-V016
P-6C	BC-V113
P-6D	BC-V101

OR



DETAIL 5

DSER OPEN ITEM 132

NOTE:
FOUND ON CORE SPRAY SYSTEM ONLY.

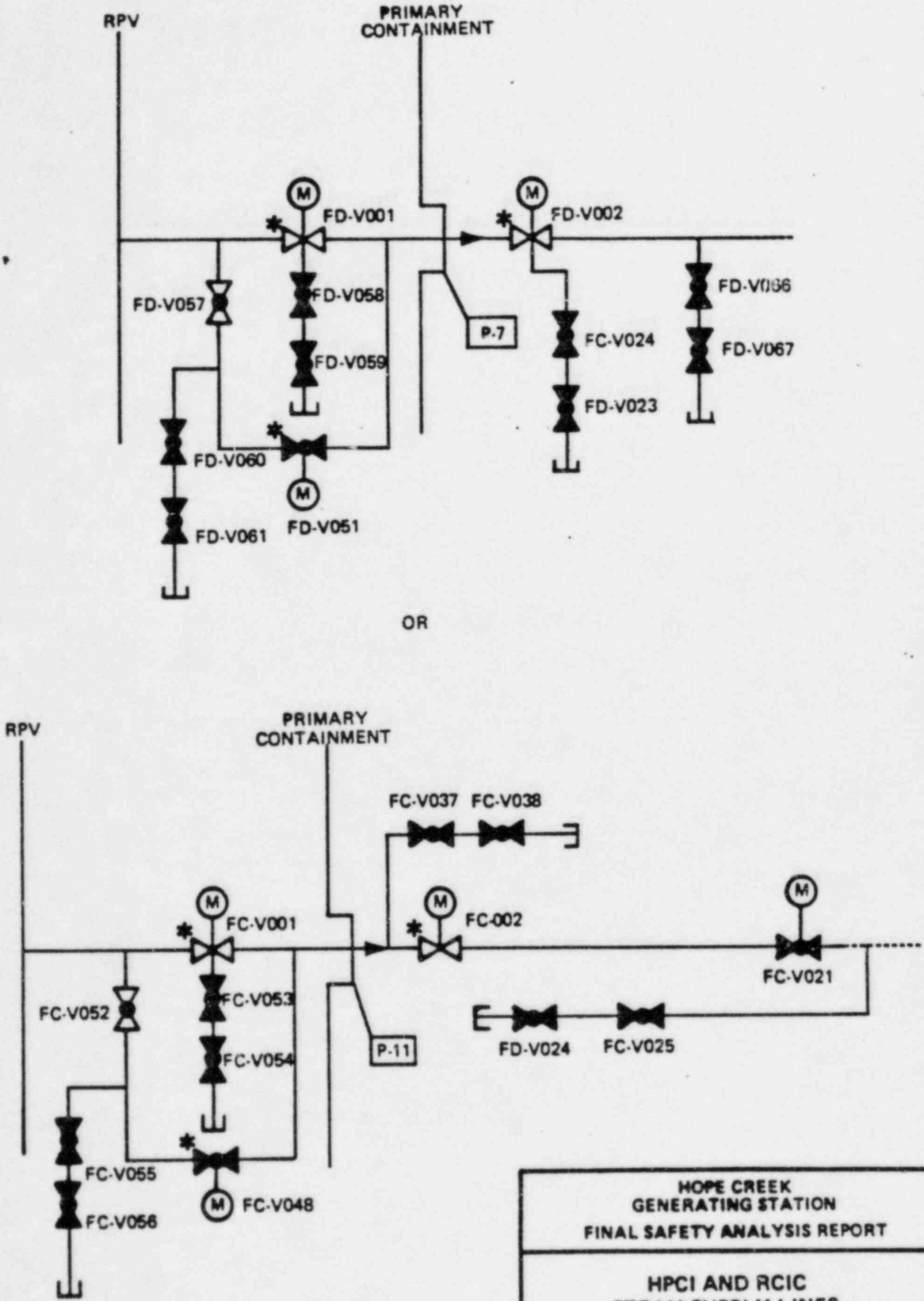
* (SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RHR LOW PRESSURE COOLANT INJECTION AND CORE SPRAY DISCHRGE LINE

FIGURE 6.2-28
SHEET 5 OF 48

AMENDMENT 6, 06/84



DETAIL 6

*(SEE LEGEND)

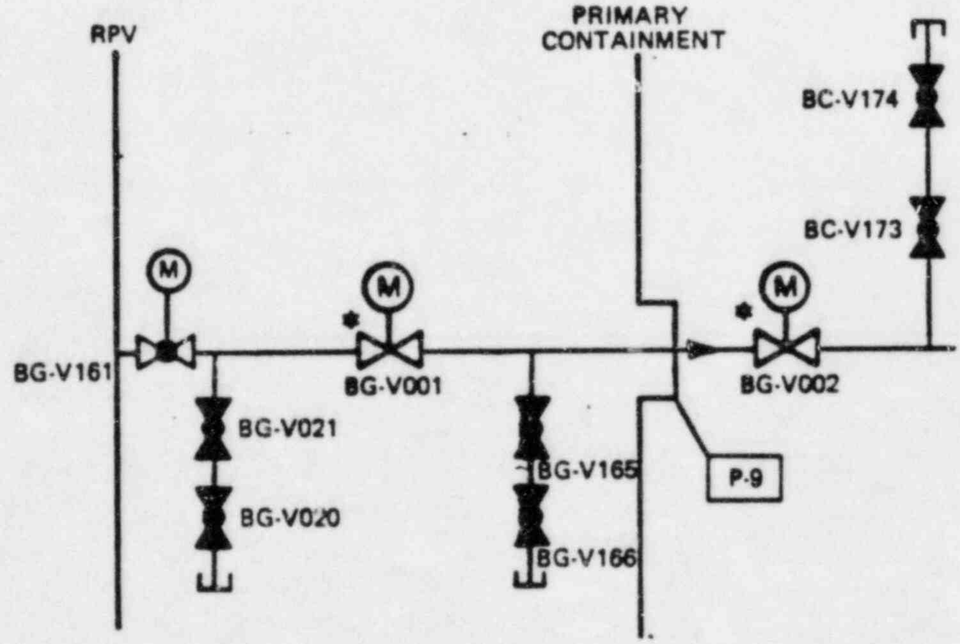
DSER OPEN ITEM /32

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

HPCI AND RCIC STEAM SUPPLY LINES

FIGURE 6.2-28
SHEET 6 OF 48

AMENDMENT 6, 06/84



DETAIL 7

DSER OPEN ITEM 132

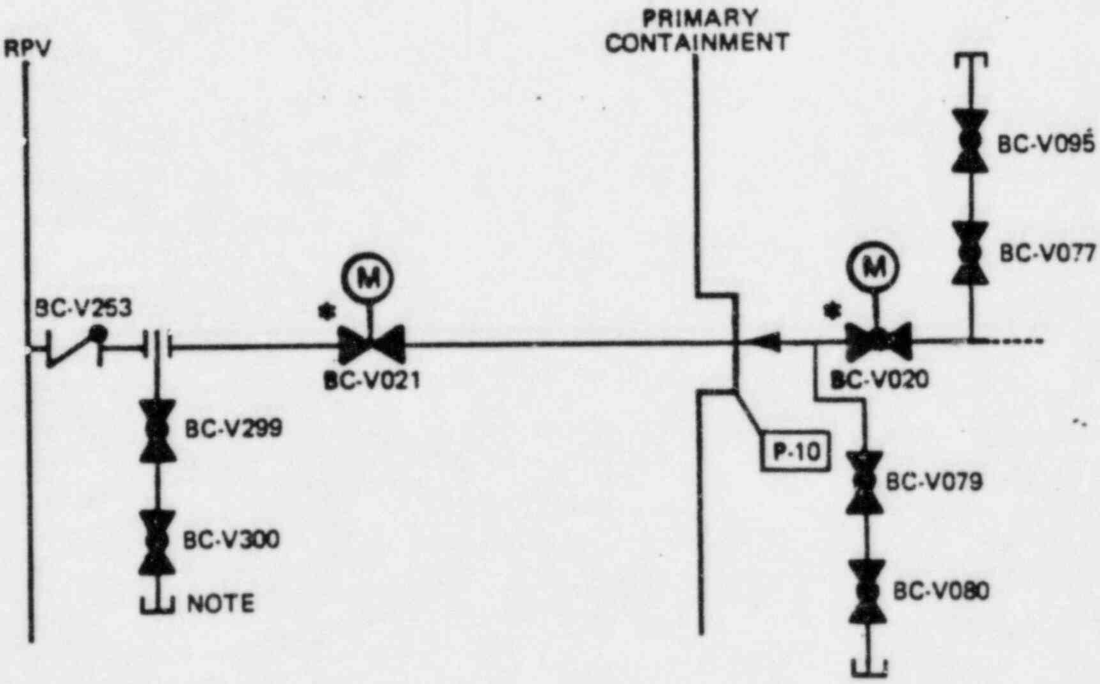
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RWCU LINE

FIGURE 6.2-28
SHEET 7 OF 48

AMENDMENT 6, 06/



DETAIL 8

NOTE: THE TEST LINE IS ATTACHED TO THE REMOVABLE BLIND FLANGE

DSER OPEN ITEM /32

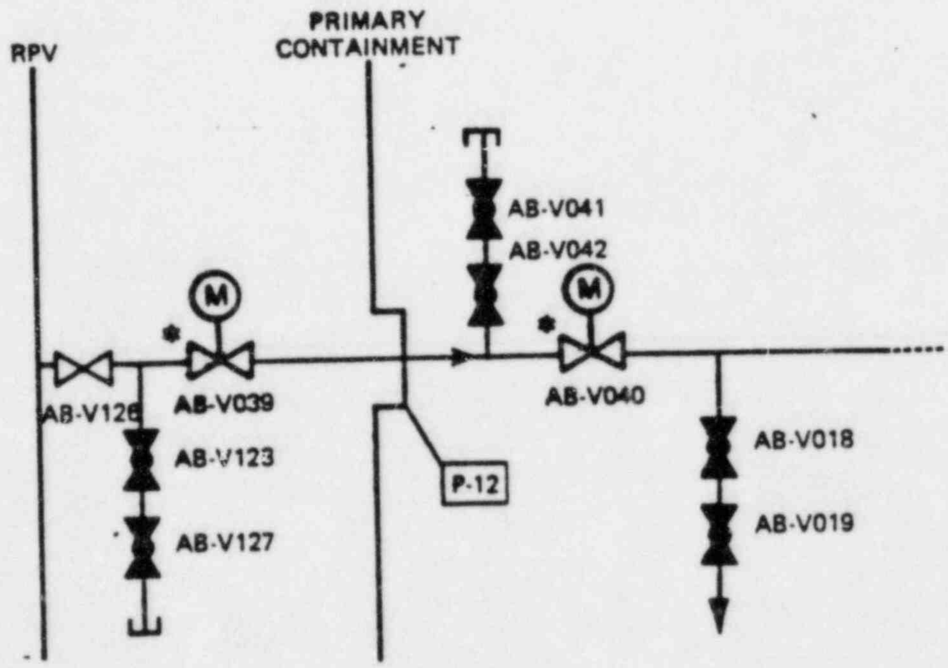
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

HEAD SPRAY LINE

FIGURE 6.2-28
SHEET 8 OF 48

AMENDMENT 6.0



DETAIL 9

DSER OPEN ITEM 132

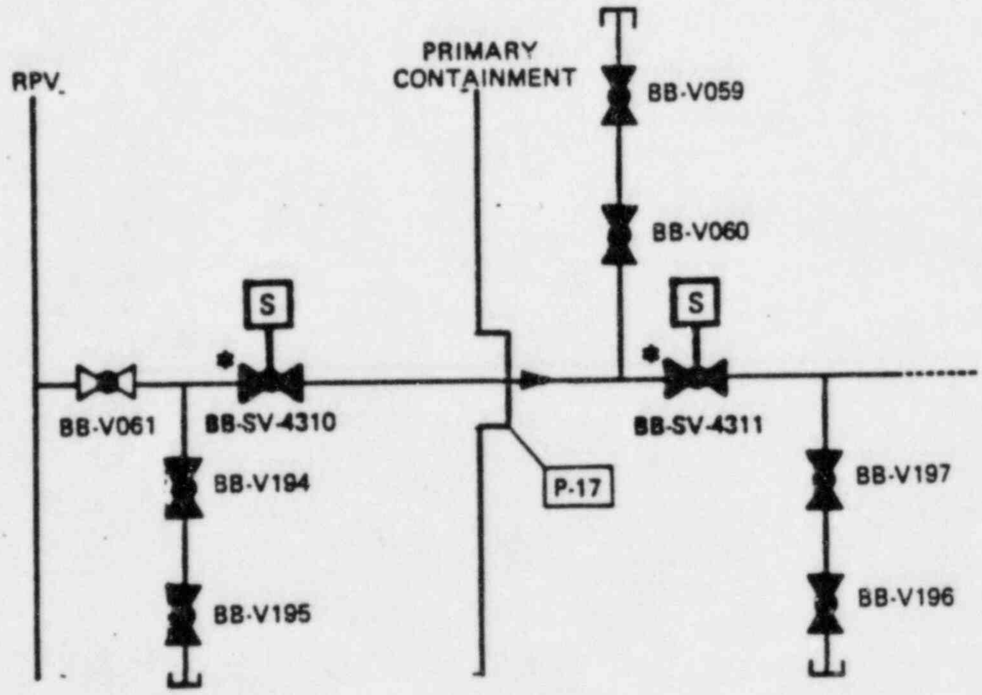
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

MAIN STEAM DRAIN LINE

FIGURE 6.2-28
SHEET 9 OF 48

AMENDMENT 6, C



DETAIL 10

DSER OPEN ITEM 132

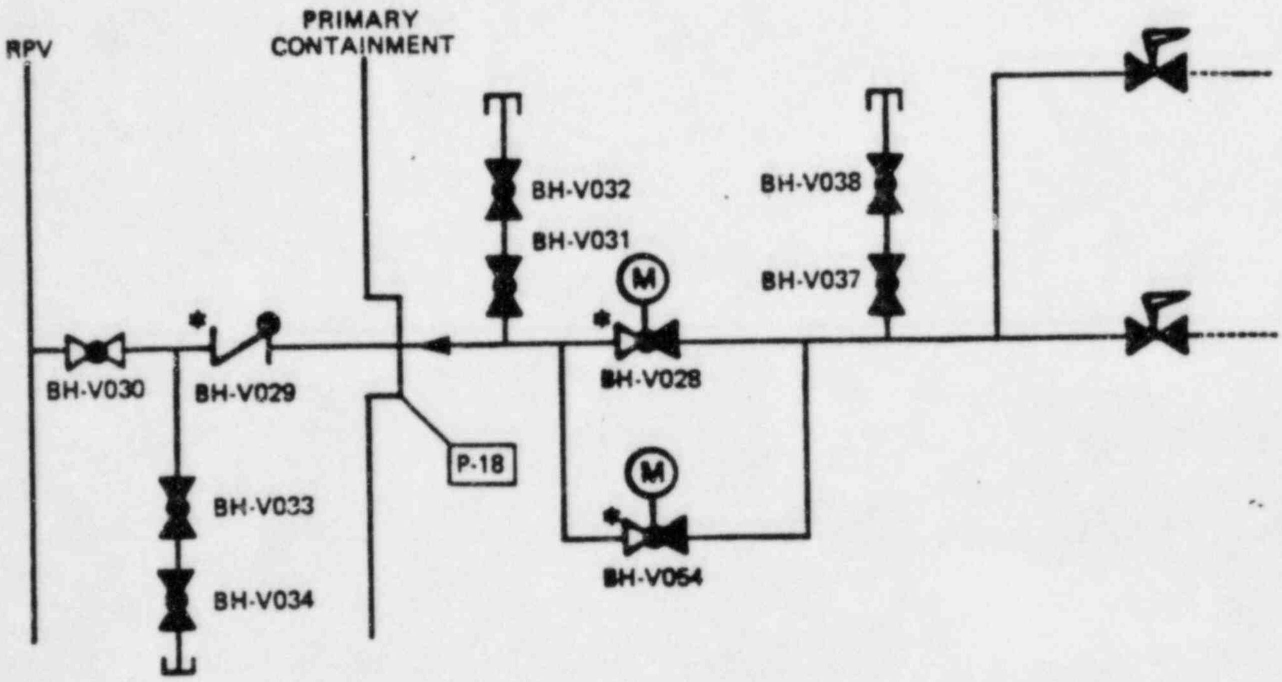
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

REACTOR RECIRCULATION PUMP
WATER SAMPLE LINES

FIGURE 6.2-28
SHEET 10 OF 48

AMENDMENT 6, 06



DETAIL 11

DSER OPEN ITEM 132

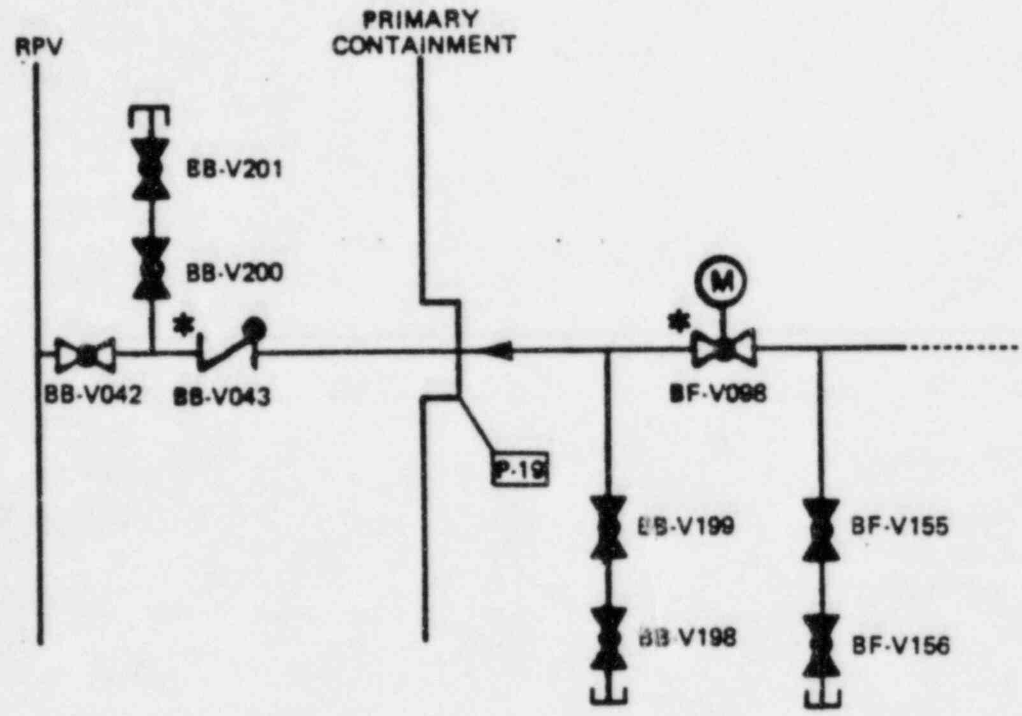
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

STANDBY LIQUID CONTROL LINE

FIGURE 6.2-28
SHEET 11 OF 48

AMENDMENT 6, 06



DETAIL 12

ISOLATION VALVES

P-19	P-20
BB-V043	BB-V047
BF-V098	BF-V099

TEST VALVES

P-19	P-20
BB-V200	V204
BB-V201	V205
BB-V199	V203
BB-V198	V202
BF-V155	V772
BF-V156	V773
BB-V042	V046

DSER OPEN ITEM 132

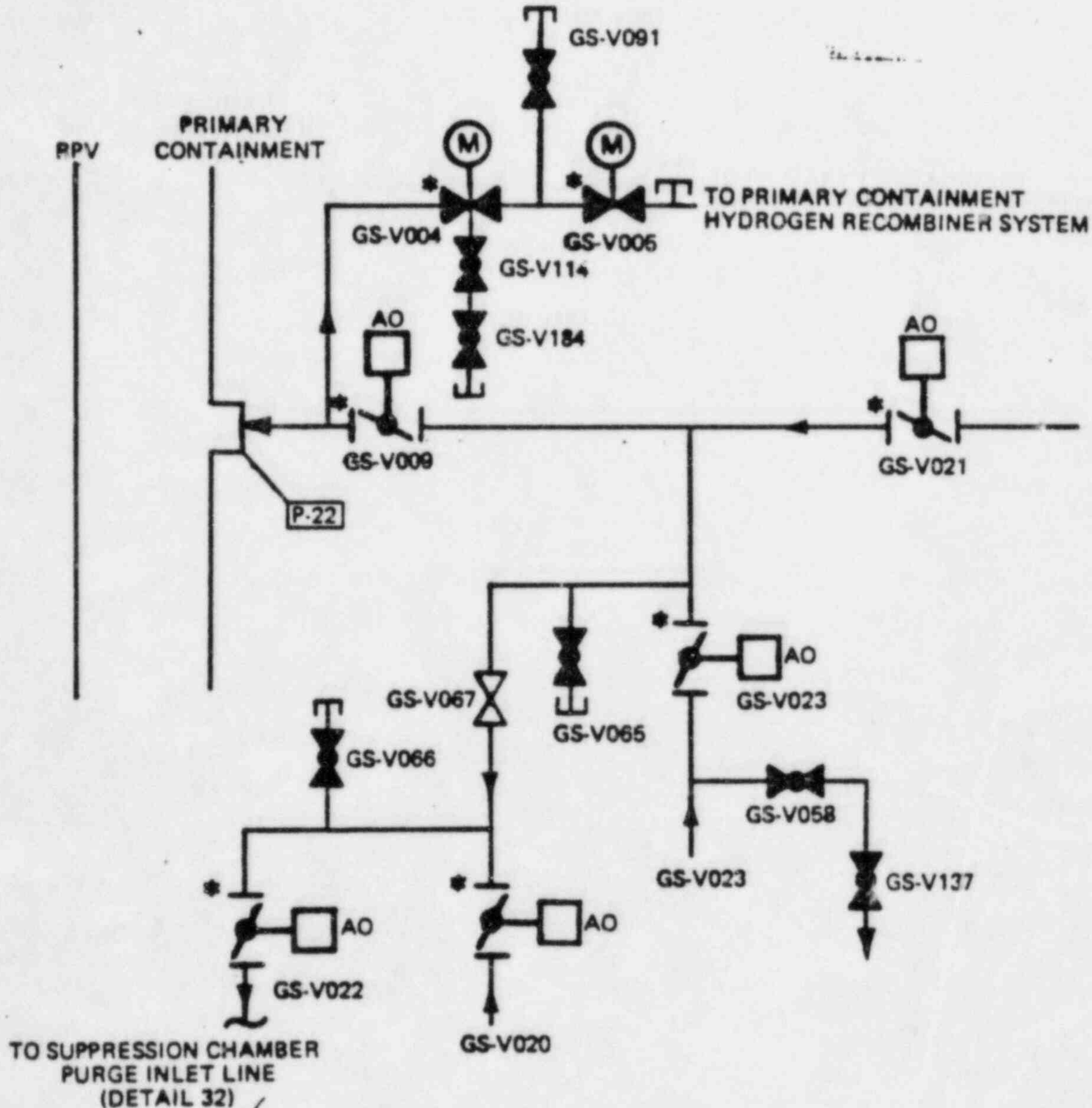
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

REACTOR RECIRCULATION PUMP SEAL WATER LINES

FIGURE 6.2-28
SHEET 12 OF 48

AMENDMENT 6, 0[illegible]



DETAIL 13

CN265

DSER OPEN ITEM 132

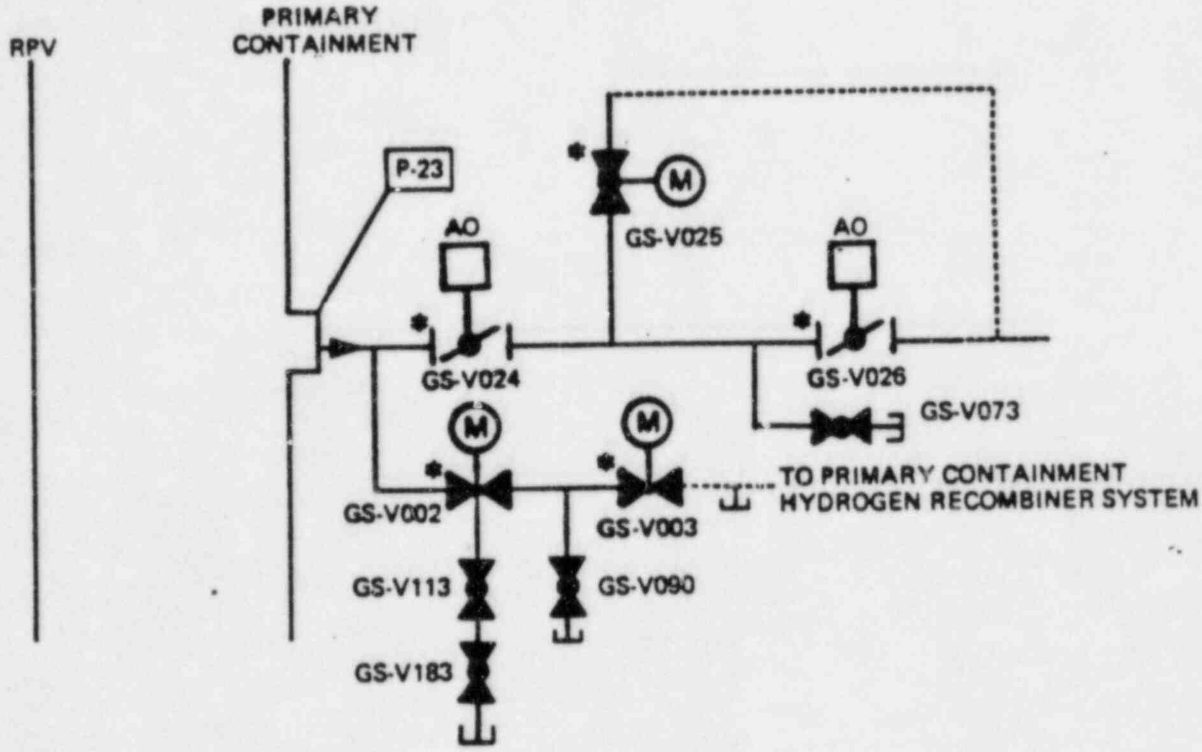
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

DRYWELL PURGE INLET VENT LINE

FIGURE 6.2-28
SHEET 13 OF 48

AMENDMENT 6, 06



DETAIL 14

DSER OPEN ITEM /32

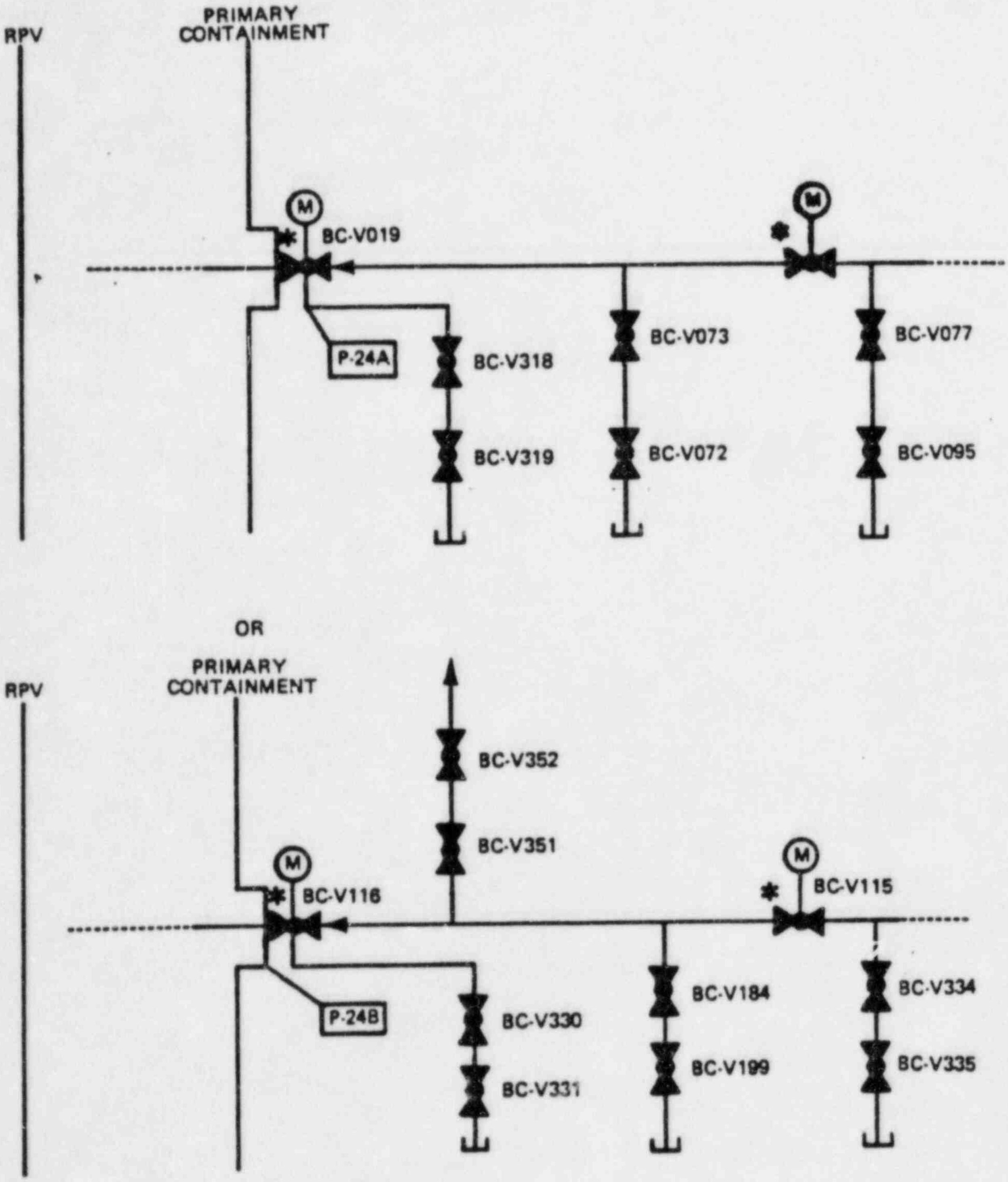
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

DRYWELL PURGE
VENT LINE

FIGURE 6.2-28
SHEET 14 OF 48

AMENDMENT 6, 08



DETAIL 15

DSER OPEN ITEM 132

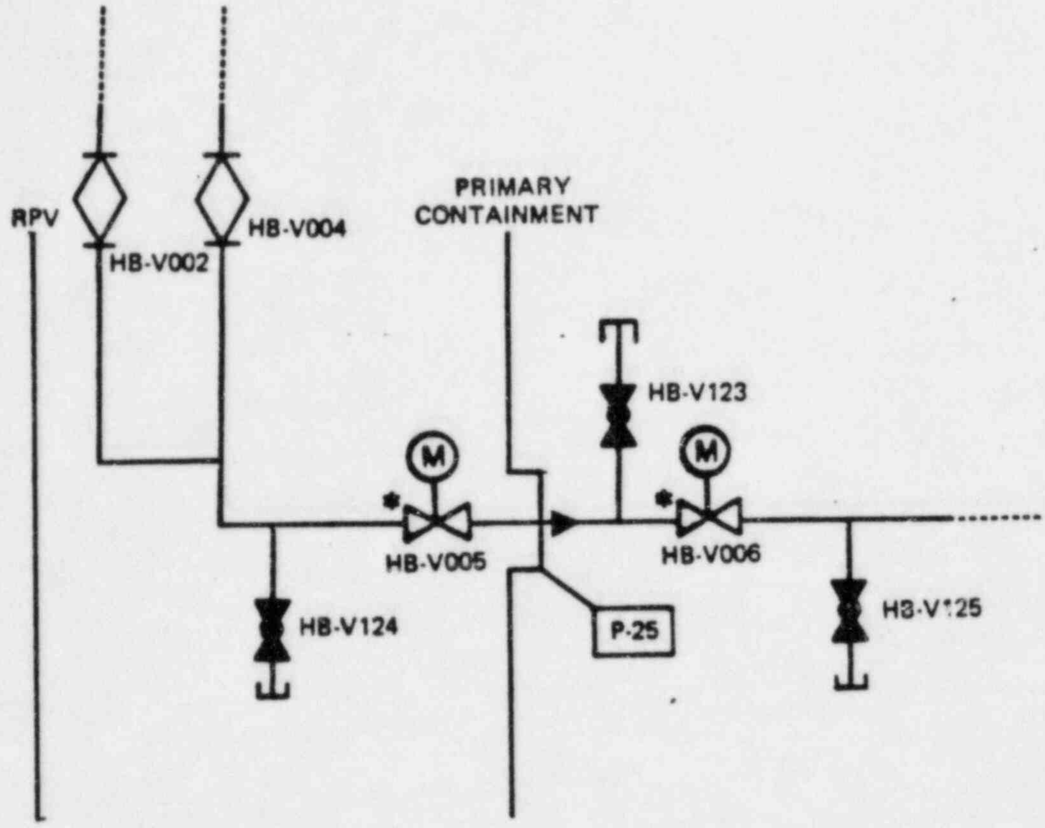
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CONTAINMENT SPRAY LINES

FIGURE 6.2-28
SHEET 15 OF 48

AMENDMENT 6, 06/84



DETAIL 16

ISOLATION VALVES

P-25	P-26
HB-V005	HB-V045
HB-V006	HB-V046

TEST VALVES

P-25	P-26
HB-V002	HB-V042
HB-V004	HB-V044
HB-V124	HB-V127
HB-V123	HB-V126
HB-V125	HB-V128

DSER OPEN ITEM 132

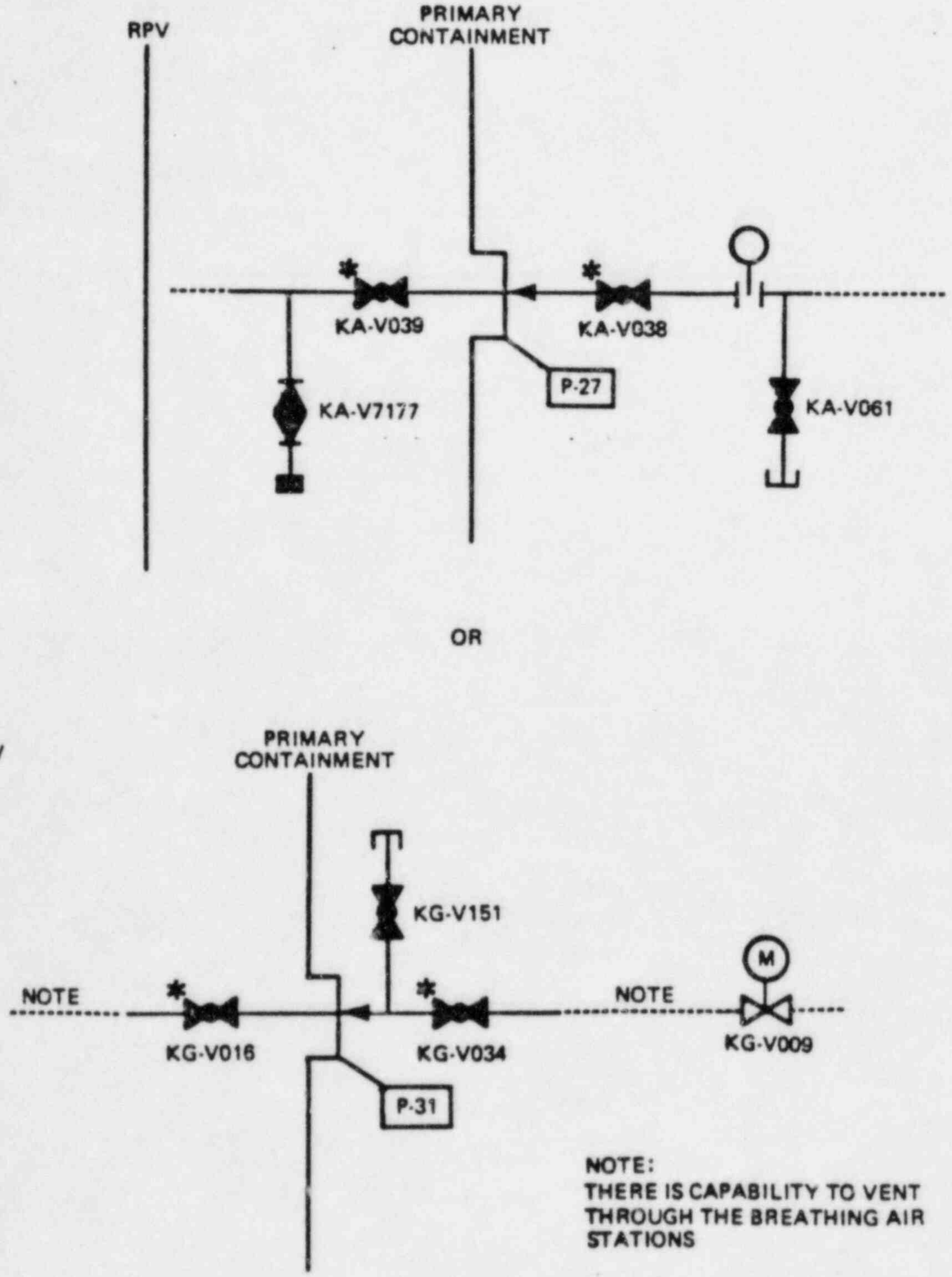
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

LIQUID RADWASTE COLLECTION LINES

FIGURE 6.2-28
SHEET 16 OF 48

AMENDMENT 6, 06



DETAIL 17

DSER OPEN ITEM 132

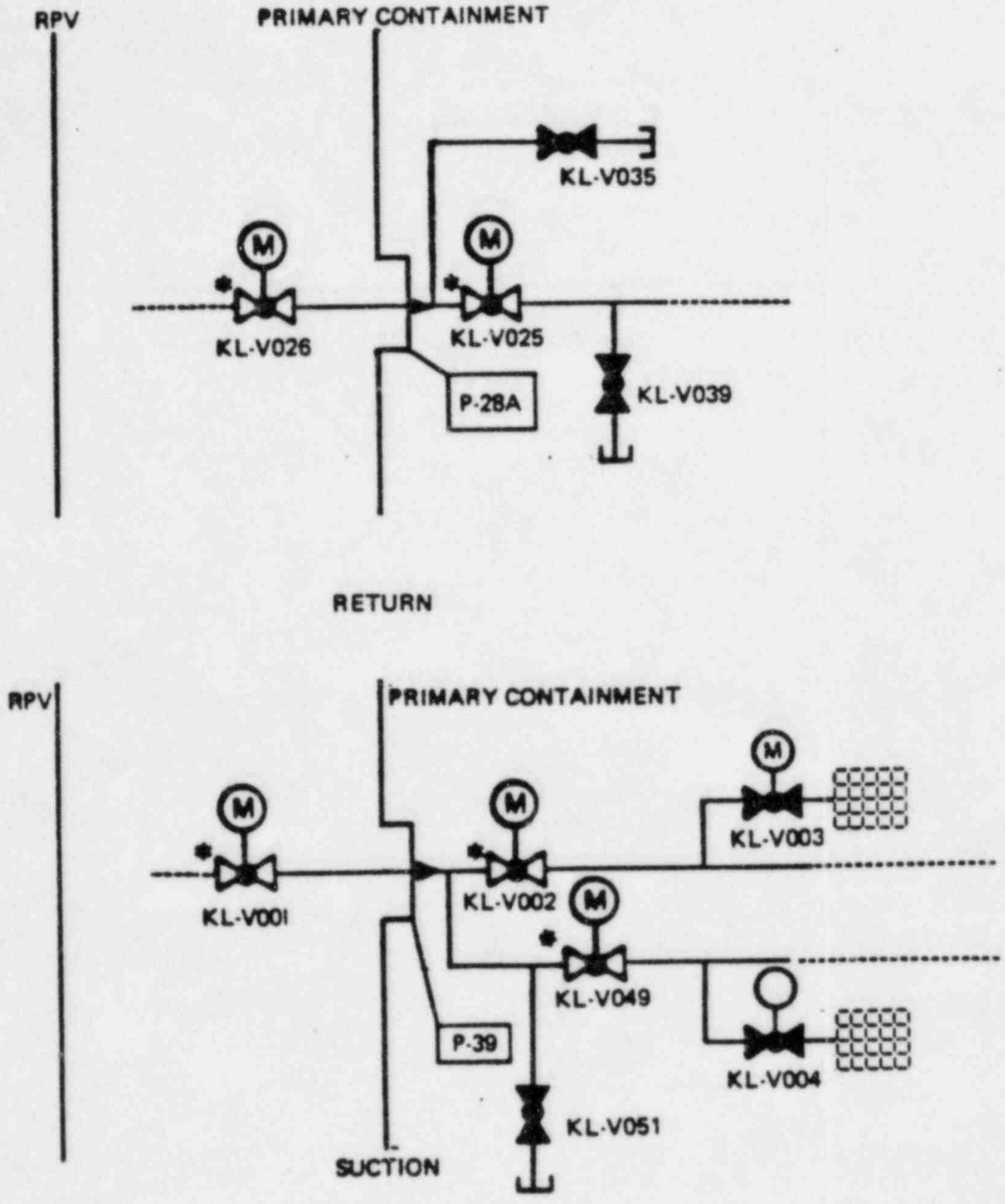
* (SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SERVICE AIR
AND
BREATHING AIR TO DRYWELL

FIGURE 6-2-28
SHEET 17 OF 48

AMENDMENT 6, 06/84



DETAIL 18

ISOLATION VALVES

P-28A	P-28B
KL-V026	KL-V028
KL-V025	KL-V027

TEST VALVES

P-28A	P-28B
KL-V035	KL-V037
KL-V039	KL-V040

*(SEE LEGEND)

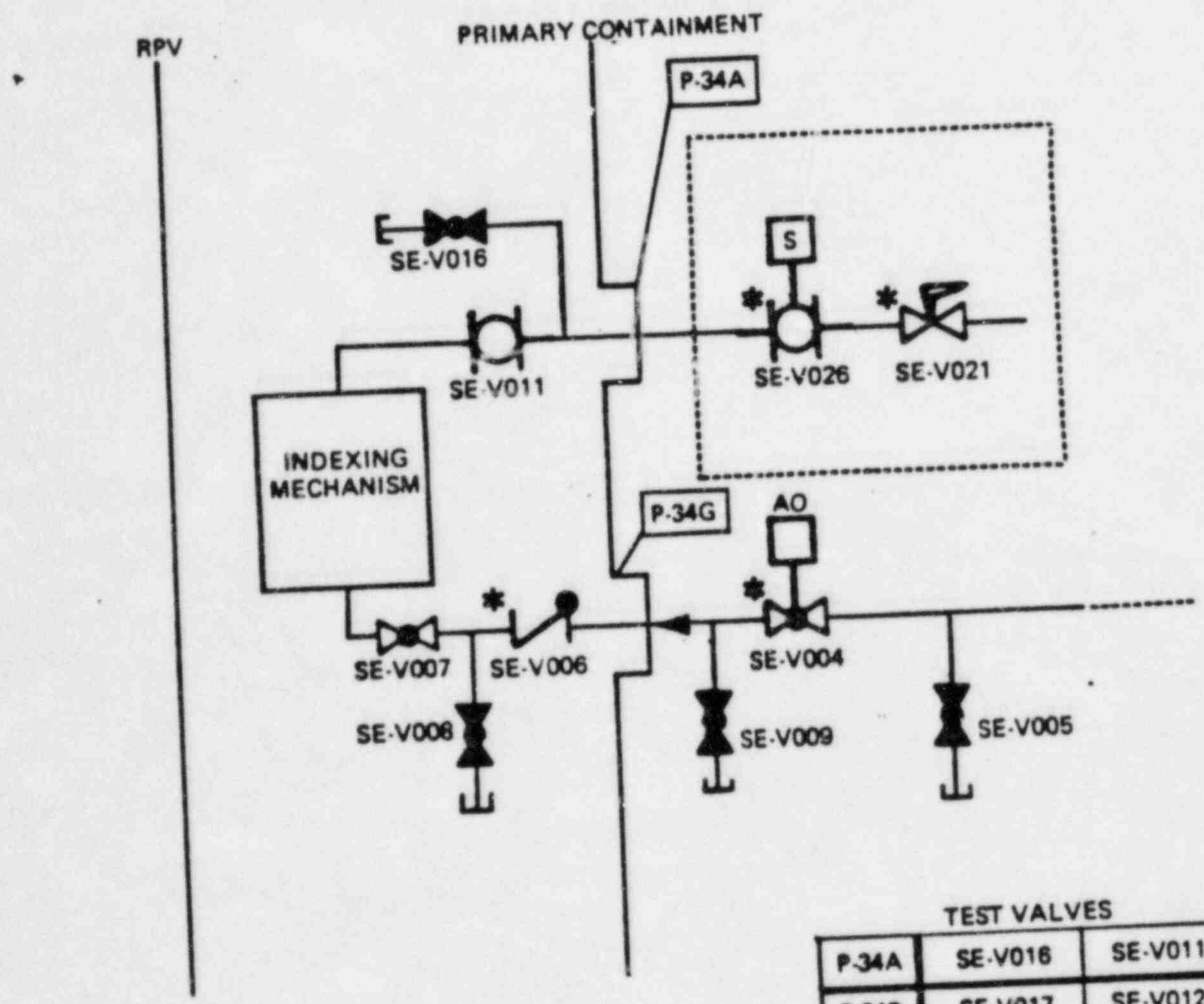
DSER OPEN ITEM / 32

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

INSTRUMENT GAS LINES

FIGURE 6.2-28
SHEET 18 OF 48

AMENDMENT 6, 06/?



DETAIL 19

TEST VALVES

P-34A	SE-V016	SE-V011
P-34B	SE-V017	SE-V012
P-34C	SE-V018	SE-V013
P-34D	SE-V019	SE-V014
P-35E	SE-V020	SE-V015

ISOLATION VALVES

P-34A	SE-V026	SE-V021
P-34B	SE-V027	SE-V022
P-34C	SE-V028	SE-V023
P-34D	SE-V029	SE-V024
P-34E	SE-V030	SE-V025

*(SEE LEGEND)

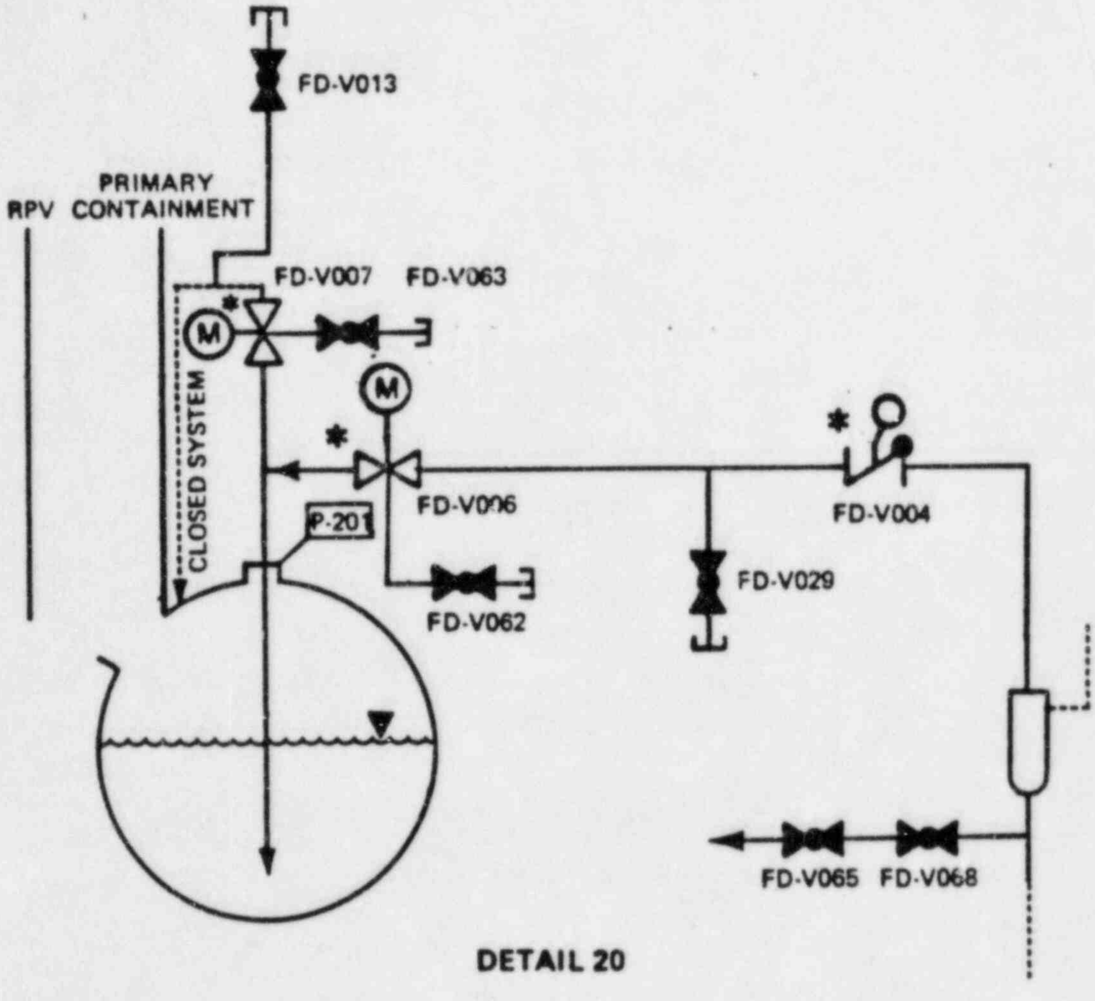
HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

TIP PROBE
GUIDE TUBE LINES

FIGURE 6.2-28
SHEET 19 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM / 32



DETAIL 20

ISOLATION VALVES

P-201	P-207
FD-V006	FC-V005
FD-V007	FC-V006
FD-V004	FC-V003

TEST/DRAIN VALVES

P-201	P-207
FD-V062	FC-V058
FD-V063	FC-V057
FD-V029	FC-V020
FD-V013	FC-V040
FD-V068	FC-V063
FD-V065	FC-V062

DSER OPEN ITEM 132

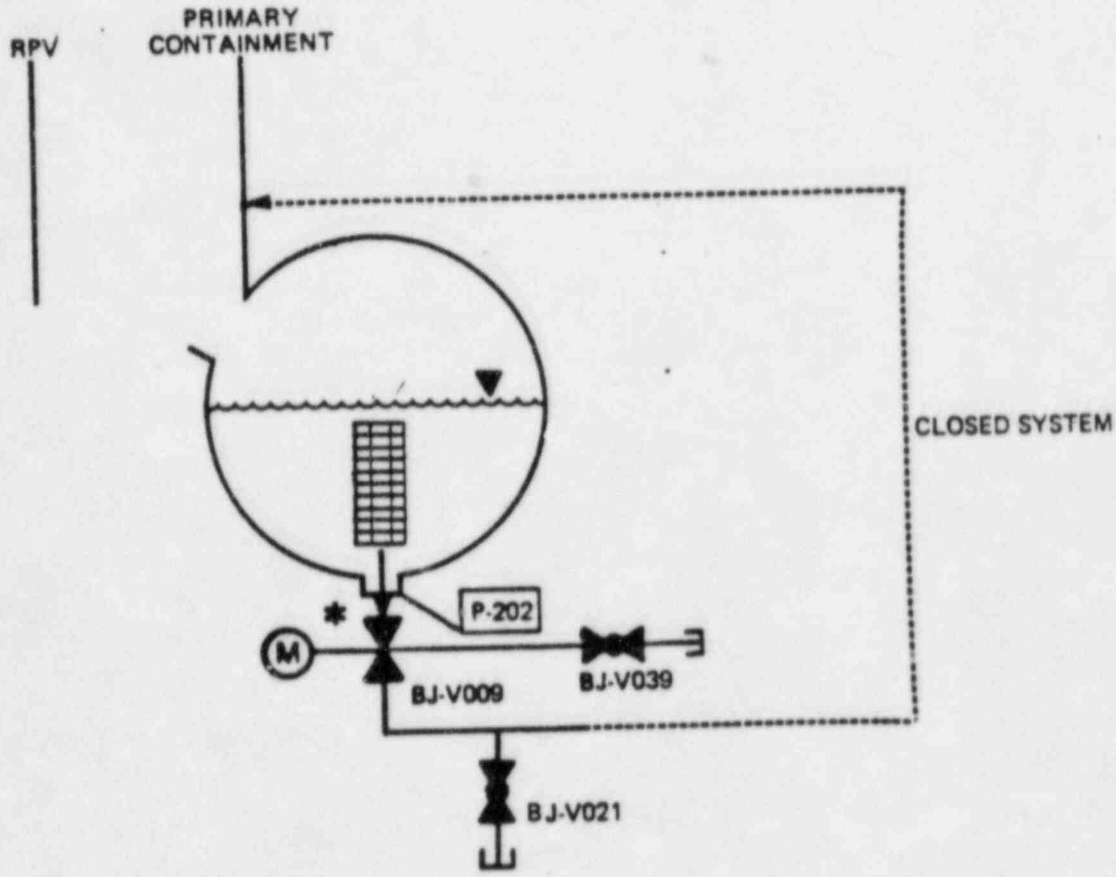
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

HPCI AND RCIC EXHAUST LINES

FIGURE 6.2-28
SHEET 20 OF 48

AMENDMENT 6, 06/84



DETAIL 21

ISOLATION VALVES

P-202	P-208
BJ-V009	BD-V003

TEST VALVES

P-202	P-208
BJ-V039	BD-V035
BJ-V021	BD-V015

DSER OPEN ITEM 132

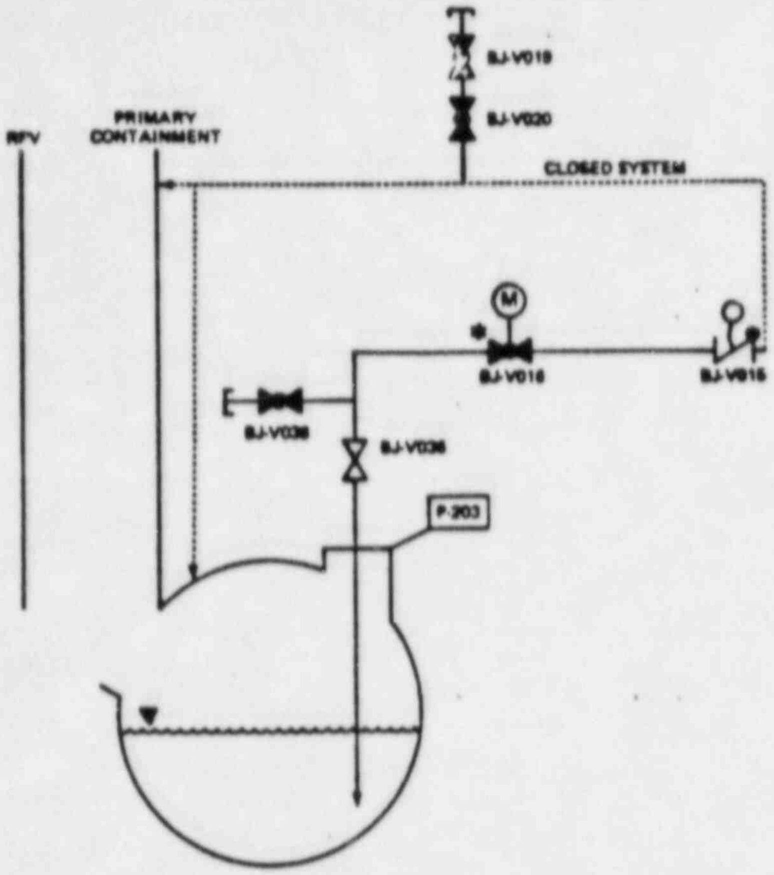
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

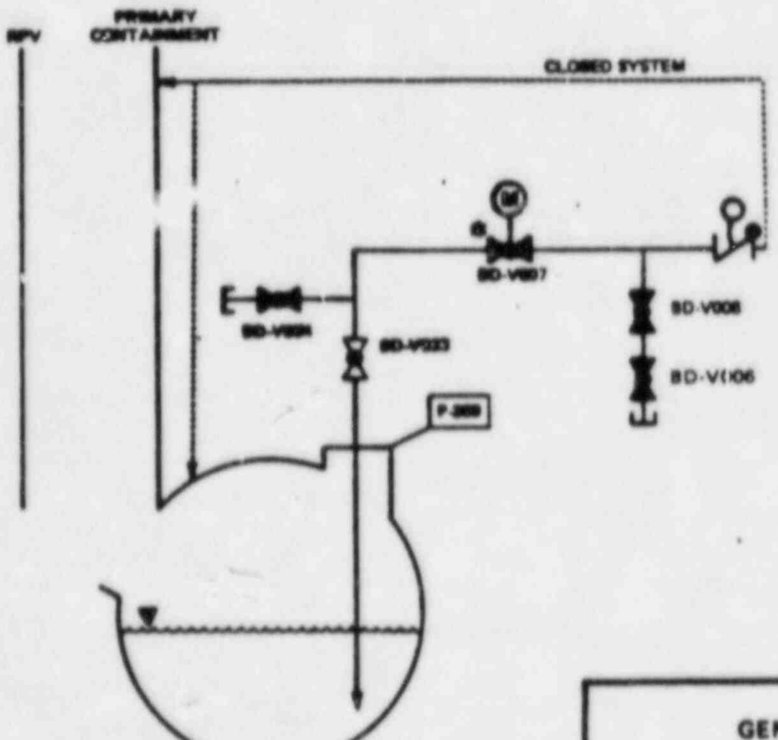
HPCI AND RCIC PUMP SUCTION LINES

FIGURE 6.2-28
SHEET 21 OF 48

AMENDMENT 6, 06/84



OR



DSER OPEN ITEM /32

DETAIL 22

* (SEE LEGEND)

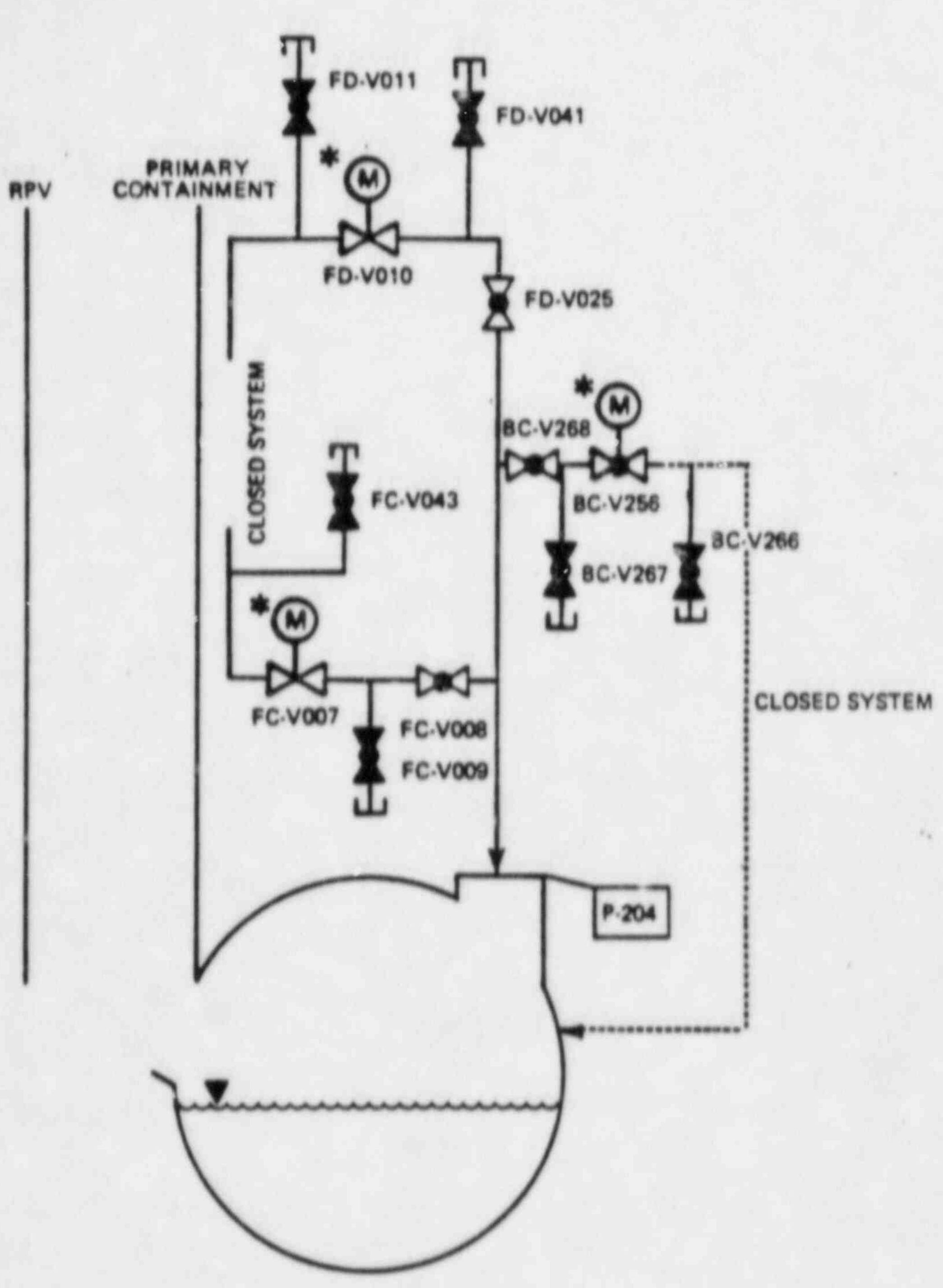
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

RCIC AND HPCI MINIMUM FLOW RETURN LINES

FIGURE 6.2-28
SHEET 22 OF 48

AMENDMENT 6, 06/84



DETAIL 23

DSER OPEN ITEM /32

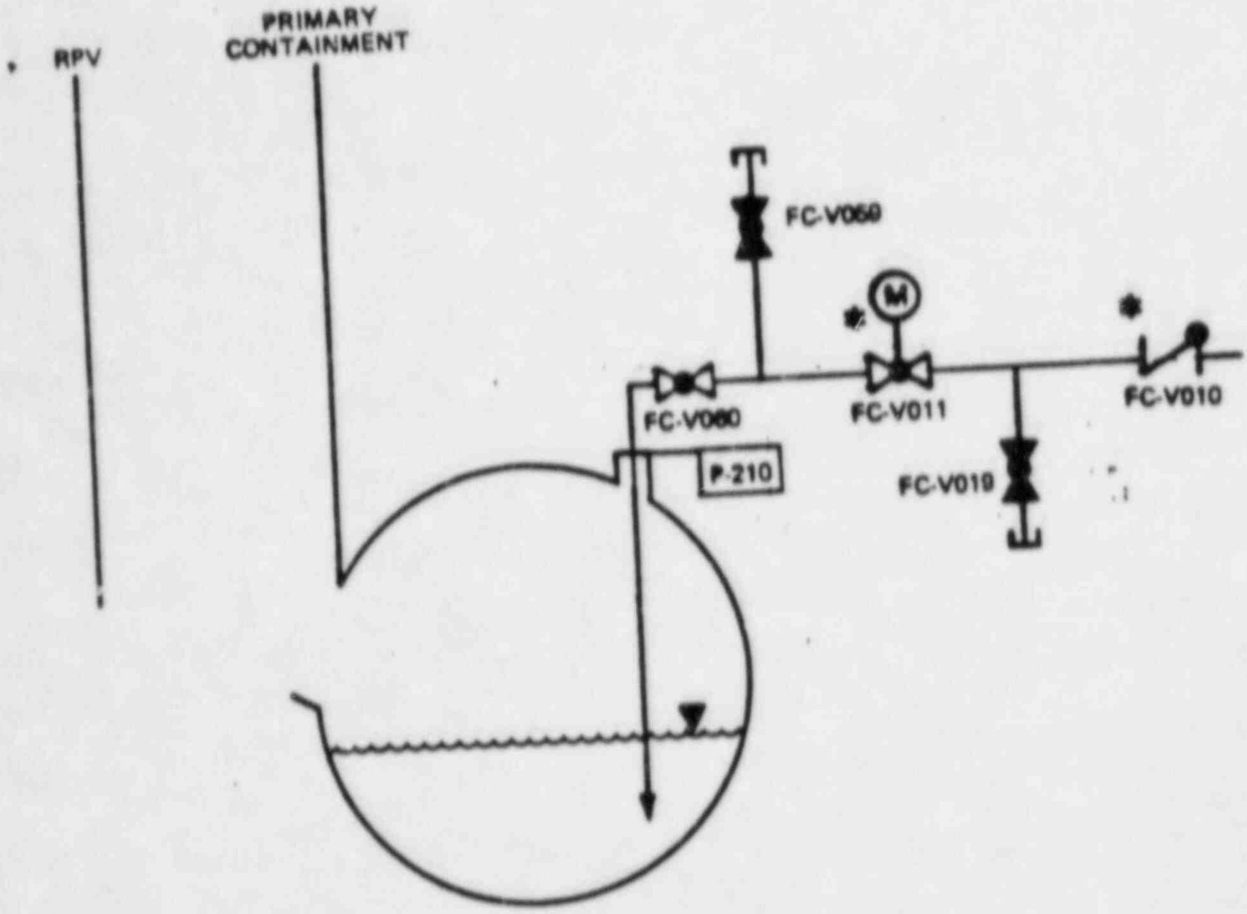
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

HPCI AND RCIC
VACUUM BREAKER NETWORK LINE

FIGURE 6.2-28
SHEET 23 OF 48

AMENDMENT 6, 06/84



DETAIL 24

DSER OPEN ITEM 132

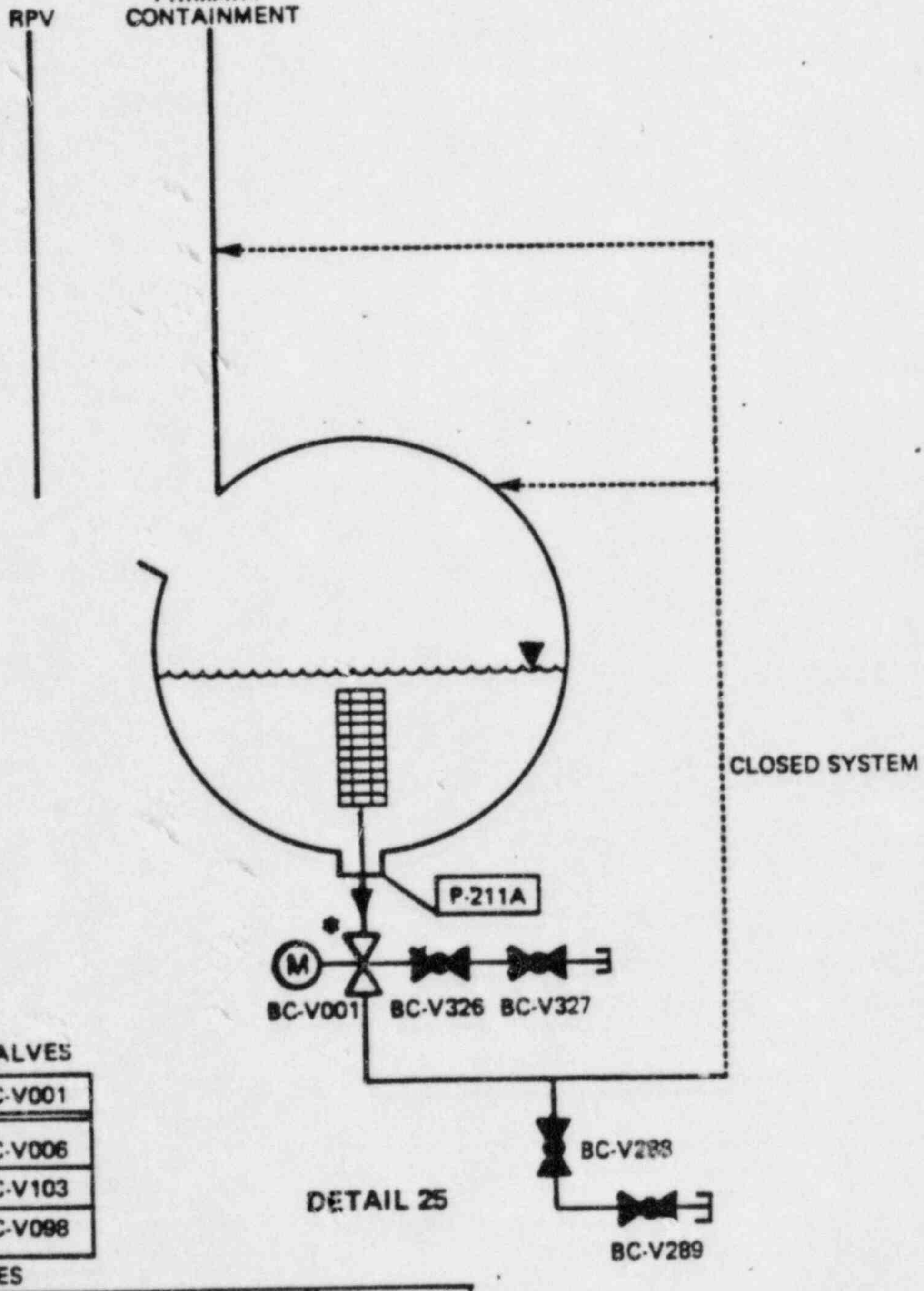
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RCIC
VACUUM PUMP LINE

FIGURE 6.2-28
SHEET 24 OF 48

AMENDMENT 6, 06/84



ISOLATION VALVES

P-211A	BC-V001
P-211B	BC-V006
P-211C	BC-V103
P-211D	BC-V098

TEST VALVES

P-211A	BC-V326	BC-V327	BC-V288	BC-V289
P-211B	BC-V328	BC-V329	BC-V282	BC-V283
P-211C	BC-V341	BC-V340	BC-V293	BC-V294
P-211D	BC-V342	BC-V343	BC-V292	BC-V291

*(SEE LEGEND)

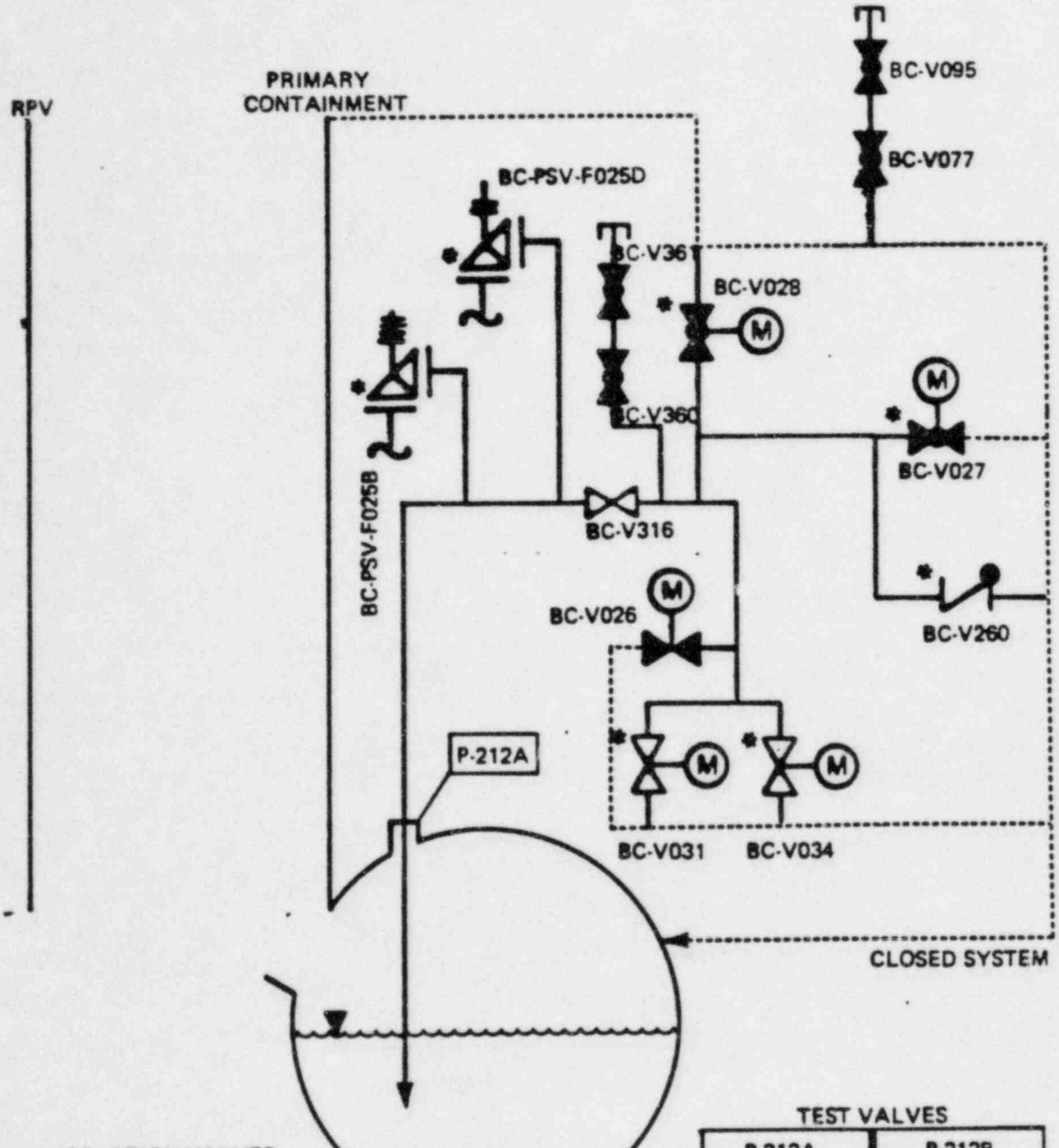
HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RHR PUMP SUCTION LINES

FIGURE 6.2-28
SHEET 25 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM /32



DETAIL 26

TEST VALVES

P-212A	P-212B
BC-V360	BC-V338
BC-V361	BC-V339
BC-V077	BC-V334
BC-V095	BC-V335
BC-V316	BC-V317

ISOLATION VALVES

P-212A	P-212B
BC-PSV-F025D	BC-PSV-F025C
BC-PSV-F025B	BC-PSV-F025A
BC-V028	BC-V124
BC-V027	BC-V125
BC-V026	BC-V126
BC-V034	BC-V128
BC-V031	BC-V131
BC-V260	BC-V206

NOTE: FOR THE PROPER VENTING OF BACKPRESSURE, FLOW CONTROL VALVES BC-LV-F053A AND B (NOT SHOWN) MUST BE OPENED DURING TESTING.

*(SEE LEGEND)

HOPE CREEK GENERATING STATION

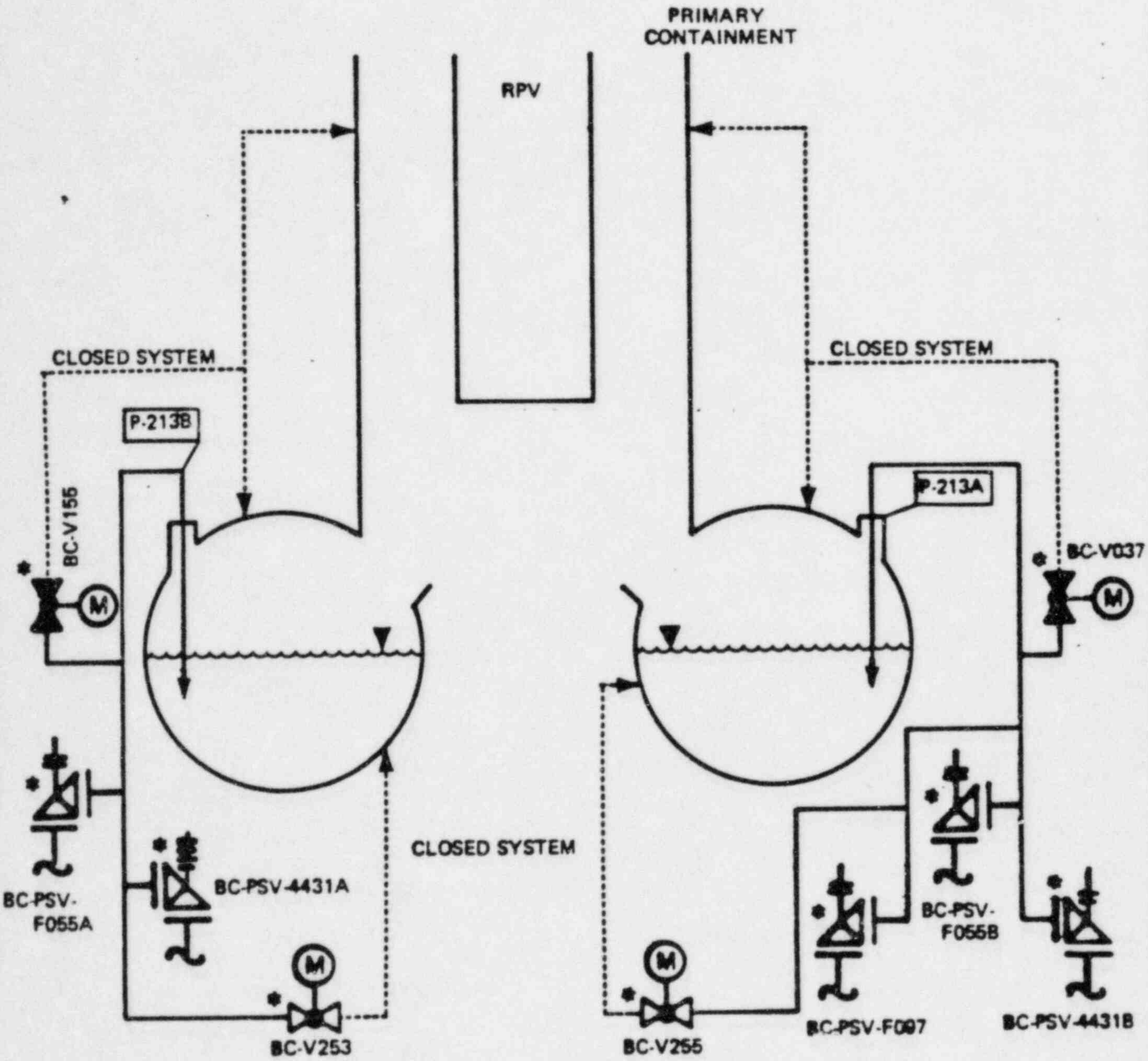
FINAL SAFETY ANALYSIS REPORT

RHR SUPPRESSION POOL WATER COOLING & SYSTEM TEST LINES

FIGURE 6.2-28
SHEET 26 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM / 32



DETAIL 27

DSER OPEN ITEM 132

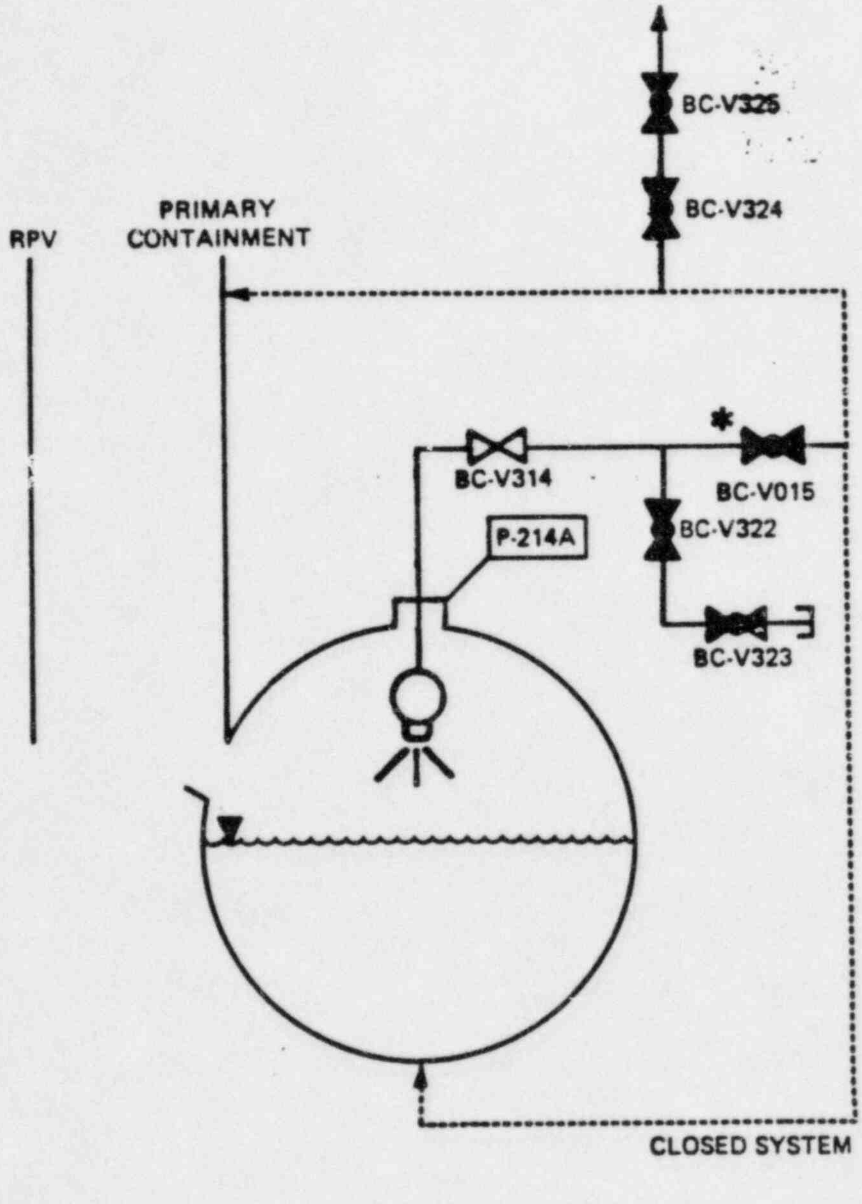
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RHR RELIEF TO
SUPPRESSION CHAMBER LINES

FIGURE 6.2-28
SHEET 27 OF 48

AMENDMENT 6, 06/84



DETAIL 28

ISOLATION VALVES

P-214A	P-214B
BC-V015	BC-V112

TEST AND DRAIN VALVES

P-214A	P-214B
BC-V322	BC-V336
BC-V323	BC-V337
BC-V324	BC-V353
BC-V325	BC-V354
BC-V314	BC-V315

*(SEE LEGEND)

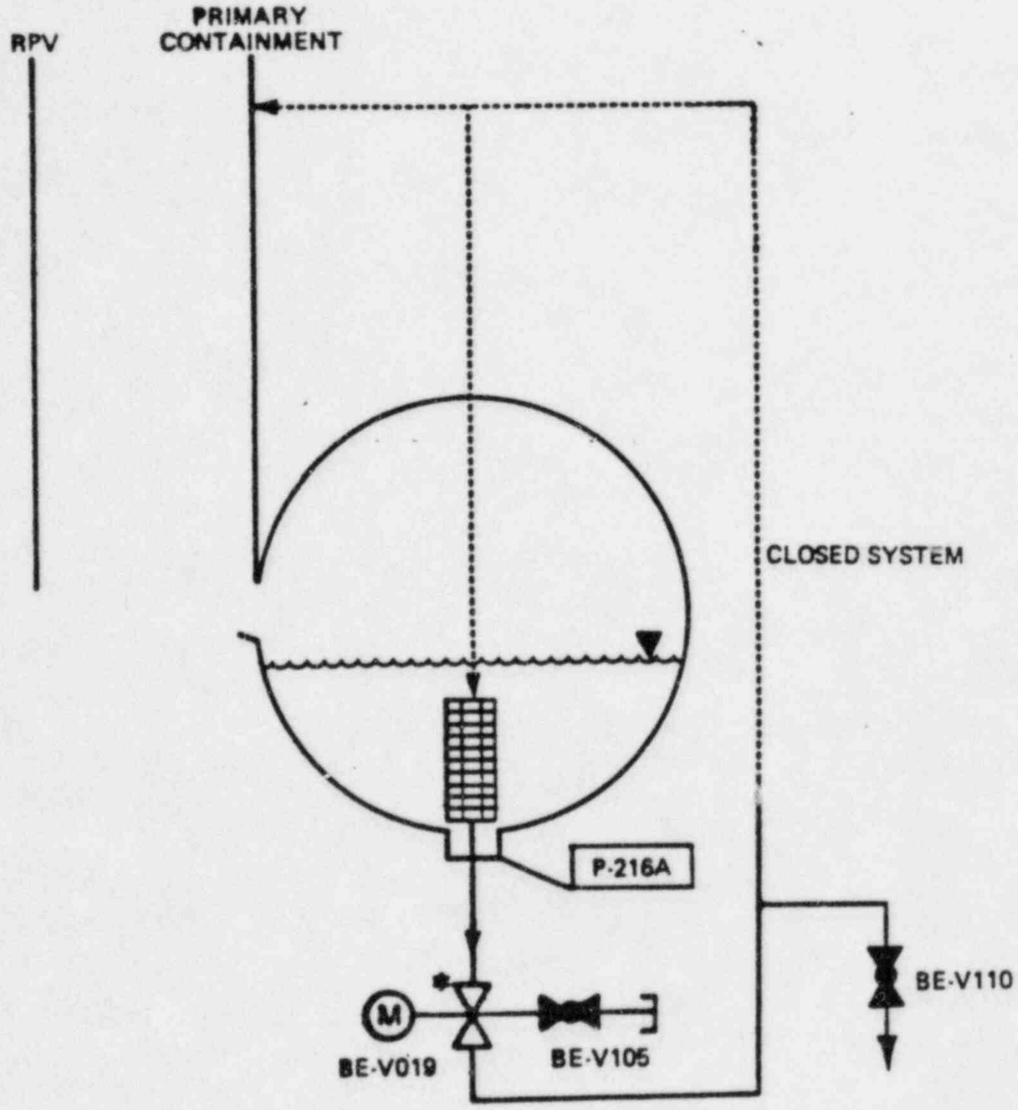
DSER OPEN ITEM / 32

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

RHR TO SUPPRESSION CHAMBER
SPRAY HEADER LINES

FIGURE 6.2-28
SHEET 28 OF 48

AMENDMENT 6, 06/84



DETAIL 29

ISOLATION VALVES

P-216A	BE-V019
P-216B	BE-V020
P-216C	BE-V018
P-216D	BE-V017

TEST AND DRAIN VALVES

P-216A	BE-V105	BE-V110
P-216B	BE-V104	BE-V111
P-216C	BE-V106	BE-V109
P-216D	BE-V107	BE-V108

* (SEE LEGEND)

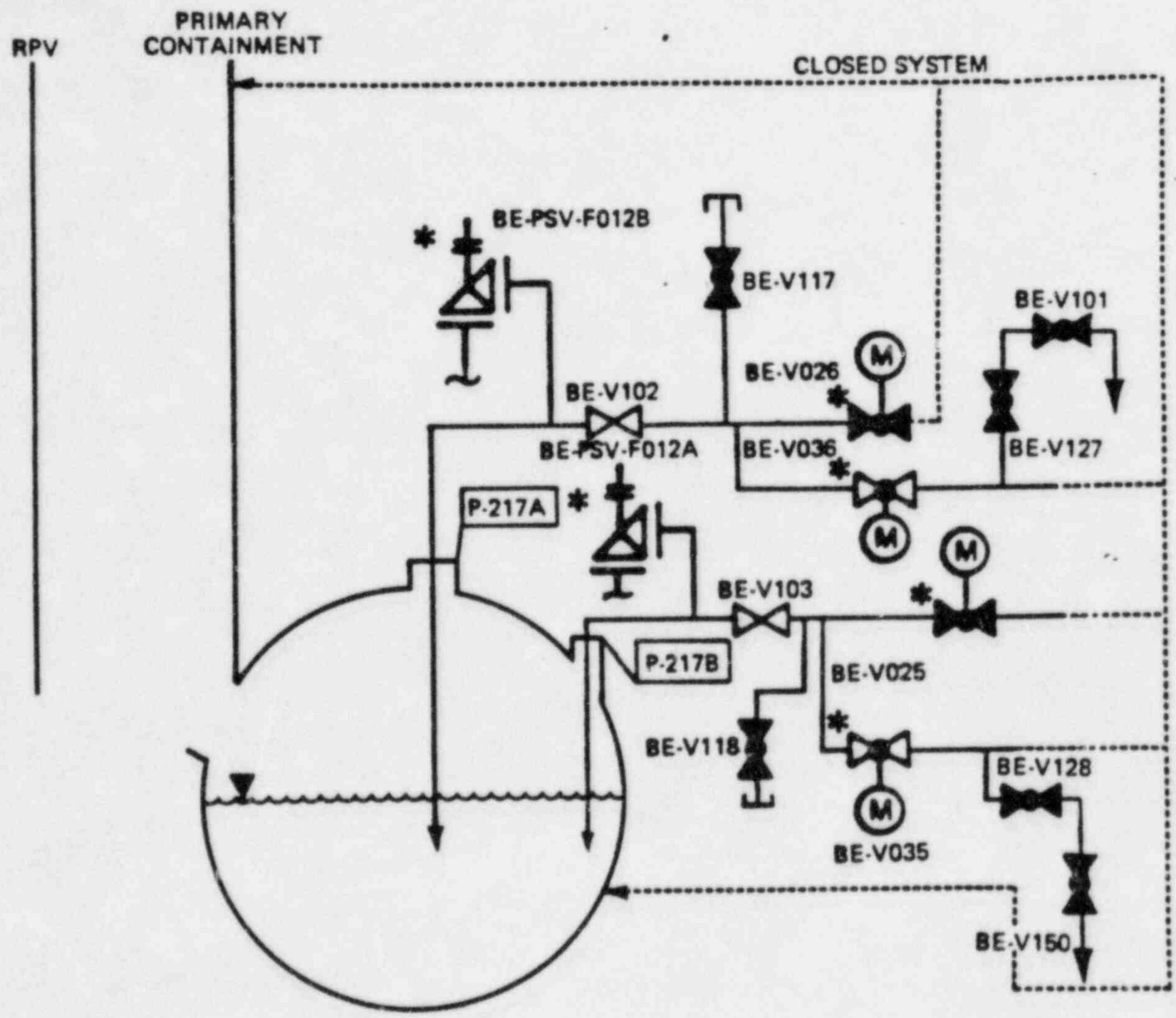
HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CORE SPRAY PUMP SUCTION LINES

FIGURE 6.2-28
SHEET 29 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM 132



DETAIL 30

HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

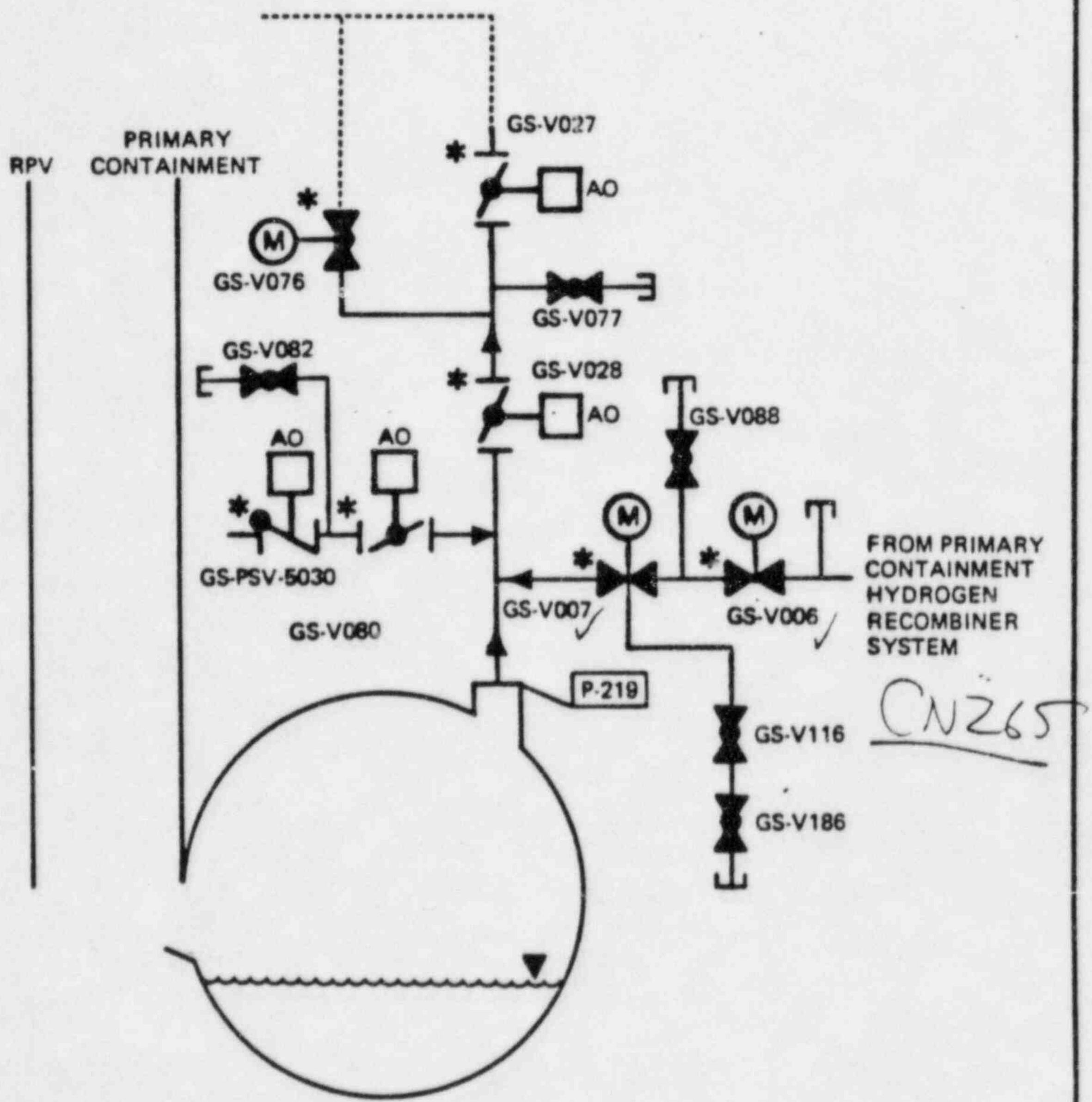
CORE SPRAY TEST TO SUPPRESSION CHAMBER LINES

FIGURE 6.2-28
SHEET 30 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM 132

*(SEE LEGEND)



DETAIL 31

DSER OPEN ITEM 132

*(SEE LEGEND)

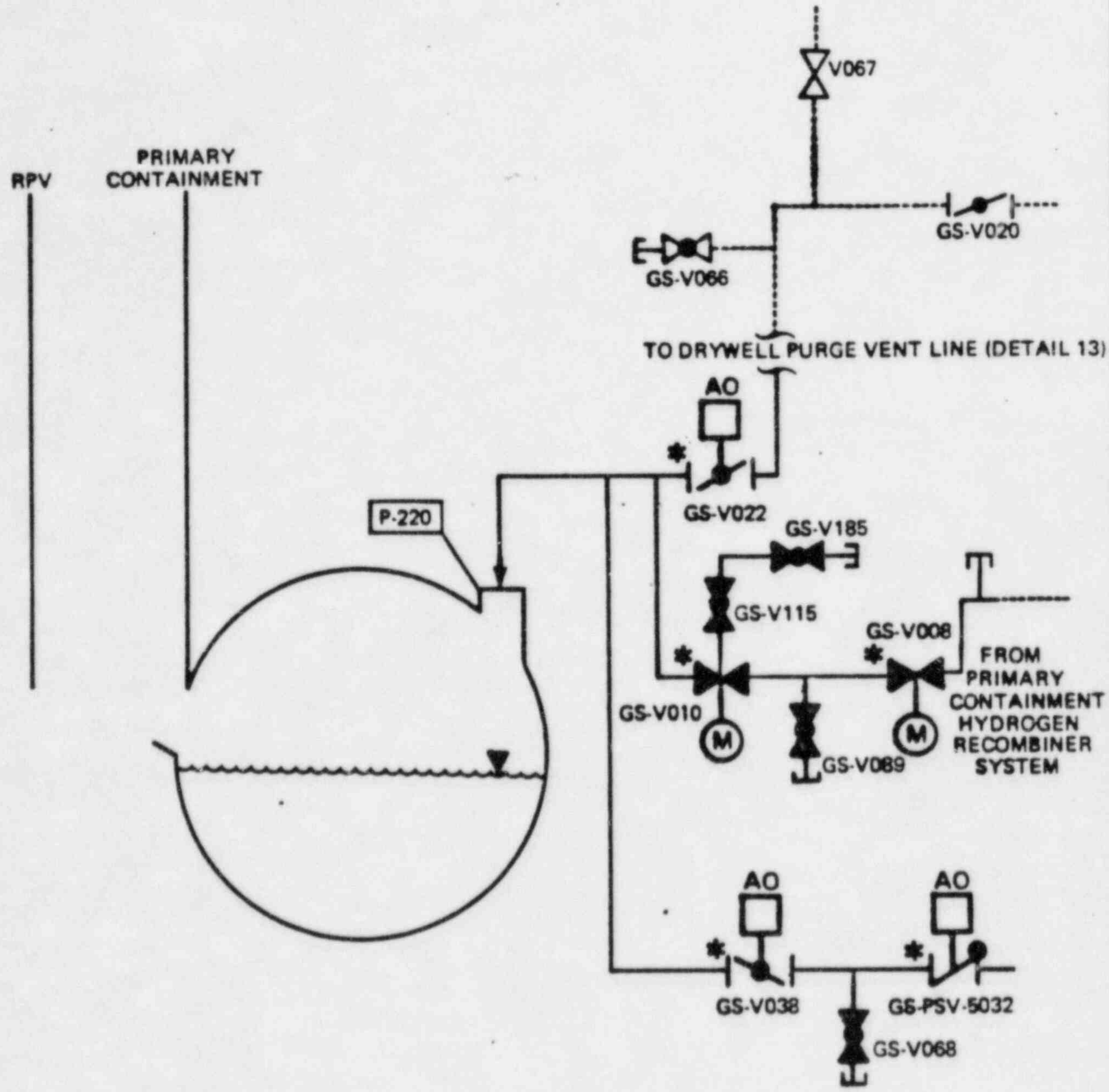
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

SUPPRESSION CHAMBER PURGE & SUPPRESSION CHAMBER VACUUM RELIEF LINE

FIGURE 6.2-28
SHEET 31 OF 48

AMENDMENT 6, 06/84



DETAIL 32

DSER OPEN ITEM 132

*(SEE LEGEND)

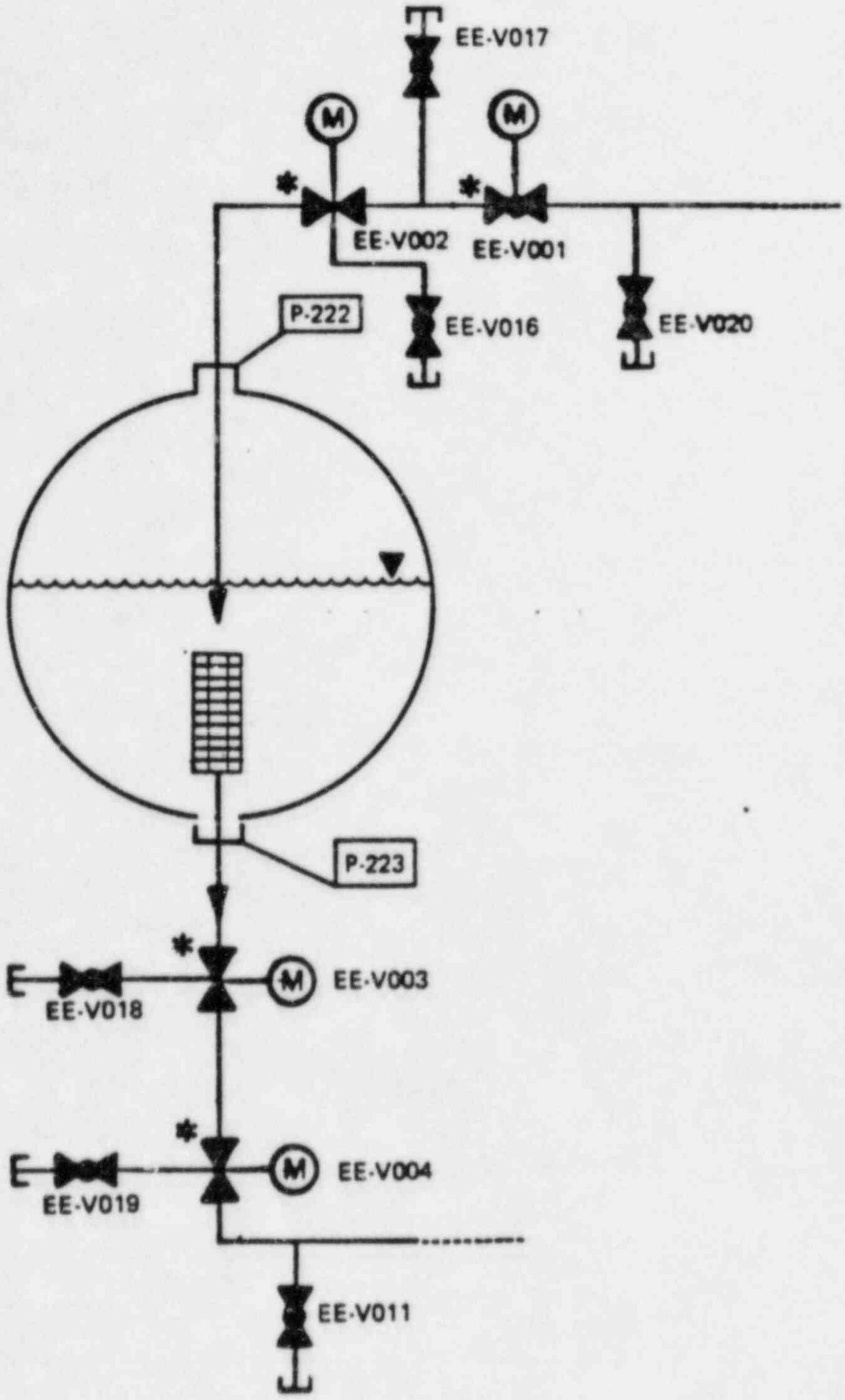
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

SUPPRESSION CHAMBER PURGE INLET & SUPPRESSION CHAMBER VACUUM RELIEF LINE

FIGURE 6.2-28
SHEET 32 OF 48

AMENDMENT 6, 06/84



DETAIL 33

DSER OPEN ITEM /32

*(SEE LEGEND)

HOPE CREEK
GENERATING STATION

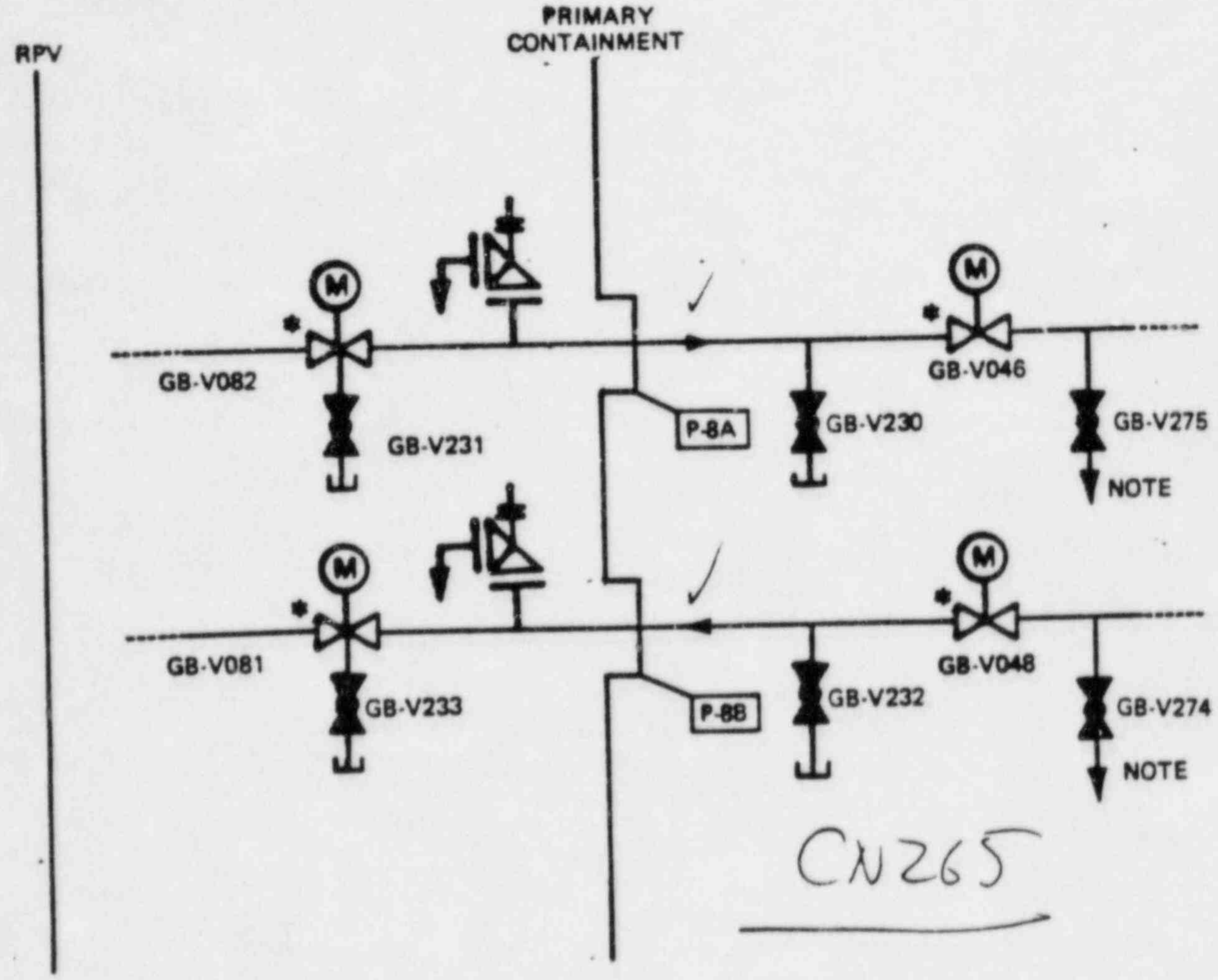
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION POOL CLEANUP
SUPPLY AND RETURN LINES

FIGURE 6.2-28
SHEET 33 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM / 32



DETAIL 34

ISOLATION VALVES

P-8A	GB-V082	GB-V046
P-38B	GB-V084	GB-V071

ISOLATION VALVES

P-8B	GB-V081	GB-V048
P-38A	GB-V083	GB-V070

TEST VALVES

P-8A	GB-V231	GB-V230	GB-V275
D-38B	GB-V237	GB-V236	GB-V241

TEST VALVES

P-8B	GB-V233	GB-V232	GB-V274
P-38A	GB-V235	GB-V234	GB-V240

*(SEE LEGEND)

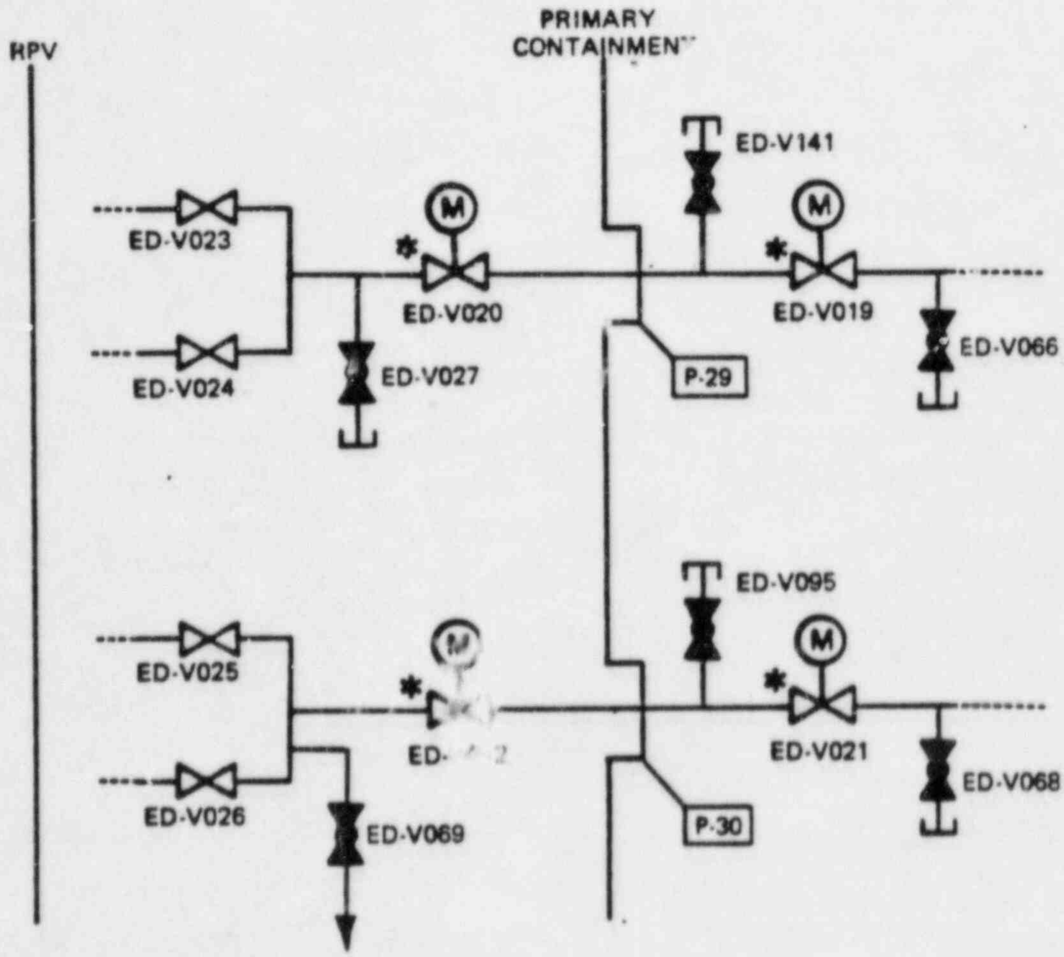
NOTE:
THIS IS A TEST TAP ON P-38A AND P-38B

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CHILLER WATER SYSTEM LINES

FIGURE 6.2-28
SHEET 34 OF 48

AMENDMENT 6, 06/84



DETAIL 35

DSER OPEN ITEM /32

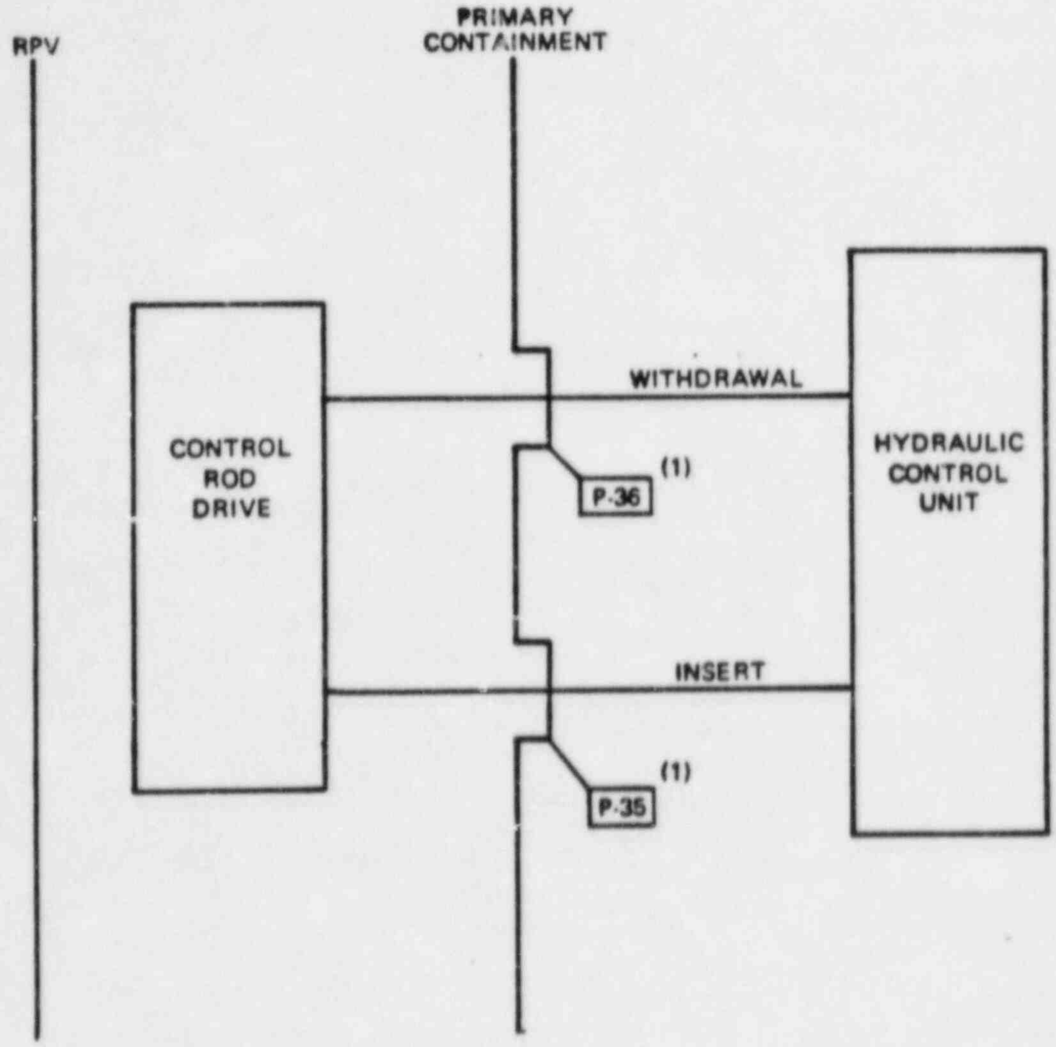
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

REACTOR AUXILIARIES
COOLING LINES

FIGURE 6.2-28
SHEET 36 OF 48

AMENDMENT 6, 06/84



DETAIL 36

(1) TYPICAL OF PENETRATIONS A THRU D

DSER OPEN ITEM /32

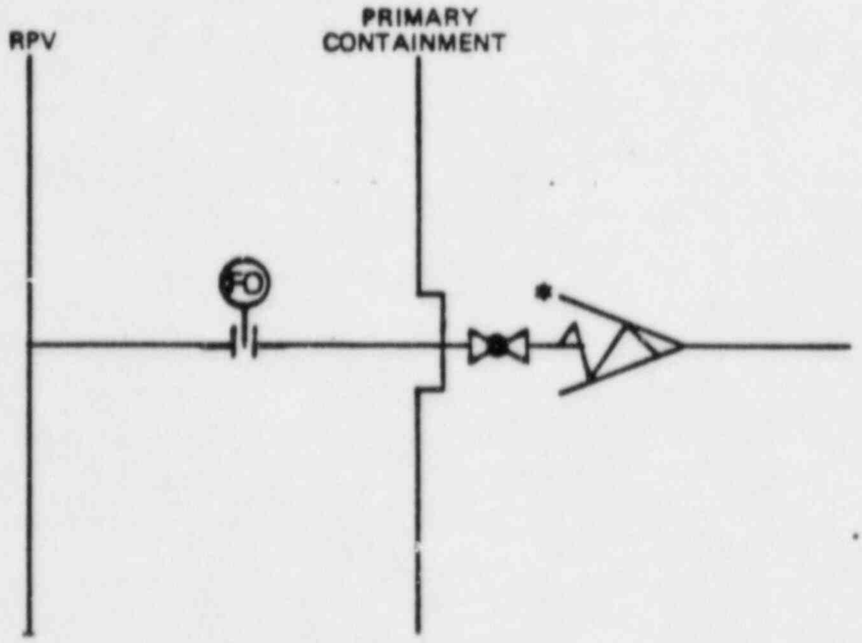
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

CRD SYSTEM

FIGURE 6.2-28
SHEET 36 OF 48

AMENDMENT 6, 06/84



TYPICAL OF NUMEROUS
INSTRUMENT PENETRATIONS

DETAIL 37

DSER OPEN ITEM /32

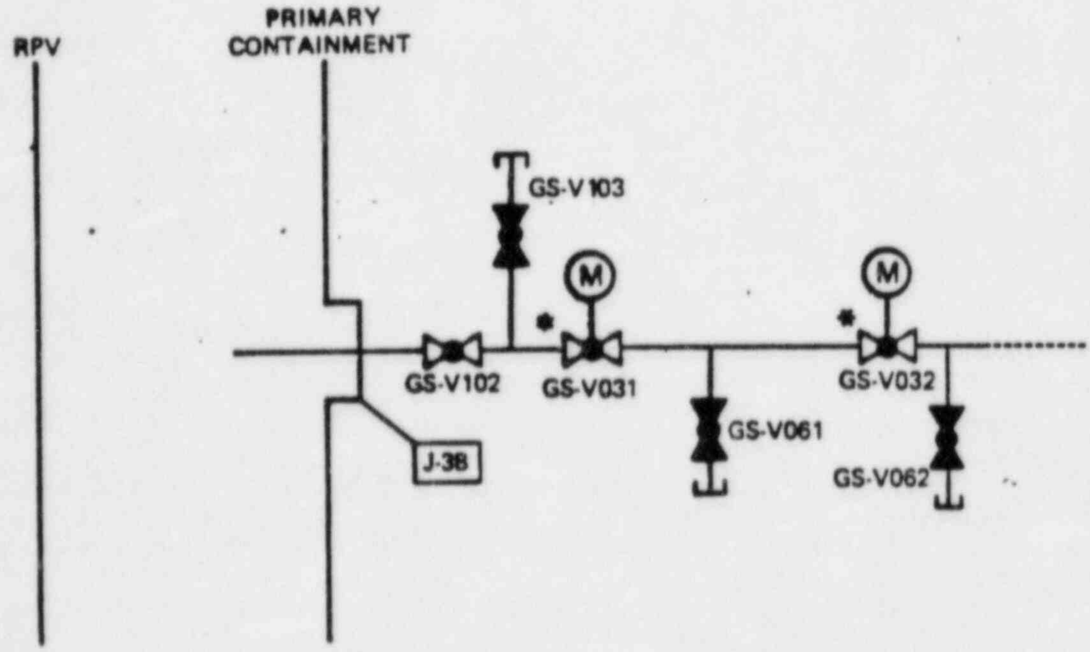
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

TYPICAL OF INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT AND CONNECTING DIRECTLY TO REACTOR COOLANT PRESSURE BOUNDARY

FIGURE 6.2-28
SHEET 37 OF 48 AMENDMENT 6, 06/84



DETAIL 38

ISOLATION VALVES

J-3B	GS-V031	GS-V032
J-6A	SK-V008	SK-V009
J-8C	SK-V005	SK-V006
J-9E	GS-V045	GS-V046
J-10C	GS-V047	GS-V048

TEST VALVES

J-3B	GS-V103	GS-V061	GS-V062	GS-V102
J-6A	SK-V016	SK-V017	SK-V024	SK-V007
J-8C	SK-V013	SK-V014	SK-V025	SK-V004
J-9E	GS-V095	GS-V074	GS-V078	GS-V094
J-10C	GS-V097	GS-V075	GS-V079	GS-V096

*(SEE LEGEND)

DSER OPEN ITEM / 32

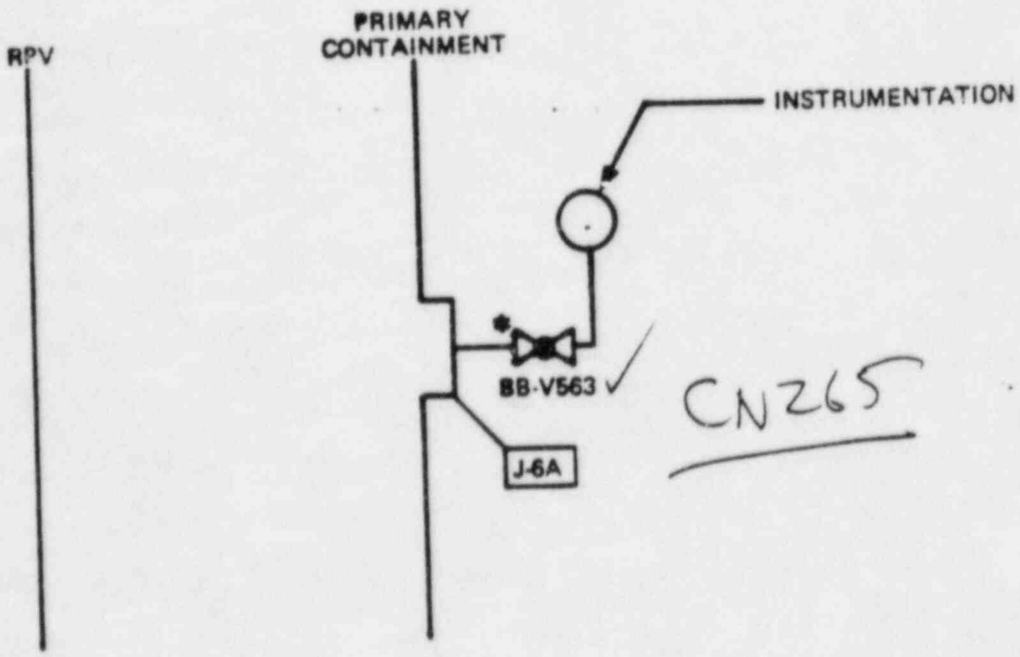
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

TYPICAL OF INSTRUMENT LINES PENETRATING PRIMARY CONTAINMENT AND CONNECTING DIRECTLY TO THE CONTAINMENT ATMOSPHERE

FIGURE 6.2-28
SHEET 38 OF 48

AMENDMENT 6, 06/84



DETAIL 39

J-6A	BB-V563
J-7A	BB-V565
J-8D	BB-V564
J-10D	BB-V566

DSER OPEN ITEM 132

*(SEE LEGEND)

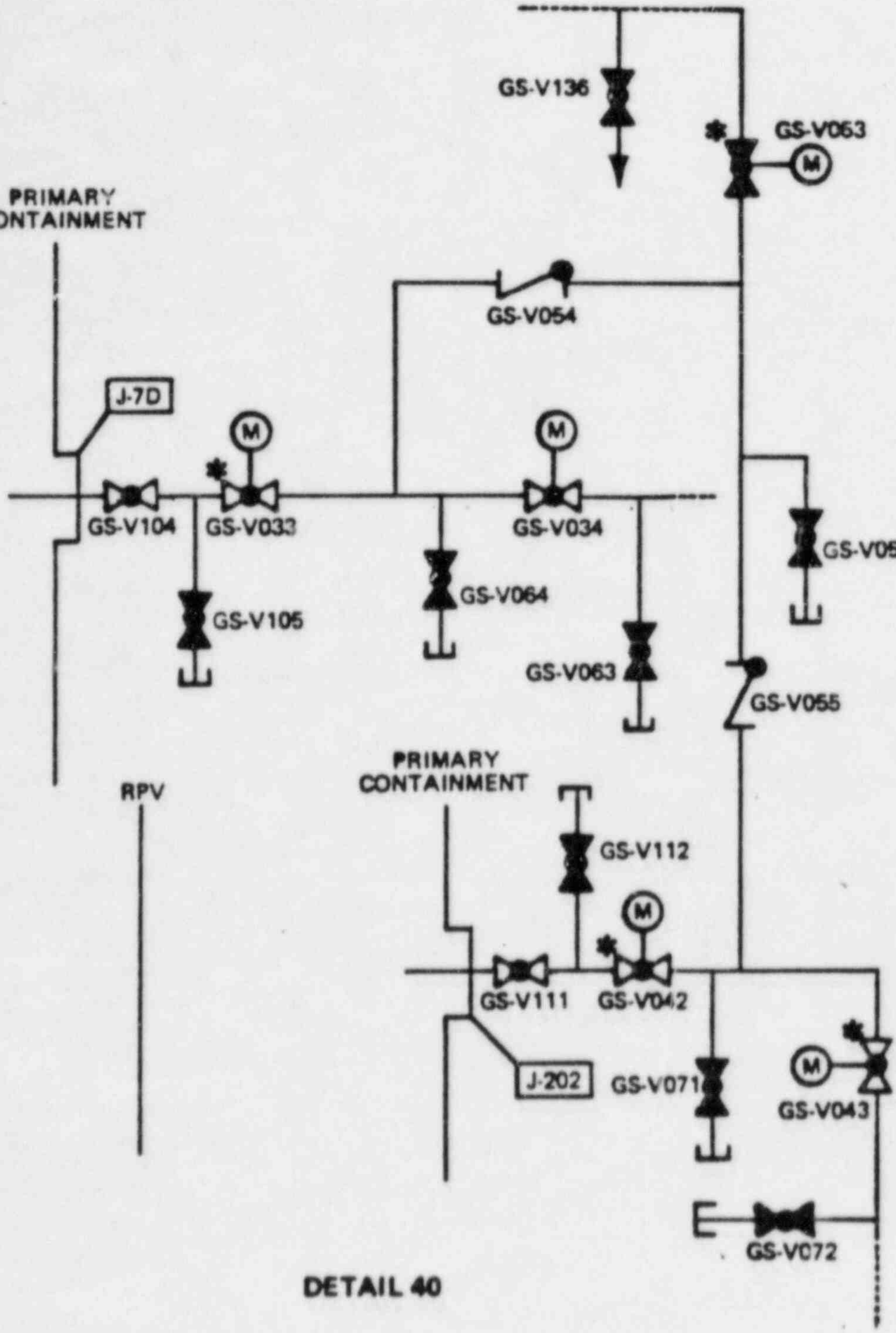
HOPE CREEK
GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

TYPICAL OF INSTRUMENT
LINES FOR SENSING
CONTAINMENT ATMOSPHERE

FIGURE 6.2-28
SHEET 39 OF 48

AMENDMENT 6, 06/84



DETAIL 40

(SEE LEGEND)

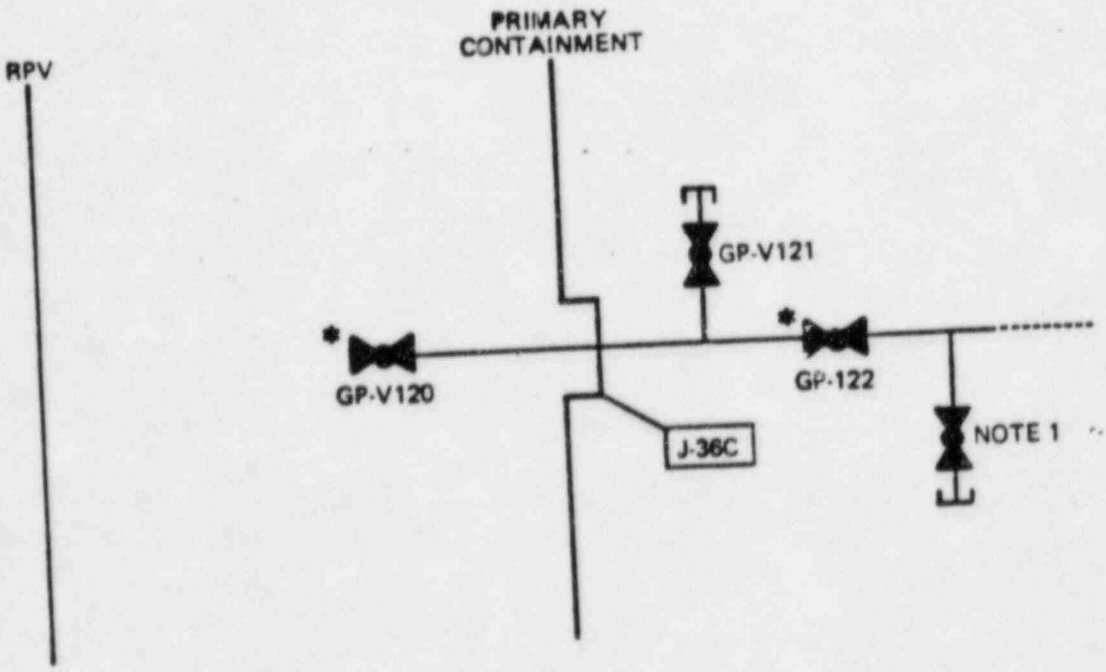
DSER OPEN ITEM / 32

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

HYDROGEN/OXYGEN ANALYZER LINE

FIGURE 6.2-28
SHEET 40 OF 48

AMENDMENT 6, 06/84



DETAIL 41

ISOLATION VALVES

J-36C	GP-V120	GP-122
J-36D	GP-V001	GP-V002

NOTE1: UNNUMBERED VALVE

TEST VALVES

J-36C	GP-121
J-36D	GP-V132

DSER OPEN ITEM 132

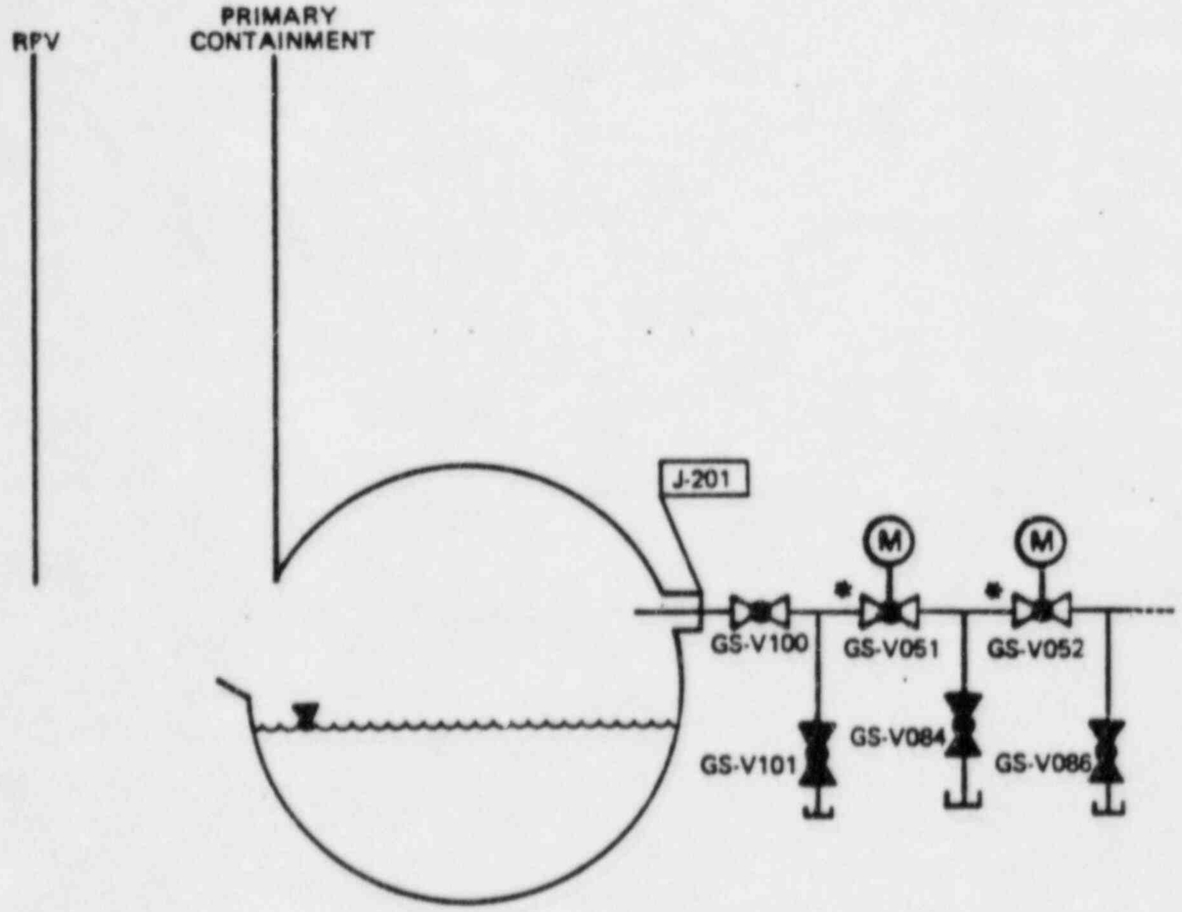
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

PRIMARY CONTAINMENT LEAKAGE
RATE TESTING LINES

FIGURE 6.2-28
SHEET 41 OF 48

AMENDMENT 6.0

*(SEE LEGEND)



DETAIL 42

ISOLATION VALVES

J-201	GS-V051	GS-V052
J-210	GS-V040	GS-V041
J-212	GS-V049	GS-V050

TEST VALVES

J-201	GS-V101	GS-V084	GS-V086	GS-V100
J-210	GS-V110	GS-V069	GS-V070	GS-V109
J-212	GS-V099	GS-V083	GS-V085	GS-V098

DSER OPEN ITEM /32

* (SEE LEGEND)

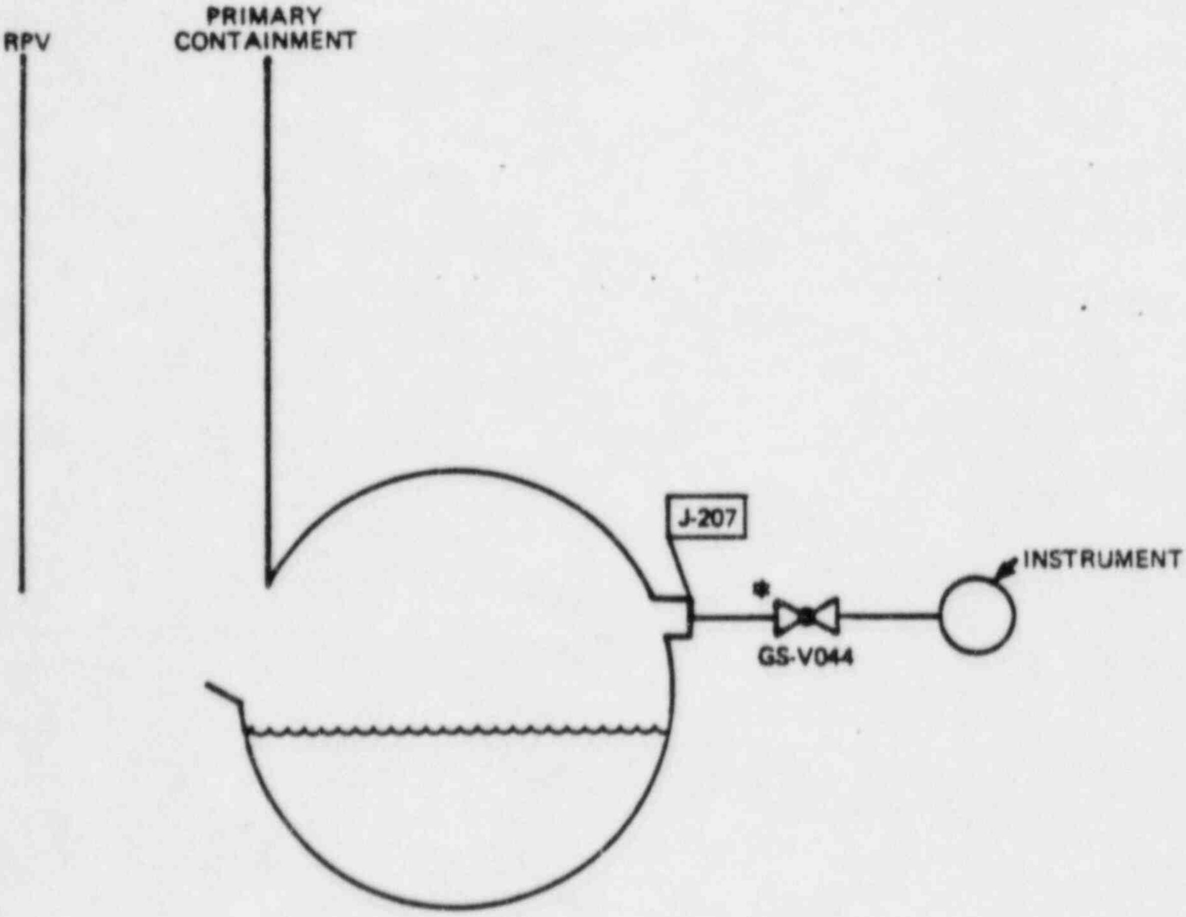
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

TYPICAL OF INSTRUMENT LINES PENETRATING THE SUPPRESSION CHAMBER

FIGURE 6.2-28
SHEET 42 OF 48

AMENDMENT 6, 06/84



DETAIL 43

J-207	GS-V044
J-208	GS-V087

DSER OPEN ITEM 132

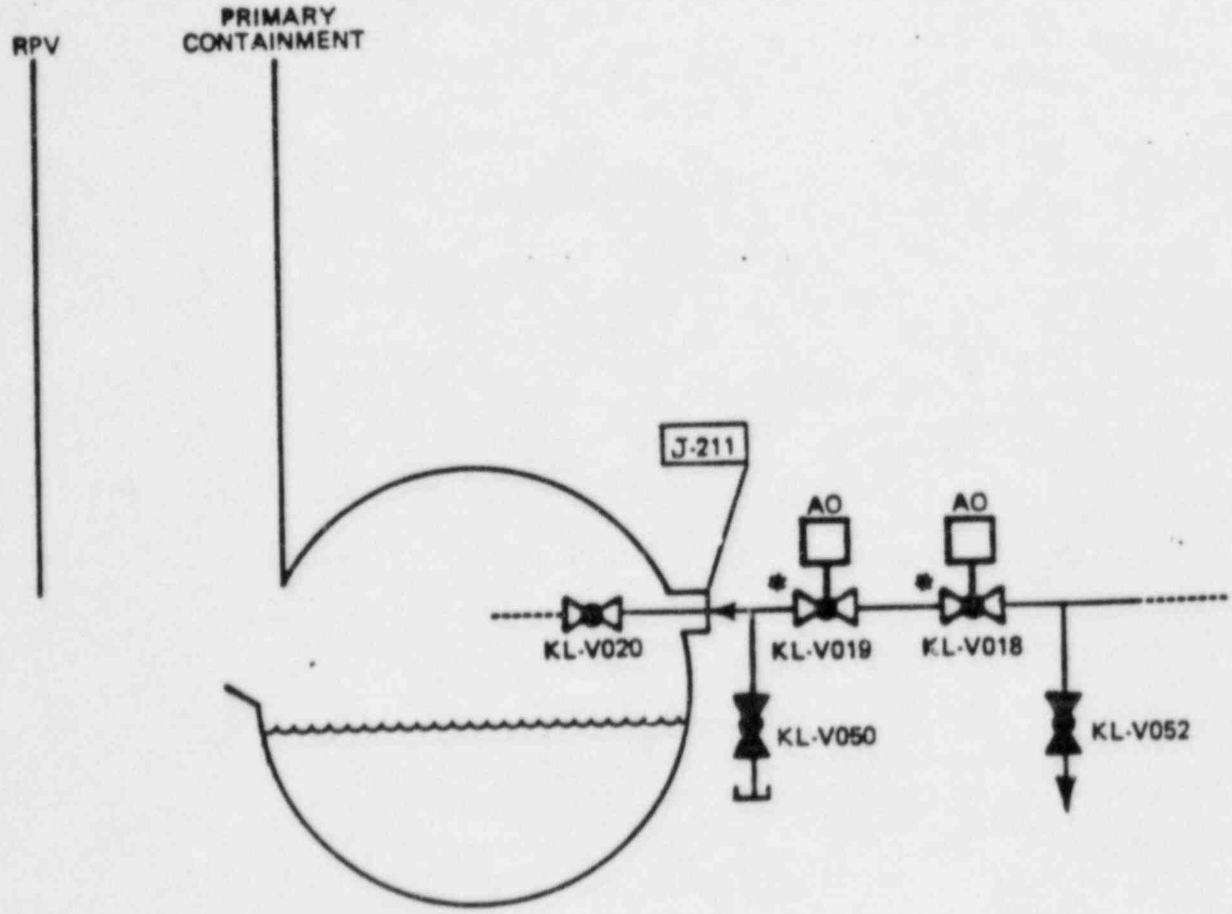
E LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

TYPICAL OF LINES FOR SENSING
SUPPRESSION CHAMBER PRESSURE

FIGURE 6.2-28
SHEET 43 OF 48

AMENDMENT 6, 06/84



DETAIL 44

DSER OPEN ITEM 132

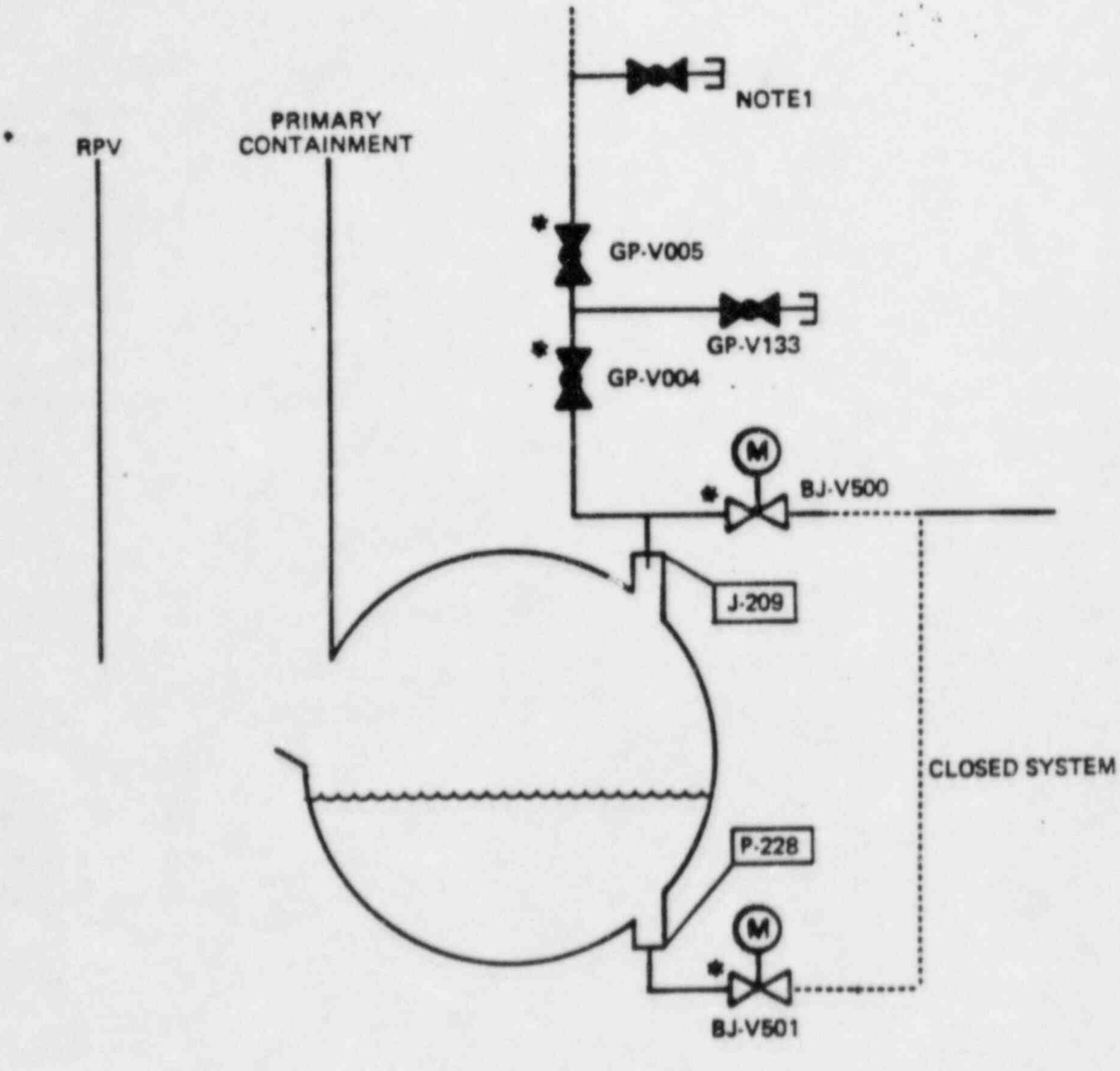
*(SEE LEGEND)

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

INSTRUMENT AIR TO SUPPRESSION CHAMBER LINES

FIGURE 6.2-28
SHEET 44 OF 48

AMENDMENT 6, 06/84



DETAIL 45

DSER OPEN ITEM 132

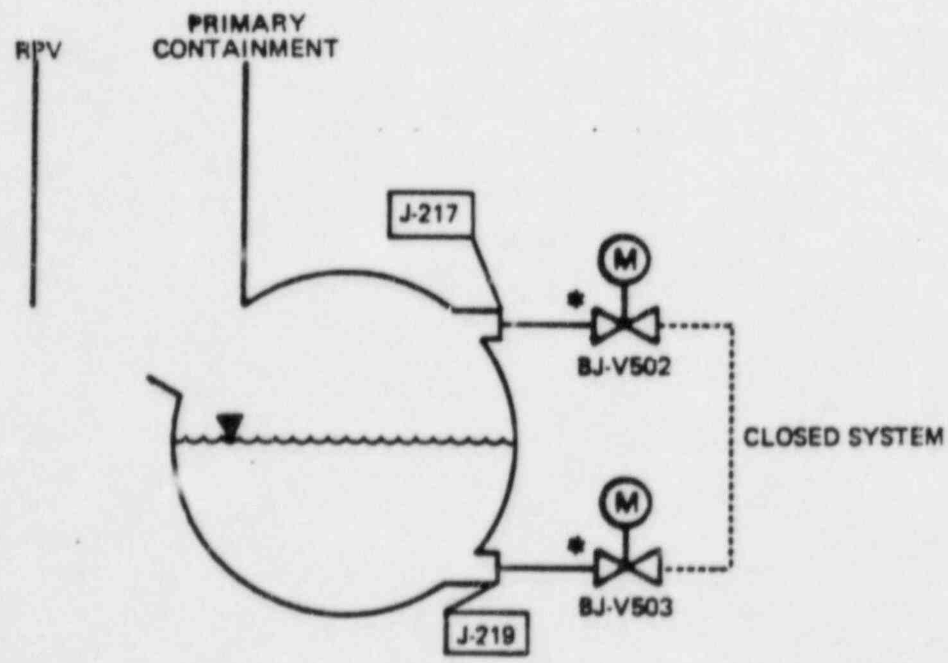
HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SUPPRESSION POOL LEVEL LINE

FIGURE 6.2-28
SHEET 45 OF 46

AMENDMENT 6, 06/84

*(SEE LEGEND)



DETAIL 46

DSER OPEN ITEM 132

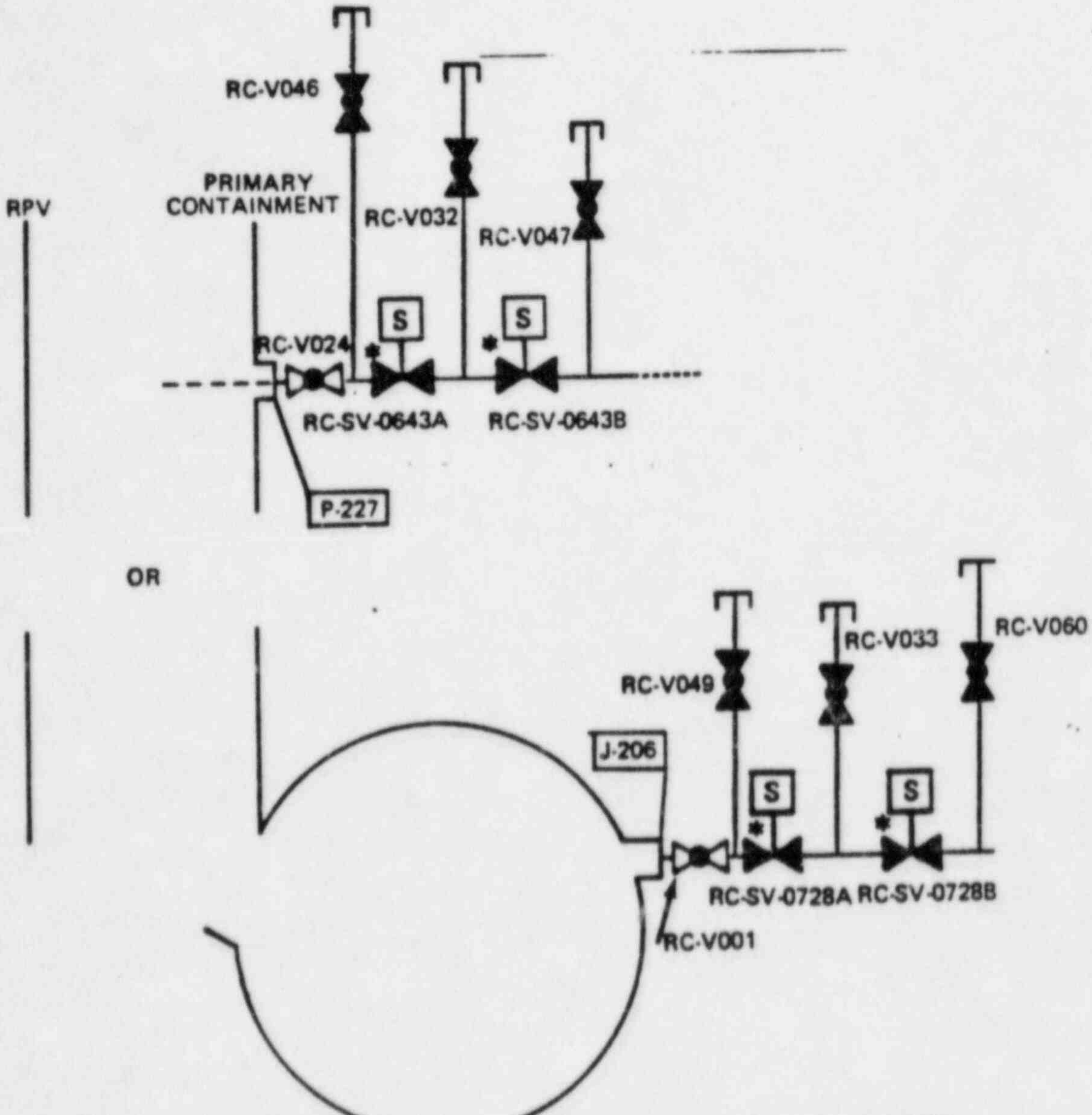
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

TYPICAL OF INSTRUMENT
LINES FOR SENSING
SUPPRESSION POOL LEVEL

FIGURE 6.2-28
SHEET 46 OF 48

AMENDMENT 6, 06/84

E LEGEND)



ISOLATION VALVES

P-227	RC-SV-0643A	RC-SV-0643B
J-7E	RC-SV-0730A	RC-SV-0730B
J-10E	RC-SV-0731A	RC-SV-0731B

DETAIL 47

ISOLATION VALVES

J-206	RC-SV-0728A	RC-SV-0728B
J-220	RC-SV-0707A	RC-SV-0707B
J-221	RC-SV-0729A	RC-SV-0729B

TEST VALVES

P-227	RC-V046	RC-V032	RC-V047	RC-V024
J-7E	RC-V055	RC-V036	RC-V056	RC-V004
J-10E	RC-V057	RC-V037	RC-V058	RC-V005

TEST VALVES

J-206	RC-V049	RC-V033	RC-V050	RC-V001
J-220	RC-V051	RC-V034	RC-V052	RC-V002
J-221	RC-V053	RC-V035	RC-V054	RC-V003

HOPE CREEK GENERATING STATION

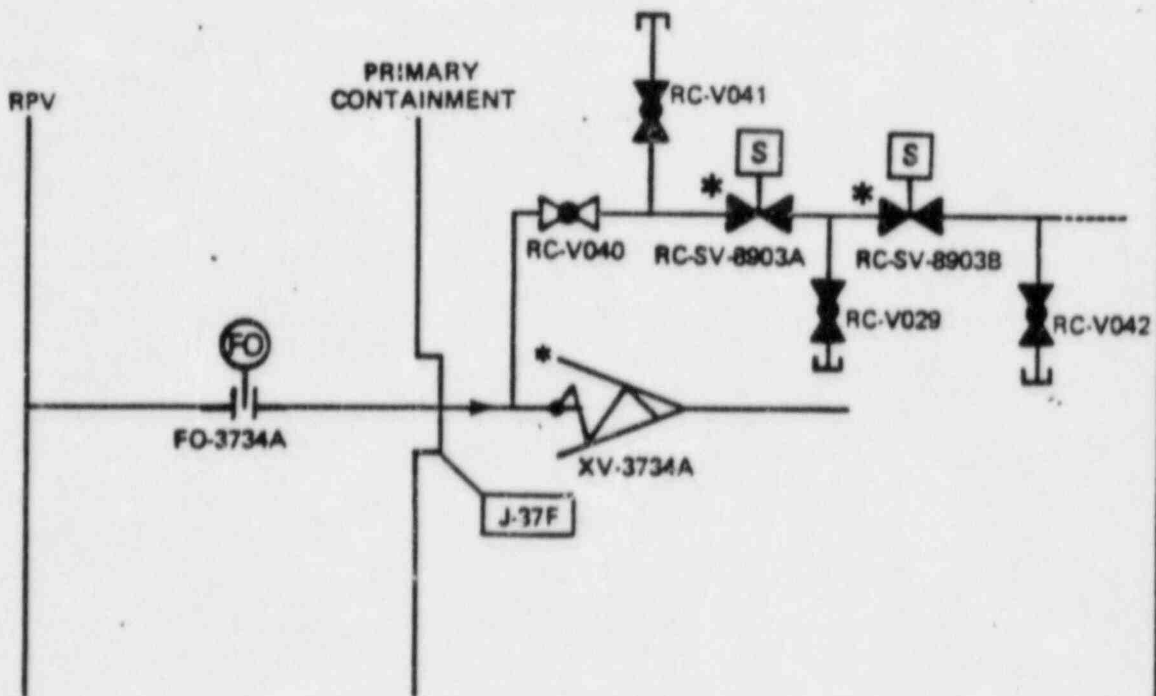
FINAL SAFETY ANALYSIS REPORT

POST-ACCIDENT SAMPLING

FIGURE 6.2-28
SHEET 47 OF 48

AMENDMENT 6, 06/84

DSER OPEN ITEM 132



DETAIL 48

DSER OPEN ITEM 132

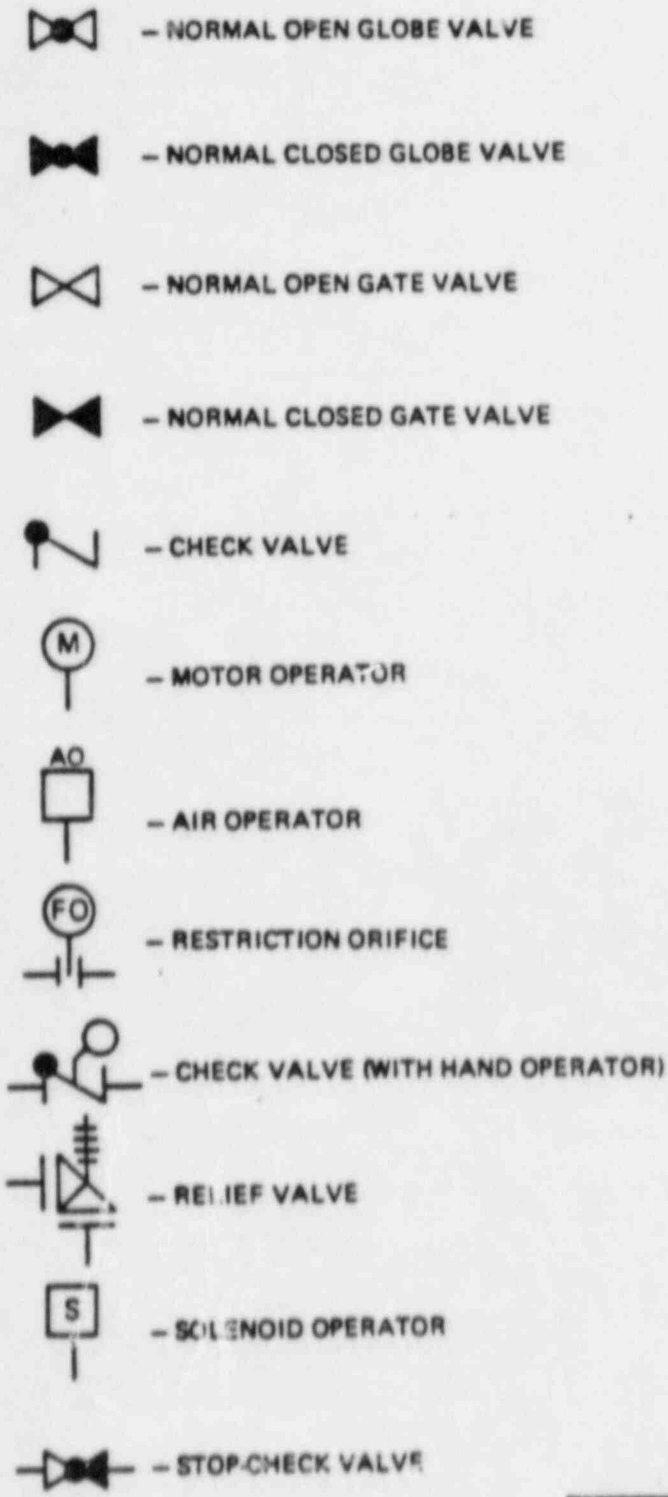
*(SEE LEGEND)

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

POST-ACCIDENT LIQUID SAMPLING
AND
REACTOR INSTRUMENTATION

FIGURE 6.2-28
SHEET 48 OF 48

AMENDMENT 6, 06/84



DSER OPEN ITEM 132

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

LEGEND

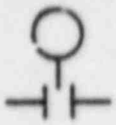
FIGURE 6.2-28a
SHEET 1 OF 2

AMENDMENT 6, 6/84

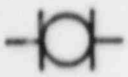
– EXPLOSIVE VALVES

*

– INDICATES CONTAINMENT ISOLATION VALVE

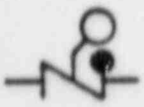


– SPECTACLE FLANGE



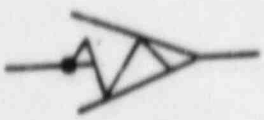
– BALL VALVE

– SCREEN

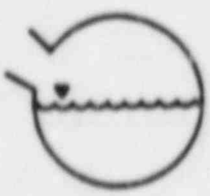


– CHECK VALVE WITH MANUAL LEVER

– SPRAY NOZZLE



– EXCESS FLOW CHECK VALVE



– SUPPRESSION CHAMBER

– UNVALVED TEST CONNECTION

– DRAINPOT

DSER OPEN ITEM / 32

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

LEGEND

FIGURE 6.2-28a
SHEET 2 OF 2

AMENDMENT 6, 6/84

<u>DSER Open Item No. 134</u> (DSER Section 6.2.6)

CONTAINMENT LEAKAGE TESTING

The applicant has provided information to demonstrate compliance with the testing requirements of Appendix J to 10 CFR 50. As a result of this review, the staff finds that the information provided by the applicant is not complete. We are awaiting information from the applicant outlined in Section 6.2.4, in order to properly conduct our review of containment leak testing. We will evaluate this information when it becomes available and report on this matter in a supplement to this SER.

<u>RESPONSE</u>

For the information requested above see the response to DSER Open Item 132.

DSER Open Item No. 141g (Section 9.1.3)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

Additionally, the information provided through Amendment 3 was not sufficient for the staff to complete its evaluation of the spent fuel pool sampling and monitoring. To complete the review, the following information is needed:

(1) Describe the sampling procedure, analytical instrumentation, and sampling frequency for monitoring spent fuel pool purity.

(2) State the radiochemical limits for initiating corrective action.

The applicant's response should consider permissible gross gamma and iodine activities and the demineralizer decontamination factor.

RESPONSE

FSAR Section 9.1.3.2.2.4 has been revised to provide the requested information.

The stainless steel filter-demineralizer vessels are of the pressure precoat type. A tube nest assembly consisting of the tube sheet, clamping plate, filter elements, and support grid is inserted as a unit between the flanges of the vessel. The filter elements are stainless steel and are mounted vertically in the vessel. Air scour connections are provided below the tube sheet, and vents are provided in the upper head of each vessel. The filter elements are installed and removed through the top of each vessel. The holding elements are designed to be coated with powdered ion exchange resin as the filtering medium.

The fuel pool filter-demineralizers maintain the following effluent water quality specifications:

Specific conductivity at 25°C, micromho/cm	≤0.1
pH at 25°C	6.0 to 7.5
Heavy elements (Fe, Hg, Cu, Ni), ppm	0.05
Silica (as SiO_2), ppm	<0.05
Chloride (as Cl-), ppm	<0.02
Total insolubles, ppm	90% removal to a minimum of 0.01 ppm

Insert A →

The filter-demineralizers are designed to be backwashed periodically with water to remove resin and accumulated sludge from the holding elements. Service air pressure loosens the material from the holding elements and the backwash slurry drains through the gravity drainline to the waste sludge phase separator in the solid waste management system.

The resin tank provides adequate volume for one precoating of one filter demineralizer vessel.

The resin eductor transfers the precoat mixture of resin to the holding pump suction line at a flow rate of 4 gpm.

The holding pumps are designed to recirculate a uniform mixture of resin through the filter-demineralizer vessel being precoated at a flow rate of 1.5 gpm/ft² of filter element surface area, and to automatically start and maintain the precoat material on the filter elements when the system flow rate falls below the value necessary to keep the precoat on the elements.

INSERT A

The influent and effluent water of the FPCC is continuously monitored by on line PH and conductivity instrumentation. In addition, grab samples of the influent H_2O will be analyzed once per week for chloride and for gamma isotopic and once per month for heavy metals (Fe, Cu, Hg, Ni). Grab samples of effluent water will be analyzed weekly for chloride, silica, suspended solids, tritium, and for gamma isotopes.

Decontamination factors (df) of greater than 10 are expected for any chloride present and greater than 5 for isotopes of Iodine and Cobalt. Resin bed(s) will be regenerated and/or replaced when these dfs are not achieved.

DSER OPEN ITEM 141g

DSER Open Item No. 145 (DSER Section 9.2.2)

ISI PROGRAM AND FUNCTIONAL TESTING OF SAFETY AND TURBINE AUXILIARIES COOLING SYSTEMS

RESPONSE

This item is not an open item per telephone conversation between J. M. Ashley (PSE&G) and John Ridgely (NRC-ASB) on April 23, 1984.

TELEPHONE NOTES
PSE&G Hope Creek Licensing (Bethesda)

DATE: April 23, 1984

FROM: J.M. Ashley

TO: Dave Wagner

SUBJECT: HCGS DSER Open Hem 145

Discussion

(This telecon corrects my telecon of 3/22/84 on HCGS DSER Open Items - that telecon referred to Open Item 145, FSAR Section 9.2.2. The reference should have been Open Item 146, FSAR Section 9.2.6.)

Ashley called to find out what NRC staff concerns existed with respect to FSAR section 9.2.2 (Item 145).

Wagner said he had checked the section and that the NRC has no outstanding concerns with the section. The item was inadvertently listed.

J M Ashley

DSER OPEN ITEM 145

HCGS

<u>DSER Open Item No. 146</u> (DSER Section 9.2.6)

SWITCHES AND WIRING ASSOCIATED WITH HPCI/RCIC TORUS SUCTION

<u>RESPONSE</u>

This item is closed per the March 22, 1984 telecon between J. M. Ashley (PSE&G), and J. Ridgely (NRC).

TELEPHONE NOTES

PSE&G Hope Creek Licensing (Bethesda)

Date: March 22, 1984

From: J.M. Ashley

To: D. Wagner, J. Ridgely (ASB)

Subject: HCGS DSER Open Items

Discussion

Ashley called to find out what NRC concerns existed with respect to FSAR Sections 3.5.1.2 (Item 30), 9.2.2 (Item 145) and 9.4.4 (Item 152). 6 See 4/23/84 J. Ashley + D. Wagner Telephone Notes.

Ridgely explained that these items were inadvertently listed as open items in the listing of open items at the front of the DSER. The NRC has no outstanding concerns with the sections.

J. M. Ashley

DSER OPEN ITEM 146

TELEPHONE NOTES
PSE&G Hope Creek Licensing (Bethesda)

DATE: April 23, 1984

FROM: J.M. Ashley

TO: Dave Wagner

SUBJECT: HCGS DSER Open Hem 145

Discussion

(This telecon corrects my telecon of 3/22/84 on HCGS DSER Open Items - that telecon referred to Open Item 145, FSAR Section 9.2.2. The reference should have been Open Item 146, FSAR Section 9.2.6.)

Ashley called to find out what NRC staff concerns existed with respect to FSAR section 9.2.2 (Item 145).

Wagner said he had checked the section and that the NRC has no outstanding concerns with the section. The item was inadvertently listed.

J M Ashley

DSER OPEN ITEM 146

DSER Open Item No. 158 (DSER Section 9.5.1.5.a)

CLASS B FIRE DETECTION SYSTEM

The applicant is providing a detection system in accordance with NFPA 72D. The portion of the system that protects the reactor building is a Class A supervised system. The portion of the system that protects the balance of the plant is provided with Class B supervision. This does not meet staff guidelines. A Class B system will not continue to provide detection capability if a fault occurs in the system.

The staff will require the applicant to provide a system that complies with NFPA 72D for a Class A system, with detectors installed in accordance with NFPA 72E.

RESPONSE

A class A early warning fire and smoke detection system, designed in accordance with NFPA 72D, is provided in the reactor building and in the following safety-related equipment areas which can be used for the first four hours of hot shutdown:

a. 250 Vdc battery and inverter room for the RCIC system,
b. 250 Vdc battery and inverter room for the HPCI system,
c. Cable spreading room,
d. Switchgear Room B,
e. Control equipment room.

In addition, a Class A system is provided in the remote shutdown panel room.

The Class A system will initiate a fire alarm even with a single break or single ground fault of the detection circuit between the local control panel and the associated detectors.

The above design was discussed with the NRC at the DSER fire protection open item meeting held April 18, 1984. The staff indicated that this design would be acceptable. As justified below, we have not provided a Class A system for two areas used for initial hot shutdown.

The remainder of the plant is provided with a Class B early warning fire and smoke detection system designed in accordance with NFPA 72D. This includes two areas used for initial hot shutdown:

a. Control equipment mezzanine,
b. Main control room.

REPSONSE (Cont'd)

The control equipment mezzanine detection system is backed up by a separate fire detection and alarm system for the automatic CO_2 system and the control room is continuously manned, thus, providing the equivalent of a Class A system in both areas. The safety-related equipment areas with Class B systems are located outside the reactor building and are accessible during normal plant operation.

The Class B system is electrically supervised to detect circuit breaks, ground faults, removal of a detection device from a detector circuit, and power failure. If any of these problems occur, fire detection system trouble is annunciated locally and in the main control room. This would alert the operators to begin repairing the system immediately to bring it back to working order without jeopardizing plant safety.

Except for the differences stated in FSAR Section 9.5.1.6.15, the fire and smoke detectors are installed in accordance with NFPA 72E. Section 9.5.1.6.14 has been revised to clarify the extent of Class A and B systems at HCGS.

areas, switchgear rooms, and other areas where potential exists for heavy smoke conditions.

As stated in Section 9.5.1.1.15.1, smoke and corrosive gases are generally discharged to the outside using the normal ventilation system and a separate smoke removal system provided for the control area. Since the switchgear rooms and diesel generator control rooms, at floor elevation 130 feet, and the heating and ventilation equipment rooms, control equipment room and inverter rooms, at floor elevation 163 feet 6 inches of the diesel area, are not provided with direct exhaust to the outside, portable blowers will be provided for the fire brigade to use as smoke ejectors.

All battery rooms are provided with air exhaust systems that can discharge smoke and gases directly outside.

9.5.1.6.13 Paragraph C.5.g.(3)

Paragraph C.5.g.(3) states that a fixed emergency communication system independent of the normal plant communication system should be installed at preselected stations.

HCGS has no fixed emergency communication system that is solely provided for emergency situations. The plant has an intraplant, five channel, page-party, public-address (PA) intercom system. The in-plant communication system is provided with reliable uninterruptible power supply (UPS) for uninterrupted communiciations between all areas of the plant. At least one channel of the PA system will be assigned for emergency use. In addition, portable radio communication units will also be available during an emergency situation.

9.5.1.6.14 Paragraph C.6.a.(2)

Paragraph C.6.a.(2) requires that the fire detection systems comply with the requirements of Class A systems, as defined in NFPA 72D.

At HCGS, the reactor building is provided with a Class A fire detection system in accordance with NFPA 72D. The balance of the plant, however, is provided with a Class B detection system, which also complies with NFPA 72D. The Class B system is electrically supervised to detect circuit breaks, ground faults,

INSERT A

The areas outside the reactor building are the 250V DC RCIC and HPCI battery and inverter rooms at Elevation 54 ft., the cable spreading room at Elevation 77 ft., the control equipment room at Elevation 102 ft., the switchgear room B at elevation 130 ft. of the auxiliary building SDG area, and the remote shutdown panel room at elevation 137 ft. The Class A system will not prevent the detection system from initiating a fire alarm due to an occurance of a single break or single ground fault of the detection circuit between the local control panel and the associated detectors.

INSERT B

This includes two areas; the control equipment mezzanine and the main control room for initial hot shutdown. The control equipment mezzanine detection system is backed up by a separate fire detection and alarm system for the automatic CO_2 system, and the control room is continuously manned. The safety related equipment areas with Class B system are located outside the reactor building and are accessible during normal plant operation.

removal of a detection device from a detector circuits, and power failure. If any of the above problems occur, a fire detection system trouble is annunciated locally and in the main control room. Plant operation will periodically test the system for proper functioning, similar to inservice testing of other plant systems.

Insert C

At HCGS, the fire and smoke detection system is in compliance with NFPA 72D except that the operation and supervision of the system is not the sole function of the plant operator. The plant operator's duties cover operation of the generating station and monitoring and supervising the fire protection systems.

9.5.1.6.15 Paragraph C.6.a.(3)

Paragraph C.6.a.(3) requires that the fire detectors be installed in accordance with NFPA 72E.

At HCGS, the location of early warning fire and smoke detectors was determined and performed under the direction of a registered fire protection engineer. The location of the fire and smoke detectors complies with the guidelines of NFPA 72E except for the location of ionization and photoelectric detectors in high-bay areas. The detectors are not located in each bay formed by deep beams. NFPA 72E allows detector locations to be determined based on engineering judgement considering ceiling shape, ceiling surfaces, ceiling height, configuration of contents, combustible characteristic and ventilation.

At locations in areas where composite construction is used, the diffusion of combustion particulates throughout the compartment volume produced during the incipient and smoldering stages of the fire will negate the effect of beam depth and result in acceptable levels of detection coverage.

9.5.1.6.16 Paragraph C.6.a.(6)

Paragraph C.6.a.(6) requires primary and secondary supplies be provided for electrically operated control valves conforming to NFPA 72D.

DSER OPEN ITEM 158

INSERT C

This would alert the operators to begin repairing the system immediately to bring the early warning fire and smoke detection system back to working order without jeopardizing plant safety.

DSER OPEN ITEM 158

MP84 93 06 3-vw

DSER Open Item 174 (Section 13.5.2)

RESOLUTION EXPLANATION IN FSAR OF TMI-2 ITEMS I.C.7 and I.C.8

The FSAR should be revised to provide a brief explanation of how Task Action Items I.C.7 and I.C.8 have been resolved, as described in 13.5.2c herein.

RESPONSE

FSAR section 1.10, items I.C.7 and I.C.8, have been revised to provide the requested information.

• I.C.6 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations, and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Response

Verification of operating activities to provide a means of reducing human errors and to improve the quality of normal operations will be incorporated in Operations Department Administrative Procedure OP-AP.ZZ-020, Operations Management Audit Program (available March 1, 1985).

• I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

Position

Obtain nuclear system supply system vendor review of power-ascension and emergency operating procedures to further verify their adequacy.

Response

All startup test procedures from core load through power ascension will be reviewed by GE. This review, as well as vendor review of test results, will be documented prior to commercial operation.

INSERT A →

DSER OPEN ITEM 174

INSERT "A"

The HCGS Emergency Instructions will be developed based on the NRC - approved BWR Owners Group Emergency Procedure Guidelines (EPGs). Due to GE's involvement in the development of the EPGs, it has been determined that an additional NSSS vendor review of the plant specific Emergency Instructions is not necessary.

DSER OPEN ITEM 174

MP84 93 06 4-vw

• I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR-TERM OPERATING LICENSE APPLICANTS

Position

Correct emergency procedures as necessary based on the NRC audit of selected plant emergency operating procedures (e.g., small-break loss-of-coolant accident, loss of feedwater, restart of engineered safety features following a loss of ac power and steam-line break).

Response

INSERT B → delete

~~All emergency procedures will be written following the guidelines of INPO 82-017, Emergency Operating Procedure Writing Guideline, and the guidelines of the BWR Owners Group Procedure Committee, as long the the guidelines do not contradict existing NRC directives. These procedures will be available by March 1, 1985.~~ |

~~Corrections will be made, as necessary, based on any NRC audits of these procedures.~~

• I.D.1 CONTROL ROOM DESIGN REVIEWS

Position

Licensees and applications for operating licenses are required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Those applicants for operating licenses who are unable to complete this review prior to issuance of a license shall make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by us for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

DSER OPEN ITEM 174

INSERT "B"

A Procedure Generation Package (PGP) will be prepared in accordance with Supplement 1 to NUREG-0737. The PGP and plant specific Emergency Instructions will be based on the NRC - approved BWR Owners Group Emergency Procedure Guidelines (EPGs). As a result, it has been determined that a NRC review of the plant specific Emergency Instructions is not necessary.

DSER OPEN ITEM 174

MP84 93 06 1/5-vw